U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-237/89018(DRP); 50-249/89017(DRP)

Docket Nos. 50-237; 50-249 License Nos. DPR-19; DPR-25

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Facility Name: Dresden Nuclear Power Station, Units 2 and 3

Inspection At: Dresden Site, Morris, IL

Inspection Conducted: July 15 through August 28, 1989

Inspectors: S. G. Du Pont
D. E. Hills

R. M. Lerch

Approved By: R. M. Lerch, Acting Chief
Reactor Projects Section 1B

9-7-89
Date

Inspection Summary

Inspection during the period of July 15 through August 28, 1989 (Report Nos. 50-237/89018(DRP); 50-249/89017(DRP))
Areas Inspected: Routine unannounced resident inspection of previously identified items, licensee event reports followup, plant operations, maintenance and surveillances, radiological controls, safety assessment and quality verification, engineering/technical supports, and monthly report review.
Results:

- No violations were identified during this inspection period.
- Licensee actions taken in response to NRC concerns resulted in a significant decrease in open control room work requests.

DETAILS

1. Persons Contacted

Commonwealth Edison Company

*E. Eenigenburg, Station Manager
*L. Gerner, Technical Superintendent
E. Mantel, Services Director
C. Allen, Administrative Service Superintendent
D. Van Pelt, Assistant Superintendent - Maintenance
*J. Kotowski, Production Superintendent
J. Achterberg, Assistant Superintendent - Work Planning
G. Smith, Assistant Superintendent-Operations
*K. Peterman, Regulatory Assurance Supervisor
J. Williams, Operating Engineer
W. Pietryga, Operating Engineer
B. Zank, Operating Engineer
*M. Strait, Technical Staff Supervisor
L. Johnson, Q.C. Supervisor
J. Mayer, Station Security Administrator
D. Morey, Chemistry Services Supervisor
D. Saccomando, Health Physics Services Supervisor
E. Netzel, Q.A. Superintendent
*R. Falbo, Regulatory Assurance Group Leader
*K. Yates, Nuclear Safety Supervisor
*J. Harrington, Quality Assurance Inspector

The inspectors also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, and contract security personnel.

*Denotes those attending one or more exit interviews conducted informally at various times throughout the inspection period.

2. Previously Identified Inspection Items (92701 and 92702)

(Closed) Unresolved Item (237/89017-02): Review circumstances of May 1989 batch waste release tank composite sample for tritium and gross alpha that was inadvertently discarded before being sent for offsite analysis. This sample was discarded prior to analysis when a chemistry foreman gave a chemistry technician permission without verifying the samples to be discarded were the correct ones. The monthly composite sample was required by Dresden Technical Specification Table 4.8.3. However, due to other sampling, it can be ascertained that these samples, if analyzed, would have been within limits. Gaseous effluent tritium did not increase during May 1989. In addition, alpha and gamma activities were normal and tritium analysis on samples close to the same time frame were normal. Another event involving iodine and particulate samples not being analyzed within required time limits occurred during the month of July. As discussed in Paragraphs 6.b and 6.c of this report, however,

the intent of Technical Specifications were still met in this latter case and appropriate corrective actions were identified. Various corrective actions have been implemented or planned to provide better administrative control of samples due to the July 1989 event. Applicable chemistry personnel were advised of this event during tailgate meetings. In addition, a designated storage area for samples requiring offsite analysis was set up in the reagents lab. Dresden Chemistry Procedure (DCP) 1017-3, Radio Assay Shipments, was also under development. This procedure was to contain instructions for preparing shipments for offsite analysis and controls to ensure these samples are not discarded. In order to facilitate easier scheduling and tracking of required samples, the chemistry surveillance schedule was also being revised to be based on a 28 day cycle instead of a 31 day cycle. The discarding of this sample required by Technical Specifications was identified as a licensee identified violation (50-237/89018-01) and is considered closed in this report due to adequately completed or planned corrective actions meeting the criteria of 10 CFR 2 listed in paragraph 3 of this report.

(Closed) Open Items (249/86009-19 and 249/86012-38): These items are administratively closed based upon generic modification program corrections noted in the NRC Safety Systems Outage Modification Inspection (SSOMI) followup review (inspection report 50-249/88200) and a NRC Engineering and Technical Support Team inspection (inspection reports 50-237/88025 and 50-249/88027).

(Closed) Violation (249/86009-34 and Open Item (249/88200-03): During the NRC Safety Systems Outage Modification Inspection (SSOMI) review, concerns with motor operated valve (MOV) overload heater selections were identified as a violation in that incorrect heaters had been installed. A total of 116 MOVs (58 on each unit) were identified and replaced during the 1988 refueling outages on Units 2 and 3. The completion of the heater replacement closes all concerns with this item.

(Closed) Open Item (249/86012-05): This item is administratively closed due to duplication by Open Items 237/88025-02 and 249/88027-02 by the NRC Engineering and Technical Support Team inspection (inspection reports 50-237/88025 and 50-249/88027). Although items 237/88025-02 and 249/88027-02 are specific to modifications to the 250 VDC system, the inspection verified that this problem was system specific and not generic to the modification process. Based upon the inspection team's finding, this item is considered to be no longer generic in scope and will be tracked by the specific items which duplicated the root cause.

(Closed) Open Item (249/86012-60): During the SSOMI followup review (inspection report 50-249/88200), this item was reviewed and remained open pending implementation of a new administrative procedure for test control. Dresden Administrative Procedure DAP 9-11 was issued on August 2, 1989. DAP 9-11 controlled the identified concerns with test pre-requisites, ambiguous test step signoffs and test result review and approval. Based upon the inspector's review of the procedure, this item is considered to be closed.

(Closed) Violation (249/86012-67): During the SSOMI followup review (inspection report 50-249/88200), this item was reviewed and remained open based upon the licensee's review and documentation of deficiencies encountered during the performance of a 1985 core spray system and low pressure coolant injection system logic test. The original test failures included a safety related HFA relay (1530-133 BU) which was found to have been rebuilt in 1985. The relay failed during logic testing due to a jumper that was installed during the rebuilding activities (accomplished with the relay installed in the cabinet). Although the deficiency was not documented in the original test, several corrective actions to the root cause were accomplished. The logic test were performed successfully several times since 1985, procedures were implemented to control modification activities and similar problems have not reoccurred. Based upon these, this item is considered to be closed.

(Closed) Unresolved Item (249/88200-01): During the NRC SSOMI the high steam flow setpoint for the High Pressure Coolant Injection (HPCI) system did not appear to be correct for actual application. The concern was based upon a review of a General Electric (GE) design document, GEK-786. According to the calculation in GEK-786, at the maximum expected steam flow of 125,000 pounds per hour at 1125 pounds per square inch, the differential pressure would be 141.74 inches of water (design high steam flow setpoint). The actual setpoint used for the instrument was 145.5 inches of water. During the SSOMI review, the licensee provided the design document GEK-26901 which determined a setpoint of 150 inches of water. The actual setpoint (145.5) compared to GEK-26901 (150) was considered to be conservative. However, given the conflicting documents, the GE design specifications were considered questionable.

The licensee performed a special test of the HPCI Steam Flow and instrumentation to resolve the concern. The results of the test were reviewed by both the licensee's and GE's engineering. The results of the actual performance test were determined to be consistent with GE's recently issued Service Information Letter (SIL)-475 which agrees with design document GEK-26901. This item is considered to be closed.

(Closed) Open Item (249/88200-02): During the SSOMI review, a major root cause to many of the issues was determined to be a generic failure to adhere to existing procedures. The licensee implemented a corporate policy, applicable to all nuclear operations, establishing the requirement to adhere to procedures. Dresden also implemented an administrative procedure (DAP 9-11) providing guidance on adherence of procedures and training of all station personnel was accomplished. Additionally, the Quality Assurance department monitored the effectiveness of the training through observation of nuclear operations. This item is considered to be closed.

(Open) Open Item (249/89010-02): The licensee was to provide a written response describing planned corrective actions to ensure usage of the isolation condenser for extended time periods without offsite power would not result in radioactive releases. This response was provided in the letter from J. A. Silady to A. B. Davis dated July 21, 1989. Actions already taken included revision of appropriate operating procedures to

specify the undesirability of using contaminated demineralized water for shell side makeup and to include guidance for the order of preferred system usage. The letter also described several design changes under consideration as long term corrective actions. These included pre-charged or atmospheric tanks on the isolation condenser floor, motor driven pumps at grade level, or upgrading the present clean demineralizer makeup system. A status report was to be provided within two months of the date of the letter to identify the selected long-term improvement and status. Therefore, this item is to remain open pending receipt and review of that status report.

(Closed) Various Items: The following Unit 3 items are administratively closed due to duplication with Unit 2 items. These items are being tracked and reviewed under the applicable Unit 2 item numbers. For reference, the applicable Unit 2 items (Unit 2 item number) are listed adjacent to the closed Unit 3 items.

249/86017-01	(237/86015-01)
249/88014-04	(237/88012-04)
249/88014-05	(237/88012-05)
249/88018-21	(237/88017-21)
249/88018-28	(237/88018-28)
249/88018-30	(237/88018-30)

No violations or deviations were identified in this area.

3. Licensee Event Reports (LER) Followup (90712 and 92700)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished or planned in accordance with Technical Specifications.

(Closed) LER 237/89020: Potential Violation of Secondary Containment Integrity Due to Interlock Door Strike Failure. This event is discussed in paragraph 5.b.1 of this report.

(Closed) LER 237/89018: Auto Start of Standby Gas Treatment System Due to Spurious Ventilation Radiation Monitor Trip. On July 7, 1989, the Unit 2 Reactor Building Ventilation (RBV) System isolated and the Standby Gas Treatment System (SGTS) automatically started during a reactor building radiation monitor functional surveillance test. While the monitor indicator and trip unit was being pulled out of the control room panel, a nick in the insulation of a wire caused the exposed wire to contact the chassis. This resulted in the actuation. The RBV system and SGTS were returned to normal lineup. The exposed wire was wrapped with electrical tape pending permanent repair. As of the end of the inspection period, permanent repair had not been completed. In addition, Dresden Instrument Surveillance (DIS) 1700-7, Reactor Building Ventilation Radiation Monitor Functional Test, was to be revised to require checking the RBV radiation monitors for exposed wiring during

performance of the surveillance. Exposed wiring has not been a problem at Dresden in the past.

(Closed) LER 237/89019: Scram/Group I Isolation Due to Main Steam Line (MSL) Radiation Monitor Lockup and Spurious Steam Tunnel Temperature Trip. Upon transferring Unit 2 Reactor Protection System (RPS) Bus A from its reserve back to its normal power supply on July 12, 1989, during surveillance testing of the RPS Motor-Generator (MG) Set Electrical Protection Assemblies (EPA), MSL logarithmic radiation monitors (LRM) A and C momentarily lost power as would be expected. A corresponding half scram and half primary containment group I isolation occurred. However, MSL LRM A received a lockup unknown alarm preventing resetting of the half group I isolation and half scram signals. While in this condition, a spurious Channel B MSL tunnel area high temperature signal occurred completing the logic for a full group I primary containment isolation. The Main Steam Isolation Valve (MSIV) closure caused an automatic scram from 63% power.

The spurious MSL tunnel high temperature signal was attributed to setpoint drift of one of the temperature switches. All four switches associated with Channel B were replaced since it could not be determined which one had caused the action. All had drifted lower than their normal setpoint bands but were in compliance with Dresden Technical Specifications. The last calibration of these switches had occurred on February 5, 1989. MSL tunnel area temperature switch setpoint drift had been a recurring problem and several corrective actions had been previously implemented. These previous actions included revision of the calibration procedure, applying adhesive to the set screw and verifying the absence of capillary tube leakage. Following this event, diagnostic equipment was installed to monitor the new temperature switches such that temperature switches causing future actuations could be specifically identified. An engineering study was also to be performed to evaluate replacement with resistance temperature detectors and programmable logic controllers. The current switches were mechanical bulb and capillary type. By the end of the inspection period, a cost estimate to conduct the review had been received from the contractor.

The MSL LRM monitor which could not be reset, was replaced and sent to General Electric for further analysis. This analysis was directed at determining the root cause for Mode Unknown failures which prevented the monitors from being reset and previous low voltage power supply failures. Although the unit was designed to continue to operate with power interruption up to 20 milliseconds to accommodate automatic bus transfers, interruptions longer than this and less than several seconds could lead to incorrect initialization. An improved design which modifies the initialization control circuitry to correct this problem was being accomplished at Dresden. The failure of the low voltage power supplies was traced to a manufacturing error by the vendor of the power supply. A capacitor in the power supply control circuit was incorrectly installed which could lead to a power supply failure after operation over a period of a year or more. Replacement power supplies were in the process of being installed. Difficulty in resetting MSL LRMs also contributed to a reactor scram which occurred on March 30, 1989, as described in Inspection Report 50-237/89011; 50-249/89010. As a result

of that scram, steps were added to this surveillance procedure providing guidance on resetting a locked up MSL LRM.

(Closed) LER 249/89005: HPCI System Declared Inoperable Due to Discovery of Cable Terminal Blocks That Were Not Environmentally Qualified. The Unit 3 HPCI System was declared inoperable on April 12, 1989, when a licensee inspection discovered two environmentally unqualified cable terminal blocks in a cable pull box associated with the HPCI Steam Supply outboard isolation valve. This inspection was conducted in response to an NRC request to verify weep holes in various junction boxes. The unqualified terminal blocks were removed and replaced with taped splices utilizing an approved environmentally qualified taping procedure. The environmental qualification (EQ) program at Dresden was originally implemented by qualifying every identified EQ component type. The initial identification of EQ components was based on review of electrical drawings. These particular terminal blocks did not appear on the electrical drawings and, as such, were never identified for EQ purposes.

The initial EQ program only utilized partial system walkdowns and thus this problem was not identified during these walkdowns.

The licensee developed a justification for continued operation for terminal blocks without EQ documentation.

A walkdown inspection to document the as-built configuration for splices, terminal blocks and junction boxes was in progress. This included a walkdown from the sensing instrument until exit from the Reactor Building or entry into the cable tray system. Accessible equipment for both units was to be completed prior to the next Unit 3 refueling outage. Final physical verification of inaccessible EQ equipment was to be completed during the next refueling outage for each unit.

By the end of the inspection period, the walkdowns were well underway for Unit 3 equipment but had not yet been started for Unit 2. Several suspect terminal blocks were identified in the HPCI, Main Steam, Core Spray, Low Pressure Coolant Injection and Containment Atmospheric Monitoring (CAM) systems. These were undergoing engineering review to establish required actions.

The unqualified terminal blocks and subsequent corrective actions were already designated as an Unresolved Item (50-237/89010-01; 50-249/89009-01) in a previous inspection report.

(Closed) LER 237/89017: Loss of Batch Waste Release Tank Composite Sample Due to Management Deficiency. This event is discussed in paragraph 2 of this report as Unresolved Item 237/89017-02.

This LER was reviewed against the criteria of 10 CFR 2, Appendix C, and the incident described met all of the following requirements. Thus no Notice of Violation is being issued for this item.

a. The event was identified by the licensee,

b. The event was an incident that, according to the current enforcement policy, met the criteria for Severity Levels IV or V violations,

c. The event was appropriately reported,

d. The event was or will be corrected (including measures to prevent recurrence within a reasonable amount of time), and

e. the event was not a violation that could have been prevented by the licensee's corrective actions for a previous violation.

No violations or deviations were identified in this area.

4. Plant Operations (71707 and 93702)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the plant operations functional area.

b. Operational Events

On August 22, 1989, the operators were attempting to isolate a ground due to a 125V DC ground alarm on Unit 3. This was accomplished by opening various breakers in accordance with procedure and verifying whether the alarm condition remained. During the course of this activity, the reactor building ventilation system for both units was lost in accordance with the procedure. The operators were to restore the system in order to maintain a negative reactor building to atmosphere differential pressure and hence to maintain secondary containment. However, in trying to restore the system, the Center Desk Operator found that Unit 2 Reactor Building Exhaust Fan B would automatically trip whenever it was started. In addition, after starting Unit 3 Reactor Building Exhaust Ban B, smoke was reported coming from the fan. This was due to a bad bearing and thus the fan had to be tripped. These failures resulted in a difficulty in restoring the system with the remaining fans. The inspector observed that the Shift Control Room Engineer (SCRE) responded immediately and directly assisted the Center Desk Operator. The SCRE also informed the Shift Engineer (SE) who was in the control room at the time. The SE provided advice as to expected instrument response. In addition, the Unit 3 Control Room Operator relayed information from the field as to the status of Unit 3 Reactor Building Exhaust Fan B. As a result of these efforts, reactor building ventilation was restored despite the fan failures.

c. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

The approach to identification and resolution of technical issues associated with plant operations during this inspection period demonstrated very good initiative on the part of the operations department to ensure safety and a genuine concern for proper functioning of equipment. Examples of the operators' and management's approach from a safety standpoint are discussed below including one example showing excellent attention to detail by the

operators and their willingness to escalate concerns to a level where appropriate actions could taken.

On July 26, 1989, the operators observed what they believed to be uncharacteristic readings for reactor building differential pressure. The reactor building fans were operated in various combinations to study their effect on differential pressure readings both locally and in the control room. The SGTS was also started and operated for 15 minutes during which time differential pressure readings were taken. These results showed difficulty in achieving the negative 0.25 inches differential pressure required by Technical Specifications. However, these readings were taken under conditions different from those established during a normal secondary containment leak rate test.

Most notably, data was taken for only 15 minutes instead of the usual one hour allowed for stabilization. The licensee believed that the readings obtained were representative of the first 15 minutes of previous leak rate tests and, in that regard, reviewed previous test results to verify that belief. The licensee also conducted a walkdown of the reactor building to ensure that no obvious leaks existed. The review of the previous test results found that only the final reading at one hour was required to be officially recorded for acceptance criteria. Although some unofficial data supporting the licensee's beliefs was found in the files, the licensee decided that unofficial data was insufficient evidence on which to judge acceptability of observed differential pressure values. As such, the licensee conducted a subsequent secondary containment leak rate test which showed a negative 0.255 differential pressure was achievable.

Other examples include the actions taken by the operators on July 19, 1989 (as described in paragraph 5.b.1 of this report) to ensure maintenance of secondary containment upon discovery of the failed latch on the Unit 2/3 diesel engine room door and the actions taken by the operators on August 22, 1989 to restore reactor building ventilation (as described in paragraph 4.b above) which resulted in expeditious maintenance of secondary containment without having to rely upon use of the SGTS. The response of the shift crew observed by the inspector for this latter event showed excellent usage of the team concept in that various crew members were automatically supported in their activities to mitigate the event.

d. Assurance of Quality, Including Management Involvement and Control

Management involvement and control in assuring quality remained excellent. An example includes the various actions taken in regard to the reactor building differential pressure problems on July 22, 1989 (as described in paragraph 4.c above) which indicated an aggressive attitude of management toward resolution of safety issues and a conservative approach in regard to safe plant operations.

During the inspection period, the inspectors continued to observe frequent visits of licensee management to the control room and other areas of the plant. Thus, licensee management was observed to be

knowledgeable of plant status and to possess a keen interest in normal, as well as abnormal, plant activities.

e. Responsiveness to NRC Initiatives

The licensee continued to alleviate previous NRC concerns by ensuring operators were trained on Technical Specification changes prior to their implementation. This was evidenced by actions taken prior to implementing Technical Specification changes regarding required equipment surveillances. The inspectors were notified of this implementation on August 23, 1989.

f. Observation of Operations

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during this period. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of Units 2 and 3 reactor buildings and turbine buildings were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. During this inspection period, no violations of the fire protection program were observed.

The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors reviewed new procedures and changes to procedures that were implemented during the inspection period. The review consisted of a verification for accuracy, correctness, and compliance with regulatory requirements.

The inspectors also witnessed portions of the radioactive waste system controls associated with radwaste shipments and barreling.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under Technical Specifications, 10 CFR, and administrative procedures.

No violations or deviations were identified in this area.

5. Maintenance and Surveillance (62703, 61726 and 93702)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the maintenance and surveillance functional area.

b. Operational Events

Various maintenance activities associated with the following events were observed/reviewed for which the inspectors verified items listed in paragraph 5.e below:

(1) On July 19, 1989, the licensee discovered that the latch on the Unit 2/3 diesel generator room door had failed. As a result, opening the door from the reactor building into the Unit 2/3 diesel generator interlock corridor would cause the Unit 2/3 diesel generator room door to open slightly due to normal Reactor Building to Turbine Building differential pressure. The definition of secondary containment integrity contained in Dresden Technical Specifications included maintaining at least one door in each personnel interlock in the closed position. However, both doors were open simultaneously only briefly and the licensee took immediate action by posting an individual in the area to maintain one of the interlock doors closed. The failed latch mechanism was then replaced. Thus, the reactor building to atmosphere differential pressure was maintained greater than 0.25 inches of water vacuum at all times.

(2) On July 22, 1989, with Unit 3 at 91% rated thermal power, the licensee declared an Unusual Event due to the failure of a surveillance test conducted for the main turbine stop valve scram function from the No. 2 main turbine stop valve. The scram function from all other main turbine stop valves worked correctly. The closure of any stop valve provided a signal to both channels of RPS. The logic was arranged such that the closure of any two stop valves would cause a half scram and the closure of any three stop valves would cause a full scram. Since both channels of RPS were affected, technical specifications required the insertion of all rods within four hours when greater than 45% power. The licensee reduced power to less than 45% under which this scram function was not required to be operable. The Unusual Event was declared due to the power reduction while in a technical specification required forced shutdown. The Unusual Event was terminated when power was reduced below 45%. Subsequent investigation showed that a limit switch for the No. 2 main turbine stop valve required mechanical adjustment. Following completion of this activity, the function was satisfactorily tested and power was again increased later that same day.

(3) On August 7, 1989, while performing the Low Pressure Coolant Injection (LPCI) System quarterly flow rate surveillance test, the flowrate through the B test loop was below the Technical Specification acceptance criteria. Various testing and troubleshooting activities were performed. While this activity was occurring, the flowrate returned to acceptable limits for

unexplainable reasons. Based upon the change to an acceptable flowrate and the troubleshooting activities, the licensee considered the system operable. Had the system been declared inoperable, a seven day limiting condition for operation (LCO) would have been entered. On August 21, 1989, the Unit 2 diesel generator was taken out of service to perform the semi-annual inspection. The diesel fuel oil tank was also drained for cleaning. This placed the unit into a seven day LCO. On August 22, 1989, additional troubleshooting activities on LPCI loop B determined that the subsystem was actually inoperable due to a suspected stem to disk separation on the LPCI Outboard Injection Valve 2-1501-21B. Since the Unit 2 diesel generator was previously made inoperable, this placed the unit into a 24 hour shutdown action statement. The licensee reassembled the diesel generator, filled the fuel oil storage tank and successfully tested the diesel generator later that same day. Thus, the 24 hour action statement was exited and the unit shutdown did not have to commence. Repairs to the LPCI Outboard Injection Valve, which placed the unit into a seven day LCO, were completed on August 28, 1989. This event is considered an Unresolved Item (50-237/89018-02) pending completion of a review of the activities which originally determined the valve to be operable, those that eventually determined it to be inoperable and an analysis of the failure cause.

(4) On August 8, 1989, with Unit 2 at 64% rated thermal power, a power decrease was begun to place the unit in hot standby. This was necessary to investigate oscillations occurring on the No. 1 main turbine control valve. The mode switch was placed to startup with plans to close the main steam isolation valves to allow work to commence on the No. 1 main turbine control valve. On August 9, 1989, while pulling fuses for the main steam isolation valve outage a wire lug for one of the fuse holders broke in a control room panel. This resulted in an isolation of the isolation condenser which was an engineered safety features (ESF). The wire was repaired and the system placed back to normal standby lineup. Investigation of the No. 1 main turbine control valve oscillations discovered a loose wire crimp in a junction box for the position/velocity feedback transducer that provides a signal to the electro-hydraulic control system. Following repair, the unit was placed back on line on August 10, 1989.

(5) On August 13, 1989, Unit 3 experienced a trip of one of its seven drywell coolers which could not be restarted. Various upper elevation drywell temperatures were observed to have later trended upward. The most limiting increase was the 1B main steamline air pilot valve temperature which was required by Technical Specifications to remain below 170 degrees F. This temperature increased from 140 to 165 degrees F. Due to the elevated drywell temperatures, reactor power was reduced to less than 40% to facilitate a drywell entry on August 16,

1989. The inoperable drywell cooler shared a common discharge path with another drywell cooler at that elevation. The affected drywell cooler's discharge damper was shut locally to preclude an inappropriate flowpath from the operable cooler back through the inoperable drywell cooler. In addition, the Reactor Building Closed Cooling Water inlet valve to the inoperable drywell cooler was closed in order to provide more cooling flow to the other drywell coolers. The area was also checked for possible leaks that could cause a reduction in drywell cooling capability and the inoperable drywell cooler was examined in an attempt to determine the cause of its inoperability. Nothing in this regard was noted. While in the drywell, Inboard MSIV 1C timing was also adjusted due to an excessive closure time obtained during a surveillance test. Although these actions may have helped somewhat to alleviate temperatures in various upper elevation areas the 1B Main Steam Line air pilot valve temperature still peaked at about 165 degrees F. Repairing an additional service water pump and placing it in service also did not lower this temperature. The licensee increased monitoring of this indication and requested operators to contact the Operating Engineer if it reached 169 degrees F. The licensee and inspectors are continuing to monitor the drywell temperatures.

(6) On August 27, 1989, the Unit 2 HPCI System was declared inoperable due to discovery of broken fan belts on the HPCI room cooler. LPCI Subsystem B had previously been declared inoperable as described in paragraph 5.b.3 above. Dresden Technical Specifications thus required reactor pressure to be reduced to 150 psig within 24 hours. The room cooler was repaired in sufficient time such that a reactor shutdown did not commence.

c. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

Maintenance related activities described above and elsewhere in this report indicated an good approach to resolution of technical issues. The following are of particular interest:

The licensee review of maintenance and system history records to identify recurring problems with both MSL tunnel temperature switches and MSL LRMs (as described in paragraph 3 of this report) showed excellent utilization of these resources. As a result, a review of potential modifications to replace the temperature switches with upgraded equipment was planned and an analysis of MSL LRM failures resulted in identification of common control circuitry problems and a manufacturing error in power supplies. As such, this type of equipment history review and subsequent actions can contribute greatly to the long term dependability of plant equipment and reduce adverse occurrences.

Other examples included; placing Unit 2 into hot standby on August 8, 1989, to investigate main turbine control valve

oscillations which indicated a commitment to provide for quality operations and a reduction in potential adverse events.

The power reduction and drywell entry of August 16, 1989, to address elevated drywell temperature was also indicative of a conservative approach toward Technical Specification requirements. This action was conducted although the Technical Specification limit had not yet been reached and the action statement for this limit required only a report to the NRC within seven days. In addition, the licensee indicated that the pilot valves to which the Technical Specification referred had been changed and the present pilot valves could probably justify a higher temperature. The review of Technical Specifications had revealed that a change had not been made to reflect this. The licensee is in the process of submitting a Technical Specification change to reflect the actual pilot valve design.

Finally, the various activities undertaken to analyze Unit 3 Control Rod J-13 slow scram times (as described in paragraph 5.e below) showed a concern for doing whatever was possible to fix the problem as opposed to leaving the rod inoperable and inserted which was allowable.

These examples indicated a general improvement in Dresden's approach with maintenance indicating a personal responsibility to identify and mitigate problems before they exceed Technical Specification requirements.

d. Assurance of Quality, Including Management Involvement and Control

Management involvement and control in assuring quality in regard to maintenance and operability of equipment continues to improve. The inspectors observed substantial management involvement in the decisions to conduct the activities described in paragraph 5.c above.

e. Observation of Maintenance and Surveillance Activities

The inspectors observed surveillance testing required by technical specifications for the items listed below and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors witnessed portions of the following test activities pertaining to Units 2 and/or 3:

Secondary Containment Leak Rate Test
HPCI System Operability Verification
Average Power Range Monitor (APRM) Flow Biased Scram
Rod Block and Downscale Calibration

Nuclear Engineer's Checklist
Full Closure Timing of MSIVs

In addition, the inspectors witnessed conduct of a special test to investigate slow scram times received during a surveillance test on Unit 3 Control Rod J-13. This test consisted of notching in and out of the rod to position 08, including friction testing and stall flow testing. An analysis had been previously conducted to ensure adequate shutdown margin would exist during the testing. Nothing abnormal was noted during the testing. Radiographic examination of the associated hydraulic control unit valves also failed to indicate any problems. The licensee is proceeding with assistance from engineering to evaluate the effects on fuel burnout by maintaining the control rod inoperable until the scheduled refueling outage in December 1989. A review of Technical Specifications and the design basis indicated that no limits or requirements would be exceeded with the control rod inoperable.

Station maintenance activities of systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review:

The limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented. Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The inspectors observed work completed in accordance with Modification Design Package M12-2/3-88-28, Unit 2/3 Reactor Building Roof Replacement, and associated work requests to ensure establishment of controls as stated above, as well as special controls described in Paragraph 8.c of this report. During replacement of the first of 12 roof slabs, the licensee discovered that two of the four clips that held the slab in place could not be installed due to their position in relation to the building girder arrangement. This problem affected approximately five of the slabs that needed replacement. As a result, a field change request was used to incorporate a new design to attach the slabs to the girders.

The inspectors also observed the installation of temporary area radiation monitors in the radwaste building. These were to

facilitate closer monitoring of workers during future upgrade modifications to the radwaste systems.

No violations or deviations were identified in this area.

6. Radiological Controls (93702)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the radiological controls functional area.

b. Operational Events

On July 21, 1989, the 2/3 main chimney particulate and iodine filters were removed from the 2/3 main chimney SPING monitor and delivered to the counting room. However, the analysis of the samples was not performed until July 24, 1989, contrary to the 48 hour limitation contained in DCP 1700-8, Particulate and Iodine Filter Change For The SPINGS. This requirement was obtained from Technical Specification Table 4.8.1 which required samples to be analyzed at least once per seven days and the analysis to be completed within 48 hours after removal from the sampler. This second requirement was to ensure that the more short-lived radionuclides did not decay appreciably prior to being analyzed and, thus, to ensure that the samples would be indicative of conditions at the time they were removed. The samples in question, however, were removed after only two days, instead of the normal seven days, due to maintenance activities on the SPING monitor. Thus, any decay of shorter-lived radionuclides caused by the analysis delay would have occurred in a seven day sample. Samples were provided from a backup General Electric monitor for the remaining portion of the seven days. The results of both sets of samples combined were within acceptable limits. Thus, in this particular case, the quality of sampling data was not reduced by the delay from that which would have occurred with the normal sampling frequency. Therefore, the intent of the technical specification was met.

c. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

The licensee's root cause analysis and corrective actions to the 2/3 Main Chimney SPING iodine and particulate samples which were not analyzed within the allowable timeframe were adequate. The samples were not routine and were removed to support the quarterly functional test of the SPINGs. The licensee determined that there was a lack of administrative controls involving non-routine samples. Since this was not a scheduled activity, the chemistry surveillance sheets for the chemistry technician assigned to collect the daily plant samples and the chemistry technician assigned to the counting room did not reflect this activity. In addition, there was no designated location for samples to be dropped off that needed analysis, although it was a general understanding to place them on a desk in the counting room. This area was not checked for additional

samples by the counting room technician. As a result of these weaknesses, the licensee developed the following corrective actions:

- Formally designate a sample drop off area in the counting room and require routine inspection of this area by the counting room chemistry technician.

- Revise DCP 1700-8 to include a checklist that must be given to the counting room chemistry technician with the sample which requires a manual entry to be made on the daily surveillance sheets.

- Add an entry on the daily surveillance sheets to check the air sample drop window and designated sample drop off area at the end of each shift.

- Coordinate with the instrument maintenance department to schedule SPING surveillances at the same time as the chemistry sample pull schedule.

- Discuss the event with chemistry department personnel during tailgate sessions.

d. Assurance of Quality, Including Management Involvement and Control

The inspectors reviewed revision 4 to the project plan for the chemistry department procedures upgrade project. The inspectors regarded this project plan as an excellent management tool which made it easy to determine and track program status. This document described and provided a schedule and cost estimates for the chemistry department portion of the overall station procedures upgrade program. The entire program including the chemistry department portion was scheduled for completion by February 1992. A procedure writer's guide had been developed which specified procedure formats, technical content, writing style, guidelines for procedure graphics and procedure content specifics for station procedures. Of the ten phases identified for the chemistry department in this project, the first two consisting of identification of departmental commitments and revision of chemistry administrative procedures had been completed on schedule. Phase 3 consisting of Dresden Chemistry Surveillance (DCS) development had commenced but was not yet complete. The remaining portion of the project involving revision of the DCPs and revision of the Dresden Station Building Procedures (DSBP) were scheduled for various future dates.

No violations or deviations were identified in this area.

7. Safety Assessment/Quality Verification (35502 and 40500)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the safety assessment/quality verification functional area.

b. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

Various administrative programs and procedures reviewed by the inspectors during the inspection period were considered by the inspectors to be beneficial toward the resolution of technical issues as described below:

As a result of previous contractor control problems identified by both the NRC and licensee quality assurance audits, the licensee issued a revision to DAP 1-4, Station Contractor Control. These changes strengthened the over-site responsibilities of Engineering and Construction (ENC). ENC personnel were to remain cognizant of the number of contractor employees permitted on site at any one time and to discuss proposed increases above the limit with the Station Manager. In addition, ENC personnel were to ensure equipment was properly tagged out of service prior to contractors under ENC control commencing work. ENC field engineers were also to attend station planning meetings. Finally, the procedure revision required the contractor site managers to be informed of their obligations to comply with various laws, rules, regulations and other requirements, as well as requirements that they inform the licensee of any possibility of an employee discrimination claim for engaging in protected activities.

Actions involving another licensee program were observed on August 22, 1989, during which a half group II isolation signal resulted from a spiking drywell radiation monitor. As a result, the inspector observed the SCRE obtain the event and data check-list form as required by the scram/ESF actuation investigation program. This program provided direction for investigation into the cause of scrams, ESF actuations or near misses (including unplanned half scrams or half group isolations). Specific detailed criteria was established for data collection by the crew and the event investigation, root cause determination and restart recommendations, if appropriate, by the Scram/ESF Investigation Sub-Committee. For scrams and ESF actuation events, a final investigation report was required which was to contain a detailed event description, root cause evaluation, plant transient response assessment, recommended corrective actions and supporting plant data, participant statements and exhibits.

The scram/ESF actuation investigation program was actually one aspect of the scram/ESF actuation reduction program. In addition to incident investigation and implementation of corrective actions, this program provided for review of operating practices, procedures

and operating training relative to scram reduction on a continuing basis. The review of industry and station scram data information was also included as well as development and implementation of design changes to be performed in the area of scram and ESF prevention and reduction. The program also provided for generation of additional surveillance, work procedures and corrective measures as it pertains to maintenance, equipment reliability and component functionality. Nuclear Plant Reliability Data System and maintenance/component history records were to be accessed in this effort.

The inspectors considered this as an overall beneficial program to address the root cause and implement corrective actions to reduce adverse events. Most notable was the decision to include even near misses in the program. Examples of these corrective actions are given in paragraph 5.c of this report.

c. Assurance of Quality, Including Management Involvement and Control

During the inspection period, the inspectors accompanied the Station Manager, the Production Superintendent and the Assistant Superintendent Operations at various times on their plant rounds.

They were thorough in their observations and exhibited an excellent attitude toward equipment condition and plant cleanliness controls.

The inspectors reviewed the monthly plant status report for the month of July. The inspectors found this to be an excellent management tool for remaining cognizant and identifying trends in various departmental indicators.

d. Responsiveness to NRC Initiatives

Responsiveness to NRC concerns remained adequate. As explained in Inspection Reports 50-010/89002; 50-237/89017; and 50-249/89016, the inspectors had expressed a concern to the licensee about an upward trend in outstanding control room work requests. Response to the initial concern was slow. However, once addressed by the licensee, the issue was adequately resolved. In support of efforts to resolve this concern, Quality Assurance performed a surveillance on control room work requests. At the time of this review, there were approximately 136 work requests (outage and non-outage) open on control room equipment of which 57 were determined to be mis-classified. Instead of being identified as control room corrective maintenance, the work analysts had classified these as routine corrective maintenance. This resulted in a lower prioritization and an average number of days open of 65 versus 45 for correctly classified non-outage control room work requests. Both of these average completion times were considered excessive by the review. The review of control room work requests for the previous 12 months indicated an average completion time of 18 days. In addition, it was found that 34 of the control room work requests did not have work request stickers affixed to the affected equipment in the control room. This could impede the operator's ability to ascertain the status of equipment.

The licensee took several actions to address this problem. A policy directive was issued to provide better criteria for determining a control room work request classification. In addition, operations personnel were instructed to indicate at the top of the work request any they believed to be classified as control room (i.e. any for which they attached a work request sticker to control room equipment). An Operations Engineer also reassigned the priority to all open work requests classified as control room per the policy directive. As a result of these actions, the number of open control room work requests decreased significantly. The number of non-outage control room work requests dropped about half to approximately 20 during the month of July.

No violations or deviations were identified in this area.

8. Engineering/Technical Support (37828)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the engineering/technical support functional area.

b. Operational Events

On August 11, 1989, the licensee, as a result of a self initiated design review, informed the NRC of design deficiencies affecting the Atmospheric Containment Atmosphere Dilution/Containment Atmosphere Monitoring (ACAD/CAM) System. This system was provided to assure that containment integrity was not endangered due to a postulated ignition and combustion of flammable gas mixtures following a major loss of coolant accident. Containment radiation and hydrogen levels were monitored and control of postulated gas concentrations was accomplished by dilution of the evolved hydrogen with air.

The ACAD portion of the system consisted of one train for each unit with cross-ties to the opposite unit to provide for redundancy. However, the emergency electrical bus that supplied power to the Unit 2 subsystem also supplied power to the cross-tie isolation valves from the Unit 3 subsystem. These valves failed in the closed position. Thus, a loss of the Unit 2 diesel generator would render both trains of ACAD inoperable for use on Unit 2, if needed during a loss of offsite power coincident with a loss of coolant accident (LOCA). Under this scenario, manual operator action would be required to restore the ACAD system through use of a 4KV breaker cross-tie. This problem did not exist on the opposite unit.

A further deficiency in the CAM portion of the system was also identified. One of the two CAM channels on each unit was supplied power from a non-emergency bus which would be load shed when required during a loss of offsite power coincident with a LOCA. In this case, if that unit's diesel generator were to fail, both CAM channels would be inoperable until manual operator action were

taken to restore power to these instruments. Investigation and resolution to the ACAD/CAM design deficiencies identified by the licensee is considered an unresolved item pending NRR and RIII review for compliance with 10 CFR 50.44. (50-237/89018-03.)

c. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

The various responses to engineering technical issues observed by the inspectors were mixed as to adequacy of the approach.

Following identification of the ACAD/CAM design deficiencies, the licensee instituted temporary procedure changes to instruct operators to restore power to allow use of the ACAD/CAM system when needed. The issue of ACAD/CAM system design adequacy at Dresden in regard to NUREG 0737 requirements is currently an issue in the Office of Nuclear Reactor Regulation (NRR).

As described in paragraph 5.c of this report, the MSIV pilot valve temperature limits in Technical Specifications were not appropriate to the current Dresden design. These pilot valves had been changed such that the licensee believed a higher temperature could be justified and yet a corresponding Technical Specification change had not been previously pursued. As a result, licensee management and operator attention was needlessly diverted to address an administrative requirement that was not technically applicable.

The inspectors also reviewed Modification Design Package M12-2/3-88-28, Unit 2/3 Reactor Building Roof Replacement which the inspectors considered to be well planned (except for the problem with the panel clips described in paragraph 5.e of this report.) Special precautions developed for this modification were indicative of a good approach to safety. This modification covered replacement of the existing exterior roofing materials consisting of fiberboard insulation, asphalt, felt and gravel with a layer of insulation and a single-ply elasfomeric membrane fabric ballasted by roofblock paver blocks. The licensee believed that this would improve the ability to maintain the negative reactor building differential pressure required by technical specifications. During a pre-job inspection of the interior of the roof panels, damage to several of these precast reinforced concrete panels was discovered. These panels existed under the exterior roofing materials. Thus, several of these panels were to be either replaced, coated with epoxy or grouted as repairs through work requests in addition to the work covered under the modification.

Specific controls were established to ensure maintenance of secondary containment and protection of the fuel pools during construction activities. These controls included the following:

- Roofers were given tailgate training on the importance of secondary containment.

- Operations permission was received prior to starting work each day.

- A controlled strip-off technique was used such that any portion of the roofing material removed was replaced that same day. The roof was also made weathertight before stopping work each day.

- Special Procedure, SP 89-6-67, Reactor Building Roof Replacement Communication Guidelines for Operations and Construction, was written to ensure proper personnel were contacted in the event of a problem and that the roofers remained in contact with the control room via radio.

- Temporary seal kits of tar, plywood and plastic sheeting were available.

- Allowable load densities were pre-determined and specific criteria established to ensure that roof panels were not overstressed with equipment or materials.

- Outlines of the fuel pools and roof panels to be replaced were painted on the reactor building roof exterior and roofers were given instructions to avoid trafficking in those areas.

- Tarps were used over the fuel pools to keep out small debris.

- A debris net capable of securing a falling roof panel was suspended over the fuel pools.

- Special Procedure, SP 89-6-66, Reactor Building Roof Replacement Differential Pressure Daily Construction Test, was written to determine the reactor building differential pressure from local indications twice daily. This was in addition to existing requirements for the control room panel indication to be recorded in the center desk operator's daily surveillance log on a shiftly basis.

- A portable structure was built to be placed over each roof panel as it was replaced. This structure was tested to ensure a sufficient seal to maintain the reactor building negative differential pressure.

The inspector reviewed the modification package to ensure that appropriate procedures were written, the roof loading analysis was conducted with specific criteria established and required reviews and walkdowns were conducted. In addition, post-modification testing requirements were established.

d. Responsiveness to NRC Initiatives

The licensee continued to be responsive to the NRC in the engineering/technical support area as described below:

The licensee self initiated review which discovered the ACAD/CAM design deficiencies was the result of previously identified NRC concerns. These concerns dealt with electrical supplies to shared

systems. This review was conducted to verify whether additional problems existed in this area.

The licensee was also responsive to a regional request to provide information on the check valve testing program.

No violations or deviations were identified in this area.

9. Report Review (90713)

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for July. The inspectors confirmed that the information provided met the requirements of Technical Specification 6.6.A.3 and Regulatory Guide 1.16.

10. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, or items of non-compliance or deviations. Unresolved items disclosed during the inspection are discussed in paragraphs 5.b and 8.b of this report.

11. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on August 28, 1989, and informally throughout the inspection period, and summarized the scope and findings of the inspection activities.

The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary. The licensee acknowledged the findings of the inspection.