U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/94003(DRS); No. 50-249/94003(DRS)

Docket Nos. 50-237; 50-249 Licenses No. DPR-19; No. DPR-25

Licensee: Commonwealth Edison Company
Executive Towers West III
1400 Opus Place - Suite 300
Downers Grove, IL 60515

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

Inspection At: Dresden Nuclear Power Station, Dresden, IL

Inspection Conducted: February 28 through April 5, 1994

Inspectors: H. A. Walker — 5/5/94
H. A. Walker, Team Leader — Date

W D Shafer for — 5/6/94
R. A. Langstaff — Date

W D Shafer for — 5/6/94
C. L. Vanderniet — Date

W D Shafer for — 5/6/94
I. T. Yin — Date

NRC Consultants: J. T. Haller (Parameter, Inc.)
D. C. Prevatte (Powerdyne Corporation)

Observer: L. A. Love-Tedjoutomo (Atomic Control Board, Canada)

Approved By: W D Shafer — 5/6/94
W. D. Shafer, Chief — Date
Maintenance and Outages Section

Inspection Summary

Inspection conducted February 28 through April 5, 1994 (Reports No. 50-237/94003(DRS); No. 50-249/94003(DRS))
Areas Inspected: An announced team inspection of engineering and technical support and related management activities. The inspection was conducted

utilizing portions of inspection procedures 37700, 92701, 92702, and 92720 and draft inspection procedure 37550 to ascertain whether engineering and technical support was effectively accomplished and assessed by the licensee. Results: Based on the items inspected, overall performance in engineering and technical support was considered acceptable. The level, quality, and timeliness of engineering and technical support for the plant appeared to be acceptable. Most of the individuals contacted were knowledgeable and motivated and displayed a strong sense of ownership in their areas of responsibility, however, there were some exceptions as noted in this report.

Significant improvements were noted in the Site Engineering and Construction Organization, however, there were several findings where engineering activities had not been thorough and the necessary attention to detail was lacking. Little improvement was evident in the Systems Engineering Organization.

The most significant weakness appeared to be in the control of temporary alterations and lack of compliance to plant procedures. The most significant strength appeared to be the Site Engineering and Construction Organization management's motivation and commitment to improvement.

Two violations, one with two examples and one with three examples, were identified. Violations included weaknesses in design control and a lack of compliance to procedures.

DETAILS

1.0 Principal Persons Contacted

Commonwealth Edison Company (CECo)

* H. Massin, Site Engineering Construction Manager
* G. Spedl, Station Manager
* S. Elderidge, Modification Coordinator
* R. Jackson, System Engineering Team Leader
* R. Robey, Site Quality Verification Director
* J. Shields, Regulatory Assurance Supervisor
* J. Smentek, Site Engineering Construction Engineer
* D. Spencer, Lead Electrical - Plant Support
* M. Strait, System Engineering Supervisor
* J. Williams, Site Engineering Construction Supervisor
* R. Wroblewski, NRC Coordinator

U. S. Nuclear Regulatory Commission

* R. Crlenjak, Acting Deputy Director DRS
* M. Leach, Senior Resident Inspector
* P. Hiland, Chief, Projects Section 1B
  C. Phillips, Resident Inspector
* W. Shafer, Chief, Maintenance and Outage Section
  A. Stone, Resident Inspector

* Denotes those present at the exit meeting on April 5, 1994.

Other persons were contacted as a matter of course during the inspection.

2.0 Licensee Action on Previous Inspection Findings

A number of problems or concerns identified in past NRC inspections were reviewed for appropriate corrective actions. The items reviewed and the inspectors' evaluations of the actions to address these issues are discussed in this section.

2.1 (Closed) Unresolved Item (237/249/91025-02) -- Procedures were not followed in providing fire watch coverage. Licensee personnel indicated that fire watch coverage was not required since the fire barriers involved were non-rated barriers located in the same fire area. Procedure DFPP 4175-01, "Fire Barrier Integrity and Maintenance," was revised to clarify rated and non-rated fire barriers. This item is closed.

2.2 (Closed) Inspection Follow-up Item (237/249/93008-01) -- Evaluation to examine the possibility of conducting a temperature effectiveness test on the LPCI/CCSW heat exchangers. Plans had been made to conduct a temperature effectiveness test on the LPCI/CCSW heat exchangers; however, due to the difficulties involved, the tests might not be conclusive. The present method of performing preventive maintenance on

these exchangers appeared to be sufficient to assure proper operation. This item is closed.

2.3 (Closed) Inspection Follow-up Item (237/249/93008-03) -- Expanded acceptance criteria, used in the testing of the emergency diesel generator cooling water pump, did not meet ASME code requirements. Revision 11 of procedure DOS 6600-08 was issued to eliminate acceptance criteria for the diesel generator cooling water pump flows. The acceptance criteria was revised to be consistent with ASME code requirements and was included in table 10 of the Dresden In-Service Acceptance Criteria Manual. This item is closed.

2.4 (Closed) Inspection Follow-up Item (237/93015-01) -- There was no method or criteria for identifying excessive stem nut wear on motor operated valves. Although there was still no method or criteria for measuring excessive stem nut wear, discussions with licensee personnel indicated that maintenance personnel were aware of the stem nut wear problem and the stem nut was visually inspected in place for signs of excessive wear during normal MOV preventive maintenance activities. In addition, a search was underway for a reliable and economical method for measuring stem nut wear including acceptance criteria.

2.5 (Closed) Inspection Followup Item (237/249/93020-09) -- Root cause investigation and corrective actions for spurious group V primary containment isolations due to flow spiking. This item is closed based on the review associated with licensee event report (LER) 237/92-45 supplement 2.

2.6 (Open) Unresolved Item (237/249/93030-01(DRS)) -- Acceptability of using sampling inspection for quality control (QC) hold points. Random sampling was used to verify QC hold points for inspections required by ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants," 1989, and ANSI/ASME NQA-2, "Quality Assurance Requirements for Nuclear Facility Applications," 1989. On December 3, 1993, licensee personnel requested an interpretation from the American Society of Mechanical Engineers (ASME) concerning the use of sampling for required inspection activities. No interpretation had yet been provided at the time of this inspection. This item will remain open pending receipt and review of the interpretation provided by ASME.

2.7 (Open) Violation 237/249/93030-02(DRS) -- The procedure, which controlled the use of measuring and test equipment (M&TE), did not provide appropriate qualitative or quantitative criteria for performing evaluations of tested and inspected equipment when measuring and testing equipment (M&TE) were determined to be lost or were found to be out of calibration. An action plan had been developed to provide necessary improvements in the M&TE area. Many of the actions, required by the plan, had not been implemented. At the time of the inspection, Procedure DAP 11-22, "Control of Measuring and Test Equipment," had not been revised and the review of instrument evaluations back to January, 1993, had not been completed. This violation will remain open pending additional review when commitments have been completed.

2.8 (Closed) Violation (237/93030-03) -- Measures were not established to assure that deficiencies and deviations identified by contract quality

control inspectors on observation reports were corrected. Licensee personnel reviewed observation reports initiated in 1993 and initiated problem identification forms (PIFs) for observation reports which had not been resolved. Licensee and contractor personnel began using PIFs to document problems identified by contractors. The inspectors interviewed contract QC inspectors and confirmed PIFs were being used in lieu of observation reports. This violation is closed.

2.9 (Closed) Inspection Follow-up Item (237/93030-04) -- There was no investigation to determine the cause of the miswiring of the limit switch for the high pressure coolant injection (HPCI) containment isolation valve, which resulted in a failure of the valve to open in May of 1993. Licensee personnel's investigation identified the causes as failure to follow work instructions and inadequate emphasis on implementation of change documents (such as the field change request which implemented modification M12-2-92-001G). As part of the corrective action, licensee personnel provided training to Site Engineering and Construction (SEC) personnel involved in the modification process. This item is closed.

2.10 (Open) Unresolved Item (237/249/93034-06) -- A portion of the Unit 2 core spray leak detection instrument lines had not been tested. This item was reviewed and the majority of the work was still pending. This item will remain open.

3.0 Licensee Action on Licensee Event Reports

The inspectors reviewed the action taken on a number of licensee event reports (LERs) for appropriate corrective actions. The LERs reviewed and the inspectors' evaluations of the actions to address these issues are discussed in this section.

3.1 (Closed) Licensee Event Report 237/87-010 -- This LER, dated February 2, 1987, reported that the embedment plate for support M1150-D-62 on the Unit 2 Core Spray system had pulled away from the ceiling 1/8" to 1/4".

During the review of records related to this LER, the inspectors noted that calculations concluded that the damage was caused by water hammer. The water hammer was not reported in LER 87-010 or by a separate LER. Licensee personnel committed to update LER 87-010 to include documentation of the water hammer, root cause evaluation, and corrective actions.

Licensee personnel performed a walk down of the system and found no other damage. The damage to the support was repaired and an embedment plate assessment program was completed at both the Dresden and Quad Cities stations. The hardware changes made to correct the damage to the pipe supports appeared to be adequate. This item is closed.

3.2 (Closed) Licensee Event Report 237/91-012: This LER, dated October 3, 1990, reported significant corrosion on Unit 2 pipe support M-3212-05 at the base plate and on the exposed portion of the concrete expansion anchors. The cause was attributed to corrosion induced by standing water in the HPCI steam tunnel. The standing water was due to ground water in-leakage. Licensee personnel resolved the problem by injecting

hydrophilic polymer resin into the concrete cracks where the in-leakage was found. This item is closed.

3.3 (Closed) Licensee Event Report 237/92-045-2 -- This LER supplement, dated January 28, 1994, provided follow up information on an isolation condenser group V isolation due to spurious flow spikes. The cause of the flow spikes was determined to be process noise from operation of the shut down cooling system. To reduce the number of spurious group V isolations, licensee personnel revised operating procedures to isolate the system when not required by Technical Specifications. Licensee personnel planned to supplement the LER to correct information concerning the corrective actions. The inspectors reviewed the procedural changes and no concerns were identified. This item is closed.

3.4 (Closed) Licensee Event Report 237/93-001: This LER, dated December 23, 1992, reported a pipe support removal to allow the installation of a new support on the isolation condenser clean demineralized water fill system without analyzing for this condition. The inspectors reviewed the corrective actions taken to resolve the problem and considered them to be acceptable. This item is closed.

3.5 (Closed) Licensee Event Report 237/93-012: This LER, dated May 18, 1992, reported that the Unit 2 emergency diesel generator breaker failed to auto-close as required. The LER indicated that the Bus 24-1 main supply breaker linkage arm which, upon breaker operation, positions the breakers cell mounted auxiliary switches, had been bent. Upon opening of the main bus supply breaker, the auxiliary switch did not change position to indicate to the diesel generator breaker closing logic circuit that the main supply breaker was open. Thus the incoming diesel generator breaker closing logic was blocked. Corrective actions included the replacement of the cell switches and linkage arms with a new design mechanism similar to that used in the 4.16KV upgrade modification. This item is closed.

3.6 (Open) Licensee Event Report 237/94-006: This LER, dated February 5, 1994, reported that Unit 2 shutdown cooling pump motors 2A, 2B and 2C had been replaced with motors which had different electrical characteristics and the protective relay settings had not been changed or evaluated for the new motors. Subsequent evaluation and analysis resulted in resetting the relays to accommodate the replacement motors. During a review of the documentation and analysis which documented the revised relay settings, the inspectors noted that the consultant who developed the new settings, expressed concern regarding the motor acceleration current versus time characteristics. An attachment to a letter dated February 10, 1994, stated that the consultant could not verify that the settings would allow acceptable motor starting. The CECo Systems Protection Department concurred and strongly recommended that the station conduct testing to determine the motor starting characteristic curves at the earliest convenience. When queried, licensee personnel were unable to find evidence that testing had been conducted. This LER will remain open pending performance of the testing and the NRC review of the test results.

4.0 Inspection Objectives

The objectives of the inspection were to determine if engineering activities supporting the Dresden Power Station were properly coordinated and effectively controlled and implemented. The inspectors focused on the identification and resolution of technical issues and problems, design changes and modifications, and internal assessments of engineering. This was accomplished by observation of work activities, interviews with selected personnel (including engineers and engineering management), and reviews of records, procedures, and associated documentation.

4.1 Performance Data and System Selection

The selection of systems and components for emphasis during this inspection was based on a review of data from licensee event reports, latest SALP information, and discussions with cognizant NRC personnel. The systems selected were the Reactor Containment, Standby Liquid Control System and Standby Gas Treatment systems. Modifications and records for specific electrical, mechanical, and instrumentation components of these systems were selected for review. Activities and documentation involving other systems and components were selected and reviewed during the inspection to supplement the selected systems. Consideration was given to the systems and components considered most safety significant.

4.1.1 Reactor Containment

The inspectors reviewed selected records and walked down the accessible portions of the reactor building. The following observations were noted:

4.1.1.1 Containment Isolation Valves Operability Assessments

The inspectors reviewed two operability assessments performed by Site Engineering and Construction (SEC) personnel. These assessments, dated March 8, 1994 and March 10, 1994, concerned the HPCI, reactor water cleanup, and isolation condenser containment isolation valves. The assessments were performed in response to Electric Power and Research Institute testing of motor operated valves (MOVs) which indicated that higher actuator thrust might be required for certain types of MOVs. Based on the assessment results, the Unit 3 HPCI containment isolation valves were declared inoperable on March 3, 1994. As a result, the Unit 3 refueling outage, D3R13, was started early. This issue will be followed through the LER reporting system (LER number 94-006).

The assessment concluded that the Unit 2 HPCI containment isolation valves were operable because of sufficient margin provided by MOV upgrades completed during a May 1993 outage. The other valves evaluated also had sufficient margin to be considered operable. No concerns or problems with the operability assessments were identified.

4.1.1.2 Design of the Containment Hardened Vent System

The inspectors reviewed modifications M12-2-90-029 and M12-3-90-029, which required the addition of a hardened containment wetwell vent to meet Generic Letter 89-16. Installation of the modification was completed February 11, 1993. The following design inadequacies or weaknesses were noted:

a. Inoperability of Containment Vent and Purge System -- The hardened containment vent was specifically required for conditions beyond the original plant design/licensing bases which include loss of offsite power and failure of non-safety-related systems such as the instrument air system. Although the new vent valves added by this modification were capable of being operated upon loss of normal instrument air, the existing containment vent and purge system valves, which are in the vent path, were designed to fail closed upon loss of power or instrument air. Since the existing valves could not be opened, the system would be inoperable for the event for which the system was intended. These valves also could not be operated manually because they had no manual operators and, in case of an accident, they could be inaccessible due to high radiation levels.

b. Failed fuel events were not properly considered -- Failed fuel events were not considered in the design of these modifications. It was evident from statements in Generic Letter 89-16, as well as other related documents, that failed fuel was intended to be addressed.

c. The effects of entrained water were not properly considered -- The system would be required to handle significant masses of water and the effects of this water were not adequately considered. Some of these effects were the weight of the water, the potential for water hammer, and the reduction in venting capacity. Examples of where these effects weren't properly considered were Calculation XCE065.0200.001, "Determination of Required Hardened Wetwell Vent Flow Rate and Vent Pipe Size," Revision 0, which did not consider the flow capacity reduction, and operating procedure DOP 1600-21, "Draining Augmented Primary Containment Vent System," Revision 0, which prohibited system draining for containment pressures greater than 1.86 psig, making this procedure unusable for the targeted accident.

d. The strengthened vent system was incomplete -- The hardened vent line exhausted into the non-hardened ventilation ductwork located outside the turbine building which connected with the plant stack. There was no evidence that the weight of the water, the dynamic exhaust effects, and the potential for backup into the other branches of this ductwork was considered. There was no evidence that the stack would handle the water.

e. Radiant Heating was Not Considered -- The purchase specification for the hardened wetwell vent valves and associated hardware identified 120° F as the maximum temperature for the valve operators and associated equipment. No consideration was given to

radiant heating from the adjacent vent piping containing 310° F steam.

f. Design of Backup Air Accumulators

Accumulator Sizing -- The backup air accumulators were installed to provide air for operation of the valves installed during the modifications. Only starting air pressure and pressure loss due to valve stroking were used to determine the size of the backup air accumulators. Other factors which should have been considered were (1) minimum pressure to operate and hold the valves against the maximum differential pressure and flow conditions, (2) system leakage for the event duration, and (3) potential cooling during the event.

Accumulator Testing -- Modification test procedure, M12-3-90-029, did not require adequate testing of the leak tight integrity of the valves' air operator systems. A minimum pressure to which the system could decay and still be operable for the required number of cycles was not established. The minimum pressure specified did not account for pressure losses for the five operating cycles and due to cooling. A starting pressure corresponding to the minimum instrument air supply pressure and a test period correlated to the accident duration were not established.

Licensee personnel contended that the design was consistent with the BWR Owners Group (BWROG) recommended design criteria for this system and that NRR had agreed with the BWROG design criteria. The inspector's review of the NRC/BWROG correspondence revealed no statement which concurred with the design criteria or limited the design requirements to a long-term loss of decay heat removal event but which would not entail significant fuel failure or other factors. This matter will be forwarded to the Office of Nuclear Reactor Regulation for further review.

4.1.1.3 Containment High Energy Line Break Vulnerability

The reactor building closed cooling water (RBCCW) system provided cooling water to two heat loads inside the drywell, the drywell coolers and the reactor recirculation pump seals. MOVs were provided at both the RBCCW supply and return containment penetrations, however, these valves are not automatic. The RBCCW piping inside containment is not designed against a high energy line break event and no credit can be taken for operators closing these valves until ten minutes into the event. Therefore, in a loss of coolant accident, the primary containment could be bypassed through the open containment isolation valves and the RBCCW head tank vent line.

The MOVs, in the RBCCW supply and return lines, were non-automatic containment isolation valves for the RBCCW System. These valves, 2(3)-MO-3706 and 3769-500, were not included in the Technical Specification 3/4.18 Primary Containment Isolation Valve List. These valves were subject to 10 CFR 50, Appendix J, testing and were included in the containment local leak rate testing program.

In order to reduce the significance of this issue until final resolution, licensee personnel have incorporated a requirement to close these MOVs in case of a coolant accident in the abnormal operating procedures. The inclusion of this valve closing requirement in the emergency operating procedures is under review.

This matter will be forwarded to the Office of Nuclear Reactor Regulation for further review and generic implications.

4.1.2 Standby Liquid Control System

During the review of records for the standby liquid control (SBLC) system, the inspectors noted some design errors in modifications M12-2-84-119 and M12-3-84-119. These two modifications were installed in 1986 and 1987, to address an anticipated transient without scram (ATWS).

The original design pressure for the system was 1,275 psi. With two-pump operation, as required by the modification, the new injection pressure was calculated to be 1,330 psig. There were no indications that the increase in pressure was recognized by the licensee and, as a result, the design pressure of the piping and components was not increased to accommodate two-pump operation.

During discussions on the matter, licensee personnel stated that the piping was physically the same as the 1,500 psi design pressure piping located elsewhere in the system and, therefore, the piping could be expected to withstand the additional pressure without detrimental effects. Based on this information, the direct safety significance of the design verification oversight appeared to be low; however, the discrepancies indicated a lack of thoroughness and verification in the design process at the time of the modifications.

Additional errors noted included the following:

a. Inadequate Post-Modification Testing -- Post modification tests required that the system operation be demonstrated with two pumps discharging at 1,275 psig. Post-modification testing did not require that the system perform at the required design conditions and, therefore, the testing was not adequate.

b. Technical Specification not Updated -- During the inspection, the inspectors noted that Section 3/4.4 of the Technical Specification had not been updated to require surveillance testing at the higher pressure.

c. Inadequate System Testing Procedures -- Test procedures for the SBLC pumps and system had not been revised to require testing at the higher pressure. These procedures, as noted during the inspection, were:

(1) DOS 1100-01, "Standby Liquid Control System Pump Test," Revision 15.

(2) DOS 1100-03, "Standby Liquid Control Injection Test," Revision 16.

(3) DOS 1100-04, "Quarterly Standby Liquid Control System Pump Test for the Inservice Testing (IST) Program," Revision 10.

d. Standby Liquid Control Design Basis Document Errors -- The inspectors reviewed portions of the Standby Liquid Control Design Basis Document, DBD-DR-139, Revision A, which was issued September 22, 1993. The system pressure errors described above were found to be included in this document.

These problems indicated a lack of accuracy, thoroughness and verification in the design process. The ability of the modified system to perform as required had never been demonstrated. The inaccuracies in the DBD provide a flawed design basis for future modifications, operability determinations, procedure changes, etc., all of which could affect the safe operation of the plant. This failure to provide adequate design control is an example of a violation of Criterion III of 10 CFR 50, Appendix B (237/249/94003-01(DRS)).

During discussions near the end of the inspection, licensee personnel indicated that steps had been initiated to correct the design errors and impacted documents. The SBLC system DBD had been withdrawn from use and was to be thoroughly checked. Steps were in progress to ascertain the quality of the other issued DBDs.

An example of one violation was identified in this area.

4.1.3 Standby Gas Treatment System

The inspectors reviewed selected records and walked down the accessible portions of the standby gas treatment (SBGT) system. In this area, the inspectors reviewed an operability assessment for the SBGT system, dated January 26, 1994. The evaluation documented a decision that the SBGT system was operable even though the control room SBGT system temperature indicators provided inaccurate readings. The control room indicators were not used for operating procedures and the SBGT system surveillance procedures were revised to use local temperature indicators instead. The local temperature indicators provided more conservative data than the control room indicators. Although this operability assessment appeared to be acceptable, operating for an extended period in this condition was not considered desirable, since the inability to use these control room temperature indicators added to the normal operator workload.

4.2 Observations of Plant Conditions

Early in the inspection, the inspectors performed walkdowns to determine the material condition of the plant. Indications of equipment problems, housekeeping and other unusual conditions were noted. Both units were operating during this portion of the inspection. Plant conditions were also observed during the review of modification and other engineering support activities throughout the inspection.

None of the problems noted appeared to have an immediate safety significance or a significant effect on the operation of the plant. Most equipment in need of repair had been previously identified by licensee personnel. Overall, the facility appeared to be in better material condition than during the previous engineering and technical support inspection. The deficiencies noted, however, indicated that more attention needed to be focused on less traveled and less accessible portions of the facility.

The following adverse or unusual conditions were noted:

a. HPCI System Pipe Supports -- The inspectors observed a significant difference between Unit 2 and Unit 3 HPCI support and restraint arrangement for the pump discharge and cooling water return to the condensate storage tank line, at the pipe riser elbow. The piping configuration for both units were similar in this area. For Unit 2 the restraint was an anchor made from a dead weight support and for Unit 3 the restraint was a sliding support.

   On March 10, 1994, a PIF was written for the Unit 2 anchor as-found condition. The anchor was not analyzed in the original piping analysis, and was not analyzed during the IEB 79-14 walkdown and evaluation program. The operability analysis performed on March 11, 1994, determined that the design stresses were within the FSAR limits. The inspectors reviewed the records, and noted that the rigid support was upgraded in 1970 by adding bracing and a vertical column below the support. During construction, the interface between the pipe and the support surface was inadvertently welded up; this in effect changed the rigid support into an anchor. A walkdown to determine IE Bulletin 79-14 discrepancies failed to identify this discrepancy.

   Since the dead weight support in Unit 3 had not received an upgrade similar to the one in Unit 2, the adequacy of the support was in question. Licensee personnel presented the Support calculations, performed in 1983 appeared to be adequate. This matter is considered resolved.

b. SBGT System Damper Operator -- Grease was noted on the electrical connection for the limit switch housing on the operator for ventilation damper MO 2/3-7505A. The damper was the inlet isolation damper for the "A" train of the SBGT system. Because of the location, the grease could have been coming from inside the actuator limit switch housing. Excessive grease in the limit switch portion of the actuator housing could adversely affect operation of the actuator. Licensee personnel agreed to remove the limit switch housing cover during the next maintenance outage for the SBGT system "A" train to determine the source of the grease. Although the system engineer was aware of the grease, he did not realize that grease might be inside the limit switch housing and could adversely affect the actuator.

c. RBCCW system -- The inspectors noted that the RBCCW system included several valves that were missing nuts from packing glands, pipe hangers disconnected or not supporting load, and

several small pipes rubbing on other pipes or hard surfaces. Although these problems were not individually significant, collectively, they showed how vibrations resulting from the RBCCW pump and line cavitations were affecting the system.

d. 4KV Switchgear -- During a walkdown of some safety-related 4KV switchgear areas, the inspectors noted that the majority of bolts on the rear panels of safety-related switchgears 23-1 and 24-1 were missing or not secured. These panels cover the switchgear bus bars and need to be securely closed to prevent any possible personnel injury or bus grounding. Additionally, several breakers had been removed from their respective cubicles and neither the breakers or the cubicle openings were properly covered. Although no requirement existed, covering the breakers and the openings with the provided covers was considered to be a good maintenance practice for prevention of personnel injury and dust and dirt intrusion.

One breaker, removed from the cubicle in the Unit 2 Turbine Building 4KV Room, was found to be unrestrained. This was not in accordance with Dresden Operating Procedure (DOP) 6500-04, "Racking Out 4160 Volt Manually Operated Air Circuit Breaker," which required that breakers removed from cubicles be restrained to prevent rolling. The failure to follow an approved station procedure is an example of a violation of Criterion V of 10 CFR 50, Appendix B (237/94003-02A(DRS)).

e. Temporary Wooden Barriers -- During plant walkdowns, the inspectors noted that temporary wooden structures had been erected in front of safety-related motor control centers 28-1 and 39-1 to serve as protective barriers during Unit 3 maintenance activities. Discussions with licensee personnel indicated that the possible impact of these barriers on plant operations had not been evaluated. When Unit 3 shut down for the outage, this ceased to be a problem. Licensee personnel were aware of the concern about evaluations of unusual conditions during plant operations.

An example of one violation was identified in this area.

4.3 Engineering and Technical Support

Engineering and technical support at the Dresden Power Station was provided by two separate organizations. Systems engineering support was provided by the technical services organization and the site engineering and construction (SEC) organization provided the support for design changes and modifications. The inspectors reviewed the engineering support provided by both organizations.

4.3.1 Systems Engineering Support

Systems engineering support was provided by the plant technical staff. Systems engineers provided oversight for the assigned systems; these engineers focused on daily operations and maintenance activities of the assigned systems or system components. The engineers aided plant operations and maintenance personnel in resolving technical issues and

problems and were involved in complex maintenance evolutions in the assigned systems. They also coordinated potential design changes with other engineering organizations.

While conducting facility tours, the inspectors noted several material condition problems with the RBCCW system and 4KV Switchgear. These conditions were very visible and should have been identified by system engineers, backup system engineers, or supervisors while performing system walkdowns in accordance with System Engineering Memo SEM-01, System Engineering System Walkdown Guidance. A review of several system engineering walkdown check sheets indicated that system engineers made routine walkdowns but apparently failed to conduct thorough system walkdowns in accordance with the intent of management guidance.

The experience and qualifications of system engineers was mixed. The inspectors noted several system engineers with less than two years experience. The apparent inexperience of some of the system engineers appeared to be a problem in some areas. Licensee personnel were aware of this condition and, during the past year, had developed a formal training program for systems engineers.

Most engineers appeared to have a reasonable understanding of the assigned system functions and attributes; however, some engineers did not always appear to be knowledgeable of the assigned systems and system related problems. Adequate involvement of system engineers in plant support activities was not always evident. Some system engineers did not appear to be involved and some were less than aggressive in their approach to involvement in activities relating to the assigned systems. This was demonstrated by less than thorough system walkdowns and a lack of involvement of system engineers in the resolution of inspection concerns. Repeated requests had to be made to get system engineering involvement in NRC concerns during the inspection.

Based on the inspection results, the inspectors concluded that the technical support for station activities, provided by systems engineering, was adequate. Even though systems engineers had been relieved of direct responsibility and involvement in design changes, very little improvement was noted in the System Engineering Organization since the E/TS inspection conducted a year ago.

4.3.2 Site Engineering and Construction

The site engineering and construction (SEC) organization had the primary responsibility for coordination, evaluation, development, and installation of design changes and modifications. This organization was divided into four groups which included engineers with mechanical, electrical and other engineering specialties. The primary purpose of SEC was to develop and coordinate plant modifications, including design, safety reviews, installation, and post modification testing in the respective discipline. Each modification was assigned to an engineer actively involved in all phases of the modification. The engineers completed walkdowns, as necessary, to ensure proper design implementation and resolution of installation problems.

Communications and coordination between site engineering and other plant organizations such as systems engineers, plant management, operations, maintenance, construction, and other plant personnel was effective.

Based on the inspection results, the inspectors concluded that, in most cases, the SEC engineers were experienced and qualified. A substantial improvement was evident. Most engineers appeared knowledgeable of the assigned areas and appeared well motivated in their areas of responsibility. The level, quality, and timeliness of engineering and support in this area appeared to be good with a few exceptions. A discussion of the exceptions follows.

4.3.2.1 Review of Modification Packages and Records

The inspectors reviewed selected portions of both open and closed modification packages and supporting records, with emphasis on the selected systems. The records were reviewed to verify the packages were complete and accurate, the modifications were adequately controlled, and regulatory requirements were met. The review included verification that the description of the modification, the 10 CFR 50.59 safety screening or evaluation, installation instructions, documentation of work performed, post-modification testing requirements and test records were adequate. In some cases, other supporting records associated with the modifications, such as calculations and drawings, were selected and reviewed to verify the adequacy and accuracy of the engineering process.

The 26 modification packages reviewed were:

M12-0-91-019F	M12-2-84-119	M12-2-88-60	M12-2-89-004
M12-2-91-020	M12-2-90-028	M12-2-90-029	M12-2-91-021
M12-2-91-022	M12-2-93-004	M12-3-84-119	M12-3-88-60
M12-3-89-004	M12-3-90-013A	M12-3-91-020	M12-3-91-021
M12-3-91-022	M12-3-93-004	P12-2-90-710	P12-2-90-713
P12-2-90-718	P12-2-91-660	P12-3-91-729	P12-3-91-730
P12-3-91-731	P12-3-92-612.		

Problems or concerns noted during the review were:

a. M12-2-90-029 -- This modification installed the hardened wetwell vent required by Generic Letter 89-16. Problems with this modification are discussed in Section 4.1.1.2 of this report.

b. M12-2(3)-84-119 -- These modifications installed changes to the SBLC system to comply with the ATWS rule. Problems with this modification are discussed in Section 4.1.2 of this report.

c. M12-3-90-13A -- This partial modification (including Addendum 1 and 2) was required to add an alternate 125V DC battery system, including a battery charger, a battery and connecting cables, to the Unit 3 safety-related DC control system. This would allow the normal battery system to be removed from service for required testing without having to enter into a duel unit LCO. This partial modification was completed in January of 1993.

The inspectors noted that control room drawing 12E-2322B, "Overall key Diagram, 125V DC Distribution Centers, Dresden Nuclear Power Station Units 2 & 3," Revision C, dated July 3, 1991, had not been marked or revised to show the Unit 3 alternate battery, battery charger or cabling additions, however, the diagram did show the similar additions which had been made for the Unit 2 modification.

Procedure DAP 02-09, "Control of Critical Drawings," required that control room drawings be revised or updated to reflect the correct plant configuration. The failure to implement the requirements of this procedure and update the control room drawing with this design change is considered an example of a violation of Criterion V of 10 CFR 50, Appendix B (249/94003-02B(DRS)).

Most of the modification packages and supporting records appeared to be adequate. With the exception of the noted deficiencies, records indicated that modifications were adequately controlled and were consistent with regulatory requirements. The inspectors concluded that the modification process was effective.

An example of one violation was identified in this area.

4.3.2.2 Review of Exempt Change Program

An Exempt Change (EC) Program had recently been developed to expedite the review and approval of modifications of minor significance and with low potential to significantly reduce the margin of nuclear safety. This process replaced the former minor change process which was no longer used. These minor modifications were called "Exempt Changes" and were process modifications which were exempt from the specific requirements of the modification and minor plant change processes.

The inspectors concluded that the EC process provided a viable and effective method to control minor modifications of low significance. The records reviewed indicated that exempt changes were adequately controlled, were consistent with regulatory requirements and the process was effective.

4.3.2.3 Temporary Alterations

The inspectors reviewed the methods used to control temporary alterations (TAs). The methods were described in procedure DAP 07-04, "Control of Temporary System Alterations," Revision 17. The procedure appeared to be inadequate in some areas. For example, a one time justification was required for TAs to be installed more than 90 days. Periodic review and justification for continued installation was not required.

The Dresden Temporary Alteration Report, dated March 31, 1994, was reviewed. The report listed 48 open TAs, which was more than twice the recently established goal of less than 20 open TAs. The TA procedure defined a TA to be an alteration expected to be installed for less than six months. Thirty of the TAs listed in the report had been open

greater than six months and 10 had been open longer than two years. This appeared to be a misapplication of the TA procedure.

Licensee personnel stated that some recent changes had been made to improve control and reduce the number of TAs. A report of open TAs was issued to management and the individuals assigned open TAs monthly. Although some steps, such as the establishment of goals to reduce open TAs, have been taken, management emphasis and action is needed to ensure that appropriate action is taken and that thorough and adequate control is provided for TAs. Significant systems engineering involvement appeared to be needed to eliminate and prevent the installation of unneeded TAs.

Based on the review of the TA process and selected TA packages, the inspectors considered the methods used to control TAs to be weak. Problems were noted with five of the seven TA packages reviewed and the number of TAs had increased approximately twenty percent in the last year. Several problems involving both safety and non-safety-related hardware were noted in this area. A discussion of these problems follow.

4.3.2.3.1 Review of Temporary Alteration Records

The inspectors reviewed seven TA packages to verify proper control. Problems or concerns were noted with five of the seven TAs. The packages reviewed and the results follow:

a. TA II-33-93 -- This TA allowed the injection of "Furmanite" to repair a leak in the valve packing area of valve 2-220-102 in the Unit 2 reactor recirculation system. Installation of the TA was completed May 27, 1993.

   A hole was drilled into the valve yoke and sealant was injected into the packing area to eliminate or reduce the leak. The engineering evaluation, performed at the time of installation, failed to consider potential over pressurization of the valve due to the sealant injection process. Licensee personnel estimated that the worst case injection pressure applied to the valve could have been as high as 4700 psig which was considerably higher than the valve's rated pressure of 3600 psig. NRR is currently reviewing control of the Furmanite process throughout the industry.

   The inspectors also noted that the valve affected by the TA formed part of the reactor coolant pressure boundary, which was addressed by Section 3.6 of the Technical Specifications. The screening evaluation, included in the package, incorrectly concluded that no safety evaluation was required.

   During this inspection, an operability assessment for the valve was performed and documented. In addition, a safety evaluation as required by 10 CFR 50.59 was performed. The inspectors did not identify any concerns regarding the operability evaluation and the safety evaluation. Prior to this inspection, SEC personnel recognized that oversight weaknesses existed with the sealant

injection process as early as May, 1993. However, formal tracking of plans to develop a controlling document was not initiated until February, 1994. The emphasis for such tracking was largely due to problems experienced with TA II-1-94 (feedwater pump suction valve sealant injection repair); Information Notice (IN) 93-90, "Unisolatable Reactor Coolant System Leak following Repeated Applications of Leak Sealant;" and an industry notification concerning sealant injections.

b. TA II-34-93 -- This TA disabled the Reactor Feed Pump (RFP) motor cooling fan intake and exhaust damper controls to keep the dampers in the fully opened position. Installation of the TA was completed June 11, 1993.

The TA was installed to ensure adequate cooling of the RFP motor during the summer, since there had been problems with the temperature control system. The motor was cooled by continued circulation of outside air through the ventilation ductwork with all dampers fully open. This TA was left installed through the winter with no evaluation of the possible adverse effect of over cooling the motor. With the dampers fully open, extremely cold air from outside could be blown into the RFP motor by the continuously running cooling fans. During the inspection, the systems engineer reviewed the computer point history data for the RFP stator winding and noted that the lowest temperature was above ambient. The warm air that prevented the over cooling was determined to be coming from the full open recirculation damper.

In addition, there was a lack of documentation of the problems that required the TA. Records only indicated that the exhaust damper was broken. Discussions with the responsible systems engineer indicated that none of the dampers in the RFP motor cooling system, including supply air damper, exhaust air damper, and recirculating damper were found in their expected positions. Some internal linkage slippage had also been identified. However, there were no indications that action had been taken to correct these problems to restore the temperature control system to proper operations.

The control of this TA was inadequate. The system was allowed to operate during the winter in an un-evaluated condition and the corrective action to restore the temperature control system to acceptable operations was not timely. This matter was discussed with licensee personnel.

c. TA II-60-93 -- This TA allowed the installation of inlet and outlet pressure gauges on a refrigeration control unit heat exchanger for control room heating, ventilation, and air conditioning equipment. Installation of the TA was completed in December, 1993. No problems or concerns were identified with this TA.

d. TA II-1-94 -- This TA allowed the installation of a "Furmanite" clamp on the 2A RFP suction valve bonnet to body flange to stop or

reduce water leakage to an acceptable level. Installation of the TA was completed January 11, 1994.

After the "Furmanite" clamp installation and sealant injection, water leakage returned to the previous leakage rate in less than one day. The failure of the TA appeared to be improper control of the "Furmanite" process. The curing temperature for the sealant compound not being specified in the work procedure and was not monitored during "Furmanite" injection. Engineering Department Technical Information Document TID-MS-06, "Injection Leak Sealant Application General Use," December 30, 1991, did not require monitoring of curing temperature during injection.

The work package required that a responsible engineer from SEC be present when sealant compound was injected into the "Furmanite" clamp. This item was assigned to the mechanical maintenance department and SEC was not involved in the work. These concerns were discussed with cognizant licensee personnel.

e. TA III-21-92 -- Eight thermocouples, two pressure transducers, two flow transducers, and one axial shaft displacement probe were installed, under this TA, to monitor system parameters and to assist in the determination of causes for RFP seal failures. Installation of the TA was completed June 29, 1992.

In reviewing the package, the inspectors noted that there were no design documents to identify the location of the instruments, the hardware to be used, the hardware accuracy and the functional test requirements.

In 1988, the RFP supplier recommended changing to another type seal. The recommended seals were purchased and installed in all Unit 2 RFPs in late 1992. No Unit 2 RFP seal failures had been reported since. The inspectors concluded that the TA was not adequately engineered and that the change appeared to have little or no actual value since the change in the type of seal had apparently solved the problem.

These concerns were discussed with licensee personnel.

f. TA III-22-92 -- This TA was implemented to bypass a switchboard mounted fuse holder which included a solid copper link rather than an actual fuse. Installation of the TA was completed July 2, 1992. No problems or concerns were identified with this TA.

g. TA III-40-92 -- This TA involved the existing 7-second time delay feature of the degraded grid voltage protection scheme for the Unit 3 safety-related 4.16KV bus 33-1. Installation of the TA was completed October 30, 1992.

On March 10, 1994, the inspectors reviewed the control room copy of the drawing 12-3345, Sheet 2, "Schematic Control Diagram, 4160V Bus 33-1, 4KV Swgr. Bus 40 Feed Bkr., Unit 3," Revision AF, dated March 9, 1993, titled and noted that the markings on the drawing did not agree with the TA. The markings indicated that the relay

time delay was 2-seconds rather than the required 7-seconds. The replaced relay was still indicated as an instantaneous relay, rather than the installed time delay relay, as called for in the TA documentation.

Procedure DAP 02-09, "Control of Critical Drawings," required that control room drawings be revised or updated to reflect the correct plant configuration. The failure to implement the requirements of this procedure and correctly update the control room drawing with the plant configuration for this temporary alteration is considered an example of a violation of Criterion V of 10 CFR 50, Appendix B (249/94003-02C(DRS)).

Examples of two violations were identified in this area.

4.3.2.4 Review of Safety Evaluations and Screenings

The inspectors reviewed the methods used to perform 10 CFR 50.59 safety screenings and evaluations. Records of the screenings and evaluations were reviewed for the selected modification, exempt change and temporary alteration (TA) packages to verify completeness, accuracy, and compliance with regulatory requirements.

Procedure DAP 10-02, "10CFR50.59 Review Screenings and Safety Evaluations," Revision 8, was reviewed and found to be well written and concise. Several minor weaknesses were noted and discussed with licensee personnel for possible procedural improvements. Licensee personnel acknowledged these weaknesses and indicated that the procedure would be revised.

During the review of modifications M12-2(3)-93-004, the inspectors noted that an unreviewed safety question was identified during the original 10 CFR 50.59 safety evaluation performed for the modifications. Due to discussions with the Office of Nuclear Reactor Regulation regarding the issue, the modification was subsequently redesigned to eliminate the unreviewed safety question. As a result of the identification of this problem, Information Notice 93-89, "Potential Problems with BWR Level Instrumentation Backfill Modifications," was issued. The inspectors considered the identification and resolution of this issue to be a positive application of the 10 CFR 50.59 safety evaluation process. Safety evaluations or screenings were usually performed and were acceptable. The modification and TA packages reviewed contained the required 10 CFR 50.59 safety screenings or evaluations and they appeared to be well documented. Some of the packages contained additional supporting information. Procedure DAP 10-02, if properly implemented, provided sufficient controls to ensure that proper 10 CFR 50.59 screenings and safety evaluations were performed and documented as required. Based on the review of records and subsequent discussions with licensee personnel, the inspectors concluded that the safety screenings and evaluations were acceptable.

4.3.2.5 Review of Calculations

In order to complete the assessment of the design change and modification process, the inspectors reviewed portions of selected

calculations that were performed or revised to support the selected modifications. Calculations were reviewed for completeness, accuracy, validity of assumptions, and conservatism with emphasis on how well the calculations supported the respective modification. Some of the calculations were performed by licensee personnel while others were performed by contractors.

Based on the review of calculations, the inspectors concluded that, overall, calculations were acceptable. Several calculation errors were noted. Although most of the individual problems in this area were not considered to have a significant effect on equipment function, improvement was needed in this area. This matter was discussed with licensee personnel.

The following concerns were noted during the review of calculations:

a. Calculation 8982-19-19-2, "Calculation for Contactor/Interposing Relay Coil Voltage at Pickup," Revision 1, dated December 22, 1992 -- Revision 0 of this calculation identified the marginally acceptable conditions for the contactor coils associated with the modifications P12-3-92-611, 612, 613 and 614. However, the results and conclusions of this calculation, including Revision 1, indicated that the minimum pickup voltage acceptance criteria was not met by six circuits. These control circuits were identified as those for the following motor loads:

   - HPCI auxiliary coolant pump,
   - HPCI pump area cooling unit,
   - reactor protection system MG set 3B,
   - reactor building cooler recirc pump,
   - motor operated valve 202-4A and
   - motor operated valve 202-4B.

   Licensee personnel indicated that these six circuits had been analyzed and the conditions had been resolved or justified. Documentation could not be located to support this response. Licensee personnel advised that they would continue to search for the documentation and, if the records could not be found the conditions would be reanalyzed and new documentation prepared. The inspectors considered this to be an inspection follow up item pending NRC review of the documentation and resolutions (249/94003-03(DRS)).

b. Calculation XCE065.0200.001, "Determination of Required Hardened Wetwell Vent Flow Rate and Vent Pipe Size," Revision 0, dated July 19, 1991 -- Details of this calculation were discussed in Section 4.1.1.2 of this report.

c. Piping stress analysis for the LPCI piping system -- During a visit to a CECo design contractor on March 16, 1994, the inspectors noted that only one uniformly hot temperature was used in the original piping analysis for the HPCI system. The inspectors counted five different temperature combinations. Since, in some cases, thermal loads were dominating loads, licensee personnel agreed to re-run the piping stress analysis for

the LPCI piping system in a simplified manner. The review focus was placed on the heat exchanger 2B-1503 outlet nozzle. This heat exchanger had the largest design loading imposed on the over-stressed structural wide flange in Unit 2 southwest corner room. The piping stress analysis showed significantly higher loadings in some directions. This raised a question on the validity of some of the earlier piping stress analysis.

Licensee personnel also agreed to re-run the seismic and dead weight analyses and to perform a technical audit on the previous piping stress analysis. This matter is unresolved pending review of the stress analysis and audit (237/249/94003-04).

4.4 Errors in the Updated Final Safety Analysis Report

During the inspection the inspectors noted several errors in the recently rebaselined Updated Final Safety Analysis Report (UFSAR). These errors were:

a. On page 2.4-2, the statement is made that the Probable Maximum Flood (PMF) level is 528'- 0". Table 2.4-1 states that the maximum flood is 508'-0". Licensee personnel determined that the correct PMF level was 528'- 0", which is higher than the plant ground elevation of 517' - 0". The inspectors reviewed procedure DOA 0010-04, "Floods," and noted that the procedure addressed this type flood. Several questions were raised and licensee personnel issued a PIF to ensure that the issue was adequately addressed.

b. On page 6.2-4, the description of the primary containment airlock was not correct.

c. On page 6.2-17, the pressure in the suppression chamber was described as 29 psia from an initial pressure of 0.5 psia. These terms should have been psig.

d. Table 6.5-2 of the UFSAR used incorrect units to reflect pressure drops in standby gas treatment system. Specifically, "ft H2O" rather than "in H2O" was used.

The multiple errors in the UPSAR indicated that the checking and review of changes was inadequate to assure that the document was correct. Licensee personnel indicated that the noted errors would be corrected.

5.0 Self-assessment of Engineering Activities

Self-assessment of engineering activities at the Dresden Station consisted of audits and a special assessment of some engineering activities. In addition, some assessment of engineering activities was accomplished by cause investigation and correction when problems occurred. Overall, the various assessments covered the spectrum of engineering support activities.

5.1. Site Quality Verification Audits

The inspectors reviewed recent Site Quality Verification (SQV) audit and special assessment records and interviewed personnel to determine the effectiveness of the self-assessment of engineering activities. Audits of the engineering group were normally conducted yearly with additional audits of supplemental engineering activities conducted as needed. Records of three SQV audits of engineering or engineering related activities were reviewed and found to be adequate.

The SQV audit records indicated that the scope of the audits performed was adequate to cover engineering activities. Findings and recommendations noted appeared to be appropriate. The audits were not always effective in finding engineering weaknesses. For example, the problem with the misuse of temporary alterations, noted during the inspection, was not identified during these audits.

Corrective Actions Records (CARs) were issued for audit findings and were incorporated into the NTS program. CARs were also tracked by a redundant SQV tracking system. During one of the audits, SQV identified problems in documenting TAs on control room drawings; however, actions taken to correct the problem were not thorough. As part of the response to CAR 12-93-034, a review of all open TAs was performed to ensure that control room drawings properly reflected the TAs. The inspectors identified a TA which was not properly reflected on the applicable control room drawings. This issue is discussed in Section 4.3.2.3.1 g. of this report.

The inspectors considered the audits to be adequate. Identification of engineering weaknesses and follow up to ensure effective action was taken on findings did not always happen. No other concerns were identified in this area.

5.2 Special Assessments of Engineering

Corporate SQV was primarily responsible for special assessments of engineering. No overall assessments of engineering had been performed by this group in the last two years.

In response to continued problems associated with the maintenance of drawings, however, SEC management used an outside consultant, Failure Prevention, Incorporated, to study the problem. The consultants' report, "Organizational & Programmatic Assessment of the Critical Drawing/Design Change Interface at the Dresden Nuclear Station," dated December 1993, made several recommendations and provided useful insight. As a result of the study, the changeover from manual to electronic storage of critical drawings was accelerated, and organizational changes were made to ensure qualified personnel updated critical drawings. The inspectors considered SEC management's use of an outside consultant to study process and organizational problems a positive imitative.

5.3 Trending and Corrective Action

The inspectors reviewed the methods used by engineering to trend equipment problems, investigate problems for cause, and provide adequate

corrective action to correct both the identified problem and the cause. Significant problems or failures were documented on PIFs, which were used as a mechanism for investigation to determine root causes and initiate actions to prevent recurrence. Trending and corrective action are discussed separately in the following sections.

5.3.1 Trending

The inspectors reviewed the methods used to track problems, detect repetitive equipment failures and trend hardware and other quality related problems. A tracking system had been developed to track, sort and allow oversight of equipment failures and other potential problems. The tracking system used a matrix to collect and sort information into functional groups. Because of past problems with inadequate trending, an oversight of this system was provided by the root cause committee.

The systems engineers were responsible for monitoring failure information to detect repetitive hardware failures and problem trends in the assigned systems or system related components. A significant problem had recently occurred in this area. Repetitive failures of the reactor coolant level instrumentation had not been detected and corrective action had not been taken, even though many failures had been documented. Because of this instrumentation issue, licensee management had provided additional emphasis on trending of significant hardware failures, however, it was too early to evaluate the results of this action.

Another trending method used was the Component Failure Analysis Report, which utilized information from the nuclear plant reliability data system. This system allowed a review of industry failure and trending information on selected plant components. Discussions with licensee personnel indicated that, if failure rates were above the industry average for the specified components, the issues were referred to the appropriate system engineers for investigation and possible action.

The inspectors concluded that, although an acceptable trending program appeared to be in place, additional management attention was needed to ensure adequate implementation. Some actions had been taken in this area, however, the results could not be determined at this time.

5.3.2 Corrective Action

The inspectors reviewed the methods used for root cause investigation and corrective action for hardware and other quality related problems. Significant problems or failures, which required a review for cause of failure, were documented on PIFs to track the problems for cause investigations and resolutions. Systems engineers were normally assigned follow up action for PIFs written on their assigned systems. In order to improve the correction system, the threshold level for writing PIFs had been reduced. In addition, a root cause committee had been developed. This committee met each workday to discuss items requiring root cause investigation and possible corrective action. The inspectors attended several of these daily meetings and concluded that the use of the root cause committee was an effective method for implementing the root cause program.

Based on the review of the corrective action program, the review of selected PIFs, attendance at several root cause committee meetings and discussions with licensee personnel, the inspectors concluded that actions taken to improve root cause investigations and corrective actions had been effective and that the corrective action program was acceptable. In most cases, PIFs were written, properly processed, evaluated for cause and the actions taken were appropriate and timely. Improvements in the corrective action program were evident.

5.3.2.1 Nuclear Tracking System

The Nuclear Tracking System (NTS) was the method used for tracking commitments, discrepancies, and material deficiencies. The system database was managed by the Regulatory Assurance Group and each item was assigned to a cognizant supervisor and individual. The system appeared to be dynamic in nature with growing acceptance and usage.

The inspectors noted that the number of items receiving extensions and the new items written exceeded the number of items closed for both engineering groups for the past four months. This trend had resulted in an increasing backlog of open items assigned to the engineering staff. Significant attention had been placed on the closure of items to ensure that overdue items were promptly addressed and, as a result, the daily NTS reports had very few items that were listed as overdue. At the time of the inspection, the increasing backlog did not appear to be a problem, however, unless the trend is reversed, future difficulties will be encountered. Licensee personnel stated that extensions to scheduled due dates now require approval of the responsible supervisor.

During discussions with licensee personnel, several individuals commented that all open items had the same importance and the priority was driven by the scheduled closure date, rather than safety or operational significance. This was discussed with engineering management who agreed that there was no formal prioritization of items once the items were placed in the system. Licensee personnel stated that it was the responsibility of the supervisors to manage individual workloads and, through that function, set priorities of assigned items. A review of many open items found no evidence of significant items being ignored or delayed due to work on less significant items.

5.3.2.2 Review and Evaluation of NRC and Industry Information

The inspectors evaluated the effectiveness of the methods used for review and evaluation of NRC and industry information. This review included the methods used to assure that vendor, industry, and NRC generic information was controlled, distributed, and evaluated and that corrective actions were taken as appropriate.

The Regulatory Assurance Department had the overall responsibility for coordination of review and evaluation of this information. Upon the receipt of a notice or other information an initial screening for applicability was performed.

Distribution to the responsible organization for impact evaluation and determination of possible required action was coordinated with corporate

engineering. Assigned departments were required to provide a response to Regulatory Assurance, noting any plant impact, with recommendations for action if needed. All of the applicable NRC and industry information was tracked by Regulatory Assurance until the issues were closed. Regulatory Assurance was also responsible for assembly of the response package and preparation of the cover letter if a response was required.

In order to determine the effectiveness of this system, the inspectors selected the methods for handling supplier service information letters (SILs) and NRC information notices (IN) for review. These reviews are discussed below:

a. Service Information Letters -- The inspectors reviewed the methods used for the dissemination and response to vendor SILs. Several SILs were reviewed and were found to be adequately tracked and addressed by the technical staff. The individual responsible for the SILs was knowledge of the current status of the SILs selected and readily retrieved documentation for review.

b. Information Notices -- Problems had been previously identified with the program for controlling information notices. A review of the program indicated that these problems still existed, however, because the corrective actions for these issues were still pending, no further assessment in this area was completed.

The licensee's lessons learned program, which provided interface and communication on problems at other CECo plants, appeared to be working effectively. For example, the inspectors noted that Dresden systems engineering was aware of and was tracking an incident that happened at Quad Cities where a reactor recirculation pump inadvertently went to fast speed while the unit was shutdown causing severe vibration and possible damage. Although the evaluation was ongoing, the systems engineer appeared knowledgeable of the problem, actions taken and the status.

6.0 Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. An unresolved item noted during the inspection is discussed in Section 4.3.2.5 c. of this report.

7.0 Inspection Follow-up Items

Inspection follow up items are matters which have been discussed with licensee personnel, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or the licensee or both. An inspection follow up item, noted during the inspection, is discussed in Section 4.3.2.5 a. of this report.

8.0 Exit Meeting

The inspectors met at the Dresden Nuclear Power Station with licensee representatives (denoted in Section 1) on April 5, 1994, to summarize

the purpose, scope, and findings of the inspection. The inspectors discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection, noting that three documents were identified as proprietary during the inspection. Details of these documents are not discussed in this report. Licensee personnel were asked to identify any proprietary information or material discussed during the exit meeting. Licensee personnel did not identify any information or material as proprietary.