U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	50-237/98023(DRS)
Report No:	50-249/98023(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Dresden Generating Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450-9765
Dates:	August 31, 1998 through September 23, 1998
Team Members:	E. Duncan, Team Leader A. Dunlop, Reactor Inspector N. Jackiw, Reactor Inspector R. Mendez, Reactor Inspector G. O'Dwyer, Reactor Inspector R. Carbiener, Contractor
Approved by:	John M. Jacobson, Chief Lead Engineers Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Dresden Generating Station, Units 2 and 3
NRC Inspection Report 50-237/98023(DRS); 50-249/98023(DRS)

The NRC conducted an announced inspection to review the effectiveness of the engineering and technical support organization in the performance of routine and reactive site activities and the effectiveness of the licensee's controls in identifying, resolving, and preventing problems.

Operations

- The corrective action process was effective and the threshold for identifying and correcting problems was low. The licensee had identified significant issues and implemented timely corrective actions which achieved lasting results. (Section O7.1)

- The root cause analysis program was effectively implemented. Issues were thoroughly investigated, the root causes identified were reasonable, and corrective actions were comprehensive and timely. (Section O7.2)

- The operating experience program was effectively implemented. Operating experience information was evaluated appropriately, and adequate corrective actions were identified and implemented in a timely manner. (Section O7.3)

- The nuclear oversight self-assessment program was effectively implemented and provided valuable performance insights. (Section O7.4)

Engineering

- The modifications, temporary alterations, and engineering requests reviewed were adequately designed, evaluated, installed, and tested. Two minor deficiencies were identified. (Section E1.1)

- Operability determinations were of good quality and provided adequate justification for the conclusions with two exceptions. In one case, an operability determination to address condensate storage tank vortexing concerns failed to account for all condensate storage tank water loss sources. In a second case, the licensee failed to complete a timely 10 CFR 50.59 safety evaluation to address a potential unreviewed safety question identified in an operability determination. (Section E1.2)

- Overall, the material condition and housekeeping of the station were satisfactory, and the ability of engineering personnel to identify material condition problems was acceptable. However, the team identified a number of material condition items such as loose or missing fasteners and screws, and various oil and water leaks. The as-built configuration of the plant was in conformance with the description in the Updated Final Safety Analysis Report. (Section E2.1)

- 10 CFR 50.59 safety evaluations were of good quality and the licensee had an acceptable program for ensuring that trained and qualified personnel prepared and reviewed safety evaluations. Three examples were identified in which safety evaluations were not reported as required. (Section E3.1)

- The vast majority of the temporary alterations reviewed were properly approved, installed, and documented. However, a number of deficiencies regarding the adherence to temporary alteration procedure requirements, such as the performance of quarterly walkdowns and extended installation reviews, were identified. In addition, two examples were identified in which temporary alterations were installed in the plant without proper approval. (Section E3.2)

- Overall, the surveillance tests observed and documentation results reviewed were within the required acceptance criteria. Two examples were identified in which surveillance procedures were not adequate to demonstrate that equipment met technical specification requirements. (Section E3.3)

- The licensee had made adequate progress in the implementation of the strategic reform initiatives reviewed. (Section E6.1)

- Engineering department self-assessment activities were effective. In particular, the threshold for the identification of problems was low; identified problems were elevated to the proper levels of management for resolution; operability issues were addressed; and corrective actions were adequate, timely, and properly prioritized. The engineering department self-assessment audit schedule was comprehensive and included all major engineering functional areas. (Section E7.1)

Report Details

I. Operations

The team selected a sample of issues for detailed analysis to assess the effectiveness of licensee controls to identify, resolve, and prevent issues that degrade the quality of plant operations or safety. The controls reviewed included the corrective action program, the root cause analysis program, the operational experience program, and the self-assessment program.

O7 Quality Assurance in Operations Activities

O7.1 Corrective Action Program Review

a. Inspection Scope (IP 40500)

The team assessed the corrective action program through a review of implementing procedures, problem identification forms (PIFs), corrective action effectiveness reviews, Corrective Action Review Board meeting minutes, and actions taken for previously identified trends. The team also interviewed cognizant personnel concerning the corrective action and problem identification processes.

b. Observations and Findings

Background and General Observations

In January 1997, an NRC assessment of the performance of Dresden Station concluded that although safety performance had significantly improved in the plant operations area, the level of improvement in engineering had not yet resulted in fully effective problem identification and resolution. Subsequent NRC assessments performed in September 1997 and March 1998 concluded that the identification and resolution of technical issues in engineering had improved and that overall, the controls for the identification, resolution, and prevention of problems were effective.

During this inspection, the team assessed the corrective action process including timeliness and priority of actions completed or scheduled, as well as trending of problems and tracking of corrective actions to correct problems.

The team noted that the corrective action program had been revised to provide increased consistency between the Commonwealth Edison sites, and to simplify the processes and procedures associated with the program. Procedures to reflect these changes were implemented in August 1998. The team noted that many previous expectations had become requirements. For example, a supervisory review of a PIF was required within 24 hours of PIF initiation. Also, the responsibility for ensuring proper closure of nuclear tracking system items related to corrective actions was transferred from the nuclear oversight and regulatory assurance organizations to the line

departments. The team concluded that the revisions to the corrective action program were positive.

The team reviewed Nuclear Station Procedures AP-1004 through AP-4004 regarding the implementation of the corrective action process which was revised in July 1998. Corrective action program procedures CAP-1 through CAP-3 pertaining to the corrective action program instructional guides implemented in August 1998 were also reviewed. The team determined that the corrective action process delineated in these procedures was thorough, well defined, and outlined an effective process to identify, resolve, and correct problems.

Problem Identification Process

Interviews with licensee personnel indicated a willingness to identify problems. Performance documentation indicated that between 500 and 600 PIFs were generated monthly. In addition, the licensee had a process in place to identify repeat events and had, in fact, identified three occurrences within the past 6 months.

The team concluded that the threshold for identifying a problem was low and the process for identification of problems and trends had been effectively implemented.

PIF Screenings, Evaluations, and Effectiveness Reviews

The team noted that after a PIF was generated, it was reviewed by a cognizant supervisor and a shift manager. This review addressed operability, any required immediate action, and reportability. Following this initial review, the PIF was classified based on significance, further evaluated during a daily Event Screening Committee meeting, and assigned additional investigation when deemed appropriate.

During this inspection, the team routinely attended the daily Event Screening Committee meeting and concluded that this meeting was a valuable tool to discuss, evaluate, and determine future action to resolve an identified issue. The team also attended a Corrective Action Review Board meeting on September 4, 1998, and a Plant Operation Review Committee meeting on September 14, 1998, and concluded that these meetings contained an adequate depth of discussion to address the specific issues. Examples of activities reviewed by the team and discussed during these meetings included the following:

- Reactor Water Cleanup Modification M12-3-97-006

This modification was initiated to address the scenario of a reactor water cleanup system high energy line break outside of containment where automatic isolation of the reactor water cleanup system would not occur. The team noted that the modification would upgrade the classification of the reactor water cleanup leak detection system to safety-related.

During the Corrective Action Review Board meeting, the review board members demonstrated a good questioning attitude to gain sufficient technical understanding of the problem prior to the approval of the modification. The team concluded that the Corrective Action Review Board meeting to address this issue was effective.

- Effectiveness Review of Corrective Actions to Address Licensee Event Report (LER) 50-249/97011.

The team assessed the effectiveness review associated with LER 50-249/97011 which reported a wiring error in the standby liquid control system heat trace controller. The licensee discovered that the control thermostat for heat trace circuit number two was actually wired to the heater element for control circuit number three.

The team noted that six corrective actions were identified to address this issue. Corrective actions 1 and 2 corrected the wiring discrepancy and verified that a similar problem did not exist on Unit 2. Corrective actions 3 and 4 implemented heat tracing setpoint changes. Corrective actions 5 and 6 related to improved work instructions for work planners and improved post-maintenance testing instructions by work analysts. These actions were scheduled to be completed by September 30, 1998 and October 30, 1998, respectively.

The team concluded that the effectiveness review conducted by the licensee was thorough and comprehensive to verify that the completed corrective actions were effective.

- PIF D1998-4794, Score Card Policy Inadequately Implemented by Engineering.

The subject PIF identified that there was no process in place to ensure that the requirements for performing Score Card observations in engineering were met. The team noted that the following corrective actions had been completed: 1) expectations to perform Score Card observations at least monthly had been communicated to engineering supervisory personnel, and 2) a review of Score Card data had been included on the agenda for the quarterly engineering department performance review board. The team also noted that a third corrective action to develop a database to track and trend Score Card data was scheduled for completion by October 30, 1998. The team concluded that the initial identification of the problem was good, and that the corrective actions completed and planned were comprehensive and timely.

c. Conclusions

The team concluded that the corrective action process was effective and the threshold for identifying and correcting problems was low. In addition, the team concluded that the licensee had identified significant issues and implemented timely corrective actions which achieved lasting results.

O7.2 Root Cause Analysis Review

a. Inspection Scope (IP 40500)

The team assessed the performance of root cause analyses. The following root cause analysis reports were reviewed:

Root Cause Report (RCR)	Report Title
RCR 249-200-98-00200	Unit 3 Shutdown Due to Level Switch Issue
RCR 237-200-98-00800	Failure of Pump to Produce Adequate Flow

These reports were evaluated for initial problem identification and characterization, assessment of operability, immediate corrective actions, corrective actions to prevent recurrence, and evaluation of repetitive problems.

b. Observations and Findings

The team reviewed Nuclear Station Procedures NSP-AP-1004 through NSP-AP-4004 which defined the process for investigating and determining the root cause(s) for identified problems. The team verified that this process adequately addressed the elements of identification, classification, investigation, and plant management review. The final report included corrective actions assigned to specific individuals with due dates commensurate with the safety significance of the issue. In addition, the approved investigation reports were distributed to the operating experience group for review.

The team reviewed the following root cause reports:

- Root Cause Report 249-200-98-00200, Premature Unit 3 Shutdown from Scram Discharge Level Switch Issue.

  On March 24, 1998, the licensee identified that the Unit 3 scram discharge volume level switches were not re-scaled to the revised Technical Specification Upgrade Program (TSUP) setpoint values.

  An initial review of the event determined that because the TSUP setpoint change was not implemented, the current setpoint was outside the technical specification limits. However, when a more detailed review was conducted, it was determined that the TSUP setpoint values were revised to account for additional anticipated setpoint drift due to a change in the surveillance frequency from 18 months to 30 months. Therefore, there was no immediate concern that the switches would drift out of the technical specification limits in the current 12-month period.

  The licensee determined that the root cause for this event was the lack of a questioning attitude and mis-communications. Corrective actions for this event included coaching personnel on the importance of communications, clarifying notification requirements, and formalizing the operability declaration process.

The team concluded that the issue was thoroughly reviewed, the root causes identified were reasonable, and the corrective actions were comprehensive and timely.

- Root Cause Report 237-200-98-00800, Failure of the 3C Containment Cooling Service Water (CCSW) Pump to Produce Adequate Flow Due to Foreign Material.

  On June 28, 1998, the 3C CCSW pump failed to produce adequate flow. During troubleshooting activities, the licensee identified debris in the pump. A detailed root cause investigation determined that the debris had likely entered the pump on May 22, 1998, when the inlet bay screens were removed for a quarterly intake screen and intake bay inspection. The team noted that corrective actions for this event included a revision to Dresden Maintenance Surveillance DMS 4100-05, "Unit 2/3 Fire Pump Bay Intake Screen and Bay Quarterly Inspection and Cleaning," and improved diver inspection methods. The team concluded that the issue was thoroughly reviewed, the root causes identified were reasonable, and the corrective actions were comprehensive and timely.

c. Conclusions

The team concluded that the root cause analysis program was effectively implemented. Issues were thoroughly investigated, the root causes identified were reasonable, and corrective actions were comprehensive and timely.

O7.3 Operating Experience (OPEX) Program Review

a. Inspection Scope (IP 40500)

The team evaluated the OPEX program. The following OPEX reports were reviewed:

OPEX Item	OPEX Title
NTS 2374559760700	General Electric Service Information Letter 607: T-Type Scram Solenoid Pilot Valve Inadvertent Scram
NTS 2374049715000	Potential Common Cause 4 Kilovolt Breaker Failures
NTS 2374049818000	Potential for Loss of High Pressure Safety Injection During Surveillance Testing
NTS 2374559761000	General Electric Service Information Letter 610: Incorrect Core Disc Material in Scram Solenoid Pilot Valves
NTS 2374559558401S1	General Electric Service Information Letter 584: Scram Solenoid Pilot Valve Delayed Response

b. Observations and Findings

The team interviewed cognizant personnel involved in the OPEX process and determined that the licensee had implemented several positive initiatives to improve the performance of OPEX activities. For example, procedures were revised to include a discussion of OPEX issues during pre-job briefings and corrective action reviews. Also, the OPEX website was enhanced to make it more user-friendly for station staff. The team also determined that OPEX training had been provided to about 240 station personnel and that training for remaining station personnel was planned in the near future.

The team reviewed Nuclear Operating Directive NOD-OA-26, "Intra-ComEd OPEX/Lessons Learned Program," Revision 2, dated April 14, 1998, and noted that the revised procedure included more detailed guidance regarding the processing, evaluation, and dissemination of OPEX information. In addition, a corporate assessment of the effectiveness of the generic lessons learned process, which included the implementation of the OPEX program, was scheduled for September 1998. The team also noted that a corporate OPEX coordinator position had been established to provide consistency and uniformity for all the Commonwealth Edison sites.

The team observed that increased emphasis had been placed on OPEX issues. For example, plan-of-the-day meetings included a discussion of recent industry events. In addition, Event Screening Committee meeting discussions included OPEX information.

The team reviewed the OPEX reports identified above and determined that the issues reviewed were properly addressed. Resolution of three representative OPEX issues are discussed below:

- General Electric Service Information Letter 607: T-Type Scram Solenoid Pilot Valve Inadvertent Scram.

  General Electric issued Service Information Letter 607 to alert licensees that a single control rod inadvertently scrammed at a boiling water reactor facility due to an inadequate tolerance in the scram solenoid pilot valve.

  The licensee determined that the pilot valves which were the subject of the Service Information Letter were not installed at Dresden Station. The team concluded that the evaluation of this OPEX item was acceptable.

- Potential Common Cause 4-Kilovolt Breaker Failures.

  Vermont Yankee identified that the closing springs on a 4-kilovolt emergency diesel generator output breaker were discharged, which rendered the breaker incapable of closing on demand. Additional investigation revealed that a cotter pin, intended to keep a hinge pin in position, had failed.

  The licensee determined that all affected safety-related breakers were

overhauled in 1996 by General Electric personnel. In addition, operations personnel performed weekly checks of the diesel generator breaker charging springs as a "good practice" initiative. The team concluded that the evaluation of this OPEX item was acceptable.

- Potential for Loss of High Pressure Safety Injection During Surveillance Testing.

  Beaver Valley reported that the method for testing the automatic transfer of the high pressure safety injection system to the recirculation mode had the potential for causing a loss of suction to all high head and low head safety injection pumps.

  The licensee determined that no potential existed for degradation of the high pressure coolant injection system similar to that specified in the above report. The team noted that this determination was based on a detailed review of 15 procedures which could affect, under single failure criteria, the operation of the torus and condensate storage tank suction valves. The team also noted that information on this event was placed into the Dresden Station Industry Lessons Learned Database for use during pre-job briefings. The team concluded that the evaluation of this OPEX item was acceptable.

c. Conclusions

The team concluded that the operating experience program was effectively implemented. Operating experience information was evaluated appropriately, and adequate corrective actions were identified and implemented in a timely manner.

O7.4 Nuclear Oversight Self-Assessment Activity Review

a. Inspection Scope (IP 40500)

The team evaluated nuclear oversight department self-assessment capability through a review of nuclear oversight self-assessment procedures, reports, and other records.

Specifically, the team evaluated the nuclear oversight department identification and characterization of problems, elevation of the problems to proper levels of management for resolution, disposition of operability and reportability issues, and implementation of timely corrective actions.

An evaluation of engineering department self-assessment capability is documented in Section E7.1.

b. Observations and Findings

The team reviewed nuclear oversight self-assessment procedures, detailed plans, and performance records and discussed these documents with licensee personnel. In particular, the team reviewed Nuclear Station Procedure NSP-AP-3009, "Self-Assessment Program," Revision 0, dated February 17, 1998, which implemented the

program. The team concluded that the self-assessment process and schedule were adequate to ensure that all major functional areas were sufficiently reviewed. The team also noted that recent revisions to the corrective action program assigned additional responsibility for self-assessment activities to the line organizations while the nuclear oversight organization maintained the monitoring responsibilities for the self-assessment process.

The team reviewed a sample of self-assessment reports to assess the ability of the nuclear oversight department to identify and correct problems. In particular, the team evaluated the identification and characterization of the specific problems; elevation of the problems to proper levels of management for resolution; disposition of any operability or reportability issues; and implementation of corrective actions, including evaluation of repetitive conditions.

The team determined that the self-assessments were effectively implemented and provided valuable insights. Findings from two representative self-assessments are discussed below:

- Quality Assurance Surveillance 12-98-10: Pre-Conditioning Review.

  This review was conducted to evaluate the station's response to NRC Information Notice 97-16 regarding pre-conditioning. The team noted that although this assessment did not identify any safety issues, the assessment did identify areas for improvement. For example, not all station operability surveillances were reviewed for potential pre-conditioning. The assessment also identified that procedures that control the scheduling of work and surveillances did not provide adequate guidance for preventing pre-conditioning situations. The team verified that PIF D1998-00551 and PIF D1998-00460 were initiated to identify these issues for entry into the corrective action program. The team concluded that this assessment was effective and the resulting actions were appropriate.

- Quality Assurance Surveillance 12-97-07: Operability Determination Timeliness.

  Based on one case in which the time to complete an operability determination appeared to be excessive, the independent safety engineering group initiated an assessment of operability determination timeliness. The team noted that although no technical issues were identified, the licensee determined that a significant attention-to-detail weakness existed in the quality of the operability determinations reviewed. The initiation of this assessment based upon a single example of a timeliness problem indicated that the licensee was proactively responding to potential concerns.

c. Conclusions

The team concluded that the nuclear oversight self-assessment program was effectively implemented and provided valuable performance insights.

III. Engineering

The NRC performed an inspection of engineering and technical support activities. During the inspection, the team focused primarily on the high pressure coolant injection (HPCI), containment cooling service water (CCSW), diesel generator (DG), standby gas treatment (SBGT), 125 volt direct current (VDC), 250 VDC, and reactor feedpump ventilation systems. The team reviewed selected modifications, temporary alterations, 10 CFR 50.59 safety evaluations, operability determinations, and calculations associated with these systems. Walkdowns and interviews with licensee personnel responsible for these systems were also conducted.

E1 Conduct of Engineering

E1.1 Design Modification, Temporary Alteration, and Engineering Request Review

a. Inspection Scope (IP 37551)

The methods used to control design changes, temporary alterations (TALTs), and engineering requests (ERs) were reviewed to verify adequacy, control, and compliance with regulatory requirements. Selected modifications, TALTs, and ERs were discussed with cognizant system and design engineers and the team walked down accessible portions of selected modifications and TALTs. Documents specifically reviewed included the following, where applicable:

- 10 CFR 50.59 safety evaluation
- Supporting calculations
- Operating and emergency operating procedure changes
- Operator training
- Revisions to as-built drawings
- Revisions to the Updated Final Safety Analysis Report

b. Observations and Findings

Overall, the modifications, TALTs, and ERs reviewed by the team were adequately designed, evaluated, installed, and tested. However, the team identified the following deficiencies:

- Design Change Package 9700245: Emergency Core Cooling System Room Cooler Cooling Water Supply Modification

The team determined that a preventive maintenance task to calibrate the flow instrument installed by the modification was not scheduled although the instrument was used to obtain data evaluated against acceptance criteria. In response to this issue, a pre-define in the electronic work control system was created to calibrate the flow instrument on a 2-year frequency.

- Position of Thermal Overload Did Not Match Dresden Engineering Procedure (DEP) 0040-06

During a walkdown of the reactor building 2A and 2B DC buses, the team identified that the thermal overload (heater element) switch on HPCI turbine steam inlet valve 2-2301-5 was in the "D" position although DEP 0040-06, "Safety-Related Motor Valve Data and Settings," Revision 18, indicated that the heater switch should be in the "A" position. Subsequently, the licensee determined that DEP 0040-06 was not revised to reflect the "D" position when exempt change notice P12-2-94-242 was completed which installed a new heater element for the breaker.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. The failure to update DEP 0040-06 to reflect the revised heater switch position was an example where the requirements of 10 CFR 50, Appendix B, Criterion V were not met and was a violation. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

Overall, the team concluded that the modifications, temporary alterations, and engineering requests reviewed were adequately designed, evaluated, installed, and tested. Two minor deficiencies were identified.

E1.2 Operability Determination Review (IP 37550)

a. Inspection Scope

The team reviewed the following operability determinations:

Item	Operability Determination Title
96-051	Upgrade of HPCI Exhauster System to Safety-Related
97-102	Cracked Motor Mounting Rail on Unit 2 DG Air Start Compressor
97-107	Condensate Storage Tank Level for HPCI Transfer
97-081	Minimum Water Level in CCSW Intake Bay
98-009	HPCI Drain Pot Discharge Piping Design Basis Analysis
98-024	SBGT System Iodine Removal Capability

b. Observations and Findings

Overall, the operability determinations were of good quality and provided adequate technical justification for the conclusions. However, the team identified the following weaknesses:

- Operability Determination 97-107: Condensate Storage Tank (CST) Level for HPCI Transfer

Operability determination 97-107 addressed the ability of the low-low level switches in the CSTs to automatically transfer HPCI suction from the CSTs to the torus before air entrainment due to vortexing occurred. The team questioned whether the potential for a failure of the isolation condensers on the non-accident unit had been taken into account, since this failure would necessitate the use of the non-accident HPCI system and would represent an additional "load" on the CST which would further reduce CST inventory. Subsequently, the licensee determined that this factor had not been taken into account and revised the operability determination to reflect this factor. However, since this omission did not impact the overall conclusions in the operability determination, this error was not significant.

- Operability Determination 98-024: SBGT System Iodine Removal Capability

Operability determination 98-024 addressed a higher expected iodine release than what was stated in section 6.5.3.2.G of the Updated Final Safety Analysis Report. The operability determination stated that the SBGT system was operable, but degraded since suppression pool scrubbing had to be factored into the iodine removal calculation to meet offsite dose requirements. The team determined that although the operability determination was completed on May 7, 1998, a 10 CFR 50.59 safety evaluation to review this potential unreviewed safety question was not completed until September 15, 1998, which was not timely.

Subsequently, the licensee determined that the iodine production calculation was excessively conservative since a 24-month fuel burnup cycle was assumed. At the end of the inspection, the licensee planned to re-perform the calculation using a more realistic 18-month fuel burnup cycle.

c. Conclusions

The team concluded that the operability determinations reviewed were of good quality and provided adequate justification for the conclusions with two exceptions. In one case, an operability determination to address condensate storage tank vortexing concerns failed to account for all condensate storage tank water loss sources. In a second case, the licensee failed to complete a timely 10 CFR 50.59 safety evaluation to address a potential unreviewed safety question identified in an operability determination.

E2 Engineering Support of Facilities and Equipment

E2.1 System Walkdown Observations (IP 37550)

a. Inspection Scope

The team conducted walkdowns of the accessible portions of the HPCI, CCSW, SBGT, DG, 125 VDC, 250 VDC, and reactor feedpump ventilation systems to assess material condition, housekeeping, and the ability of engineering personnel to identify problems. In addition, the team verified that the as-built configuration of the plant was in conformance with the description in the Updated Final Safety Analysis Report.

b. Observations and Findings

Walkdown Observations

The team reviewed Appendix D, "System/Component Walkdown Guidelines and Checklist," of the "Plant Engineering Handbook," Revision 1, which provided guidance regarding identification of problems during system walkdowns. Items contained in the handbook for observation during system engineering walkdowns included the following:

Electrical Components

- Mispositioned breaker switches.
- Check cables for proper end terminations.
- Missing bolts or thumbscrews.

Mechanical Components

- Dirt or debris covering cooling grates or filters.
- Equipment skid/foundation bolting and adequate thread engagement.

General Material Condition Items

- Leaks (water, steam, oil, packing, flanges, rust on components or floor).
- Filters, screens, or louvers (clogged, dirty, or missing).
- Lines or pipes (loose, unbracketed, or vibrating).
- Fasteners or bolts (loose, stripped, or missing).

Housekeeping Items

- Dirty or oily equipment or leaks.
- Rags, trash, or debris in the area.
- Loose or improperly stowed tools or equipment.

In addition, the handbook also directed system engineers to prepare an action request for problems identified during a system walkdown.

Although many problems had been identified by the responsible system engineer for resolution, a number of examples were identified by the team where problems were not documented in an action request as required by the plant engineering handbook. These examples included:

Material Condition Items

- Crib House Screen Damage

The team identified a 3-inch diameter hole in a screen covering an opening in the crib house floor above the safety-related intake bay.

- Thread Engagement Deficiencies

The team identified flange studs on the Unit 2 and Unit 2/3 DG cooling water heat exchangers which were not fully threaded into the retaining nut. Subsequently, the licensee determined that although the observed thread engagement did not meet the plant engineering handbook standards, thread engagement was sufficient to assure operability.

- Cable Terminations

The team identified a broken flexible conduit on the temperature element for cylinder 18 of the Unit 3 DG.

- Oil Leaks

The team identified a number of oil leaks associated with the emergency diesel generators and the station blackout diesel generators. Leaking components included a number of pumps as well as lube oil strainer flanges and fuel oil return line couplings.

- Fastener Deficiencies

The team identified a number of fastener deficiencies. For example, fasteners associated with instrument air lines on the Unit 2 and Unit 3 reactor feedpump ventilation systems were loose or missing. In addition, a fastener on the Unit 2 station blackout DG air start motor discharge piping was missing.

Housekeeping Items

The team determined that, in general, the housekeeping of the systems reviewed was good. However, the team identified that the instantaneous breaker trip setting positions for safety-related motor-operated valve 2-1201-1A were poorly labeled.

Updated Final Safety Analysis Report (UFSAR) Review

Although the vast majority of information in the UFSAR was complete and accurate, the team identified two instances in which the UFSAR was unclear or not up to date.

- UFSAR Section 6.5.3.2.H was unclear regarding whether the SBGT system test orifice was temporarily installed for testing or permanently installed as part of the system design.

- UFSAR Section 6.5.3.2 correctly stated that Unit 2 SBGT system motor-operated valve MO 2-7503 was retained open with remote control power removed. However, the UFSAR was not revised to reflect a similar 1991 modification to Unit 3.

The team discussed these UFSAR discrepancies with licensee personnel and PIF D1998-05040 was generated to document the issues for entry into the corrective action program.

c. Conclusions

The team concluded that overall, the material condition and housekeeping of the station was satisfactory, and that the ability of engineering personnel to identify material condition problems was acceptable. However, the team identified a number of material condition items such as loose or missing fasteners and screws, and various oil and water leaks. Housekeeping was assessed to be good overall with only minor deficiencies identified. The team concluded that the as-built configuration of the plant was in conformance with the description in the Updated Final Safety Analysis Report.

E3 Engineering Procedures and Documentation

E3.1 10 CFR 50.59 Program Review

a. Inspection Scope (IP 37001)

The team reviewed the implementation of the 10 CFR 50.59 program including procedures for screening changes, tests, and experiments and preparing safety evaluations; the processes for maintaining records, revising the Updated Final Safety Analysis Report, and reporting to the NRC; and the training and qualifications of 10 CFR 50.59 screening and safety evaluation preparers. In addition, the team reviewed a sample of 10 CFR 50.59 safety evaluations associated with procedure changes, modifications, temporary alterations, and operability determinations.

b. Observations and Findings

10 CFR 50.59 Procedure Review

The team reviewed Nuclear Station Work Procedure (NSWP) A-04, "10 CFR 50.59 Safety Evaluation Process," and verified that the guidance in this procedure was in conformance with 10 CFR 50.59 and NUREG-1606, "Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59." The team concluded that NSWP-A-04 appropriately reflected 10 CFR 50.59 safety evaluation criteria.

10 CFR 50.59 Reporting Review

The team reviewed the reporting of completed safety evaluations to the NRC as required by 10 CFR 50.59(b)(2). During this review, the team questioned numerical gaps in safety evaluations listed in the most recent 10 CFR 50.59 summary report submitted to the NRC. Subsequently, the licensee determined that although most of the gaps were due to safety evaluations which had not yet been completed, three examples were identified in which the results of completed safety evaluations had not been forwarded to the NRC on a quarterly basis as required by NSWP A-04. The oldest example identified

was safety evaluation 1997-01-06 approved on March 28, 1997. Consequently, this safety evaluation was omitted from the quarterly report three times in 1997 and twice in 1998. The licensee generated PIF D1998-05094 to document this problem for entry into the corrective action program. The team also noted that the requirements of 10 CFR 50.59(b)(2) to submit the summary report within 2 years of safety evaluation completion had not been exceeded in any case.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. The failure to report completed safety evaluations to the NRC on a quarterly basis as required by NSWP A-04 was an example where the requirements of 10 CFR 50, Appendix B, Criterion V were not met and was a violation. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

10 CFR 50.59 Program Training Review

The team reviewed the materials used in the training course for personnel that prepared 10 CFR 50.59 screenings and safety evaluations and verified that the information presented in the course was consistent with corporate procedures and NRC guidance. The team concluded that the licensee had an acceptable program for ensuring that trained and qualified personnel prepared and reviewed safety evaluations.

10 CFR 50.59 Safety Evaluation Review

The team reviewed a sample of 10 CFR 50.59 safety evaluations and determined that the safety evaluations adequately addressed the effects of the proposed changes on plant operations, interactions with other systems and components, any new failure modes, and the effects on accidents and transients; and adequately addressed unreviewed safety question criteria.

c. Conclusions

The team concluded that the 10 CFR 50.59 safety evaluations reviewed were of good quality and that the licensee has an acceptable program for ensuring that trained and qualified personnel prepared and reviewed safety evaluations. Three examples were identified in which safety evaluations were not reported as required.

E3.2 Temporary Alteration (TALT) Program Review

a. Inspection Scope (IP 37550)

The team reviewed the implementation of the TALT program delineated by Dresden Administrative Procedure (DAP) 05-08, "Control of Temporary System Alterations," Revision 8, dated August 14, 1998. In addition, the team reviewed selected TALTs to verify adequacy, control, and compliance with regulatory requirements.

b. Observations and Findings

Overall, the team concluded that the vast majority of the TALTs reviewed were properly approved, installed, and documented. However, the following deficiencies were identified:

Unauthorized TALTs

The team conducted plant walkdowns to verify that all installed temporary alterations had been approved. The team identified two instances in which temporary alterations had been made to the plant without utilizing the temporary alteration process. In one case, the team identified a water heater installed to support a temporary asbestos shower that was plugged into a welding receptacle when in use. In a second case, the team identified that the Unit 2 station blackout diesel generator battery room door was propped open with a portable fan installed to supply temporary ventilation.

In response to the unauthorized TALTs identified by the team, the licensee generated PIF D1998-05203 to identify this issue for entry into the corrective action program, and initiated a prompt investigation to determine the extent of the condition. The following specific corrective actions were accomplished:

- Additional plant walkdowns were conducted. As a result, two additional unapproved plant changes were identified; one which involved the use of a portable sump pump, and a second in which a power pack supplying sonic cleaners was plugged into a welding receptacle on the refueling floor.

- All unauthorized temporary alterations were entered into the temporary alteration review process and reviewed as required by DAP 05-08.

- Interviews were conducted with licensee staff and the TALT procedure was reviewed. Following that effort, the licensee concluded that TALT program requirements were adequately understood and the administrative process that controlled TALTs was acceptable.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. The failure to approve a TALT to supply temporary ventilation to the Unit 2 station blackout diesel generator battery room as required by DAP 05-08 is an example where the requirements of 10 CFR 50, Appendix B, Criterion V were not met and was an example of a violation (50-373/98023-01a; 50-249/98023-01a).

Extended Installation Justification Review

Step F.1.e.(14) of DAP 05-08 required that if the removal date of a TALT was expected to go beyond 90 days, then an extended installation justification form was required to be completed. The team identified a number of examples where extended installation

justification forms were not completed for TALTs with expected removal dates which exceeded 90 days. For example, TALT 3-08-98 to plug a leaking traversing incore probe indexer tube was installed on May 10, 1998 with an expected removal date of Dresden Refueling Outage 15 which exceeded 90 days. However, an extended installation justification form was not completed. In addition, TALT 1-02-98 to place a pipe clamp on a leaking fire protection header was installed on May 30, 1998 with an expected removal date of November 20, 1998. However, an extended installation justification form was not completed as required.

The licensee generated PIF D1998-05187 to document this problem for entry into the corrective action program. In addition, the licensee reviewed all existing TALTs and performed and documented extended installation justification reviews, as required.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. The failure to perform extended installation reviews and complete appropriate documentation as required by Step F.1.e.(14) of DAP 05-08 is an example where the requirements of 10 CFR 50, Appendix B, Criterion V were not met and was an example of a violation (50-373/98023-01b; 50-249/98023-01b).

Quarterly Walkdown Review

Step F.6.c of DAP 05-08 required that the TALT program owner shall coordinate a quarterly walkdown of all installed TALTs with the applicable cognizant engineers to verify the current configuration against the original design. In addition, the following was required:

- Results of the walkdown shall be documented on form DAP 05-08F, "Temporary Alteration Walkdown."
- Walkdown results shall be reviewed by the TALT program owner or designee.
- Completed forms shall be retained in the original TALT package.

The team identified that for the TALTs installed, quarterly walkdowns prescribed by DAP 05-08 had not been performed as required.

The following specific corrective actions were accomplished:

- Walkdowns were performed to verify that all accessible TALTs were configured consistent with the original design and were documented on DAP 05-08F.
- A pre-define was generated in the electronic work control system to perform a walkdown of installed TALTs on a quarterly basis as required by DAP 05-08.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. The failure to perform quarterly TALT walkdowns as required by Step F.6.c of DAP 05-08 was an example where the requirements of 10 CFR 50, Appendix B, Criterion V were not met and was an example of a violation (50-237/98023-01c; 50-249/98023-01c).

c. Conclusions

The team concluded that the vast majority of the temporary alterations reviewed were properly approved, installed, and documented. However, a number of deficiencies regarding the adherence to temporary alteration procedure requirements such as the performance of quarterly walkdowns and extended installation reviews, were identified. In addition, two examples were identified in which temporary alterations were installed in the plant without proper approval.

E3.3 Surveillance Review

a. Inspection Scope (IP 37550)

The team observed and reviewed documentation associated with HPCI, CCSW, DG, SBGT, 125 VDC, 250 VDC, and reactor feedpump ventilation surveillances.

b. Observations and Findings

b.1 Surveillance Observation Review

Overall, the surveillances observed were performed satisfactorily and in accordance with the applicable procedures. In addition, system engineering involvement during surveillance testing was good overall and met management expectations that system engineers observe surveillance testing if onsite and available. However, the following weaknesses were identified:

- System Engineer Performance During HPCI Surveillance

On September 10, the team observed the performance of Dresden Operating Surveillance (DOS) 2300-03, "HPCI System Operability Surveillance." The team noted that about a pint of water had leaked from the turbine stop valve onto the floor. The system engineer stated that this was a normal occurrence. After the team questioned whether the water was contaminated, the system engineer contacted radiation protection personnel, who determined that the water was contaminated. The team concluded that the system engineer was not sensitive to contamination concerns.

- System Engineer Presence During Diesel Generator Surveillance

The team identified that contrary to management expectations, the backup DG system engineer did not observe the Unit 2 DG monthly surveillance test performed on September 2, 1998.

b.2 Surveillance Testing Documentation Review

Overall, the completed surveillance results reviewed were within the required acceptance criteria. However, the following deficiencies were identified:

DOS 1500-02, "Containment Cooling Service Water Pump Test and IST"

The team reviewed DOS 1500-02, "Containment Cooling Service Water Pump Test and Inservice Test (IST)," and observed a portion of the surveillance when conducted as a post-maintenance test for the 3C CCSW pump. The following deficiencies were identified:

- Pump Flow Deficiencies

  The team noted that DOS 1500-02 required that Unit 2 CCSW pump flow be established between 3621 and 3721 gallons per minute (gpm) although the acceptance criteria was based on a reference flow of 3600 gpm as stated in the Dresden IST Surveillance Acceptance Criteria Manual (DISACM). A review of the CCSW pump data sheets indicated that the actual reference values for the four Unit 2 pumps were 3620, 3615, 3612, and 3604 gpm, which was less than the minimum flow required by the procedure. ASME Operations and Maintenance Standard, Part 6 (OM-6) required that a reference flow or differential pressure be established and each subsequent test be performed at the same fixed reference value. In addition, NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," permitted a 2 percent tolerance band around the fixed reference value. The flow band stated in the procedure, however, was potentially outside the allowable tolerance band identified in NUREG-1482. The team identified that on May 14, 1998, the flow established during a surveillance of the "D" CCSW pump was 3705 gpm, which was outside the 2 percent tolerance band.

  Subsequently, the licensee determined that the DISACM flow reference value should have been 3671 gpm. To address the concern identified above, the licensee revised the DISACM to document the correct pump reference value and planned to re-baseline the Unit 2 CCSW pumps during the next Unit 2 CCSW pump surveillance.

  Technical Specification 4.0.E.1 required that inservice testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the Commission. The failure to establish appropriate flow during surveillance testing of the 2D CCSW pump on May 14,

1998, was an example where the requirements of TS 4.0.E.1 were not met and was an example of a violation (50-237/98023-02a; 50-249/98023-02a).

- System Flow Acceptance Criteria Deficiencies

The acceptance criteria established in DOS 1500-2 to verify the closure of CCSW check valve 2-3999-252 was inadequate. The procedure was intended to verify that the check valve was closed based on adequate flow from the CCSW pumps to the emergency core cooling system room coolers. The required flow to the room coolers was 87 gpm, however, the acceptance criteria in the test was 50 gpm. The 50 gpm was based on the test procedure normally having only one CCSW pump operating at a time, while 87 gpm was required when two pumps were operating. Basing the acceptance criteria on analytical data was not appropriate. ASME Section XI, IWV-3522, "Exercising Procedure," required the exercising of check valves to their safety position. NUREG-1482 and Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," identified that one acceptable method to verify that a check valve was closed was to verify that the required design flow through another portion of the system could be met. This would ensure that the check valve being tested was sufficiently closed to perform its function.

To address the problem identified above, the licensee planned to revise the test acceptance criteria in DOS 1500-02 prior to the next scheduled surveillance. The operability of the check valve was not a concern since other tests demonstrated that the design flow requirement of 87 gpm was satisfied.

Technical Specification 4.0.E.1 required that inservice testing of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the Commission. The failure to establish appropriate acceptance criteria in DOS 1500-02 was an example where the requirements of TS 4.0.E.1 were not met and was an example of a violation (50-237/98023-02b; 50-249/98023-02b).

- Foreign Material Exclusion Concerns

During the preparation for a 3C CCSW pump surveillance, the licensee identified that the pump suction pressure gauge indicated negative system pressure, although other CCSW pump gauges indicated positive system pressure. The gauge was removed, re-calibrated, and reinstalled. However, suction pressure still indicated lower than the other gauges. The instrument line was then removed, cleaned, and inspected. Blockage due to mud, clay, fibers, and corrosion products was identified. The inside pipe diameter was narrowed down to about 40 percent of its original size due to corrosion. PIF D1998-04993 was initiated to identify this problem for entry into the corrective action program.

Recommended corrective actions for the problem included the inspection of similar small-bore piping.

Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," discussed flushing of instrument lines to prevent blockage in raw water systems such as CCSW. Although the licensee was flushing other instrument lines such as the discharge pressure gauge lines, the suction pressure gauge lines were not flushed because the licensee believed that negative pressure in the line during system operation would preclude the buildup of foreign material. However, since the instrument line was connected to the bottom of the suction piping, the team concluded that debris could accumulate in the line as it settled when the pump was not in use.

DOS 6600-11, "Diesel Generator Cooling Water Pump Test"

Dresden Operating Surveillance DOS 6600-11, "Diesel Generator Cooling Water Pump Test," performed a closure test of check valves 2-1599-131A(B). The test was performed by operating the diesel generator cooling water pump to determine if forward flow could be obtained through the check valves and measured at the downstream flow instrument. Because the discharge pressure of the diesel generator cooling water pump was less than the pressure required to open the piston check valves (spring to close), any flow through the valves would be the leakage past the valve seat. However, the team identified that the initial test conditions allowed service water system operation, which would pressurize the downstream side of the check valves. As such, the team was concerned that leakage past the valve seat may not occur due to the lack of differential pressure across the check valve which would invalidate the test results.

The licensee determined through calculations that sufficient differential pressure existed across the check valves with the service water system in operation. Nonetheless, the licensee issued PIF D1998-05227 to identify this issue for entry in the corrective action program and planned to revise the test procedure to ensure there was a differential pressure across the check valves during testing.

DTS 7500-11, "DOP Testing of 2/3 SBGT HEPA [High Efficiency Particulate Air] Filters"

Dresden Technical Surveillance (DTS) 7500-11, "DOP Testing of 2/3 SBGT HEPA Filters," Revision 6, dated December 10, 1996, contained a number of errors. These errors included a misplaced acceptance criteria identifier, references to deleted technical specification bases, a missing initial/date line, missing references, and the identification of an incorrect sample probe insertion point.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. The failure to correctly identify acceptance criteria data in DTS 7500-11 was an example where this requirement was not met and was a violation. However, this

failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

Overall, the surveillance tests observed and documentation results reviewed were within the required acceptance criteria. Two examples were identified in which surveillance procedures were not adequate to demonstrate that equipment met technical specification requirements.

E6 Engineering Organization and Administration

E6.1 Strategic Reform Initiative Review

a. Inspection Scope (IP 40500)

The team reviewed the implementation of the following strategic reform initiatives:

- NGG-3, Action Step 8: Implement the System Health Indicator Program.
- NGG-8, Action Step 3: Define and issue common procedures for the IST, Inservice Inspection, Maintenance Rule, Appendix R, and Generic Letter 89-13 programs.
- NGG-8, Action Step 4: Implement the engineering work management system.
- NGG-13, Action Step 1: Assess effectiveness of the Safety Review Board, Plant Onsite Review Committee, and departmental self-assessment processes. Also assess the effectiveness of other programs such as the Corrective Action Process and OPEX.
- NGG-13, Action Step 4: Implement a generic lessons-learned process.

b. Observations and Findings

The team reviewed the strategic reform initiatives identified above with cognizant licensee personnel and verified that the action steps were being implemented. No deficiencies were identified.

c. Conclusions

The team concluded that the licensee had made adequate progress in the implementation of selected strategic reform initiatives.

E7 Quality Assurance in Engineering Activities

E7.1 Engineering Department Self-Assessments

a. Inspection Scope (IP 40500)

The team evaluated engineering department self-assessment capability through a review of the following engineering department self-assessments.

Document Number	Document Title
NTS 237-251-98-02500	Engineering Effectiveness
NTS 237-251-98-02100	Engineering Rapid Response Team Effectiveness
NTS 237-251-98-01900	Timeliness of Corrective Action Completion
NTS 237-251-98-00100	Compliance with the Procedures and Policies Associated with the Standardized Corrective Action Program

Specifically, the team evaluated the engineering department identification and characterization of problems, elevation of the problems to proper levels of management for resolution, disposition of operability and reportability issues, and implementation of corrective actions. In addition, the team reviewed the self-assessment audit schedule.

An evaluation of nuclear oversight department self-assessment capability is documented in Section O7.1.

b. Observations and Findings

The team reviewed the engineering department self-assessment audit program, including the audit log and schedule for 1997 and 1998, and verified that all major engineering functional areas were included in the schedule.

Records for completed audits identified above indicated that the audits were performed adequately. In particular, the team verified that the threshold for the identification of problems was low; identified problems were elevated to the proper levels of management for resolution; operability issues were addressed; and corrective actions were adequate, timely, and properly prioritized.

The team concluded that the engineering department self-assessment program was effectively implemented.

c. Conclusions

The team concluded that engineering department self-assessment activities were effective. In particular, the threshold for the identification of problems was low; identified problems were elevated to the proper levels of management for resolution; operability issues were addressed; and corrective actions were adequate, timely, and properly prioritized. In addition, the team concluded that the engineering department self-assessment audit schedule was comprehensive and included all major engineering functional areas.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Violation 50-237/93019-01; 50-249/93019-01: Updated Final Safety Analysis Report Change Lacked Prior NRC Approval.

The team verified the corrective actions described in the licensee's response letters, dated September 3, 1993, and November 21, 1995, to be reasonable and complete. No similar problems were identified.

E8.2 (Closed) Violation 50-237/93019-02; 50-249/93019-02: Inadequate Safety Evaluation for Updated Final Safety Analysis Report Change.

The team verified the corrective actions described in the licensee's response letters, dated September 3, 1993, and November 21, 1995, to be reasonable and complete. No similar problems were identified.

E8.3 (Closed) Violation 50-237/94014-03; 50-249/94014-03: Untimely Corrective Actions for Degraded High Pressure Coolant Injection Cooler Fans.

The team verified the corrective actions described in the licensee's response letter, dated September 22, 1994, to be reasonable and complete. No similar problems were identified.

E8.4 (Closed) Violation 50-237/96002-06; 50-249/96002-06: Inadequate Corrective Action for Breaker Maintenance and Foreign Material Exclusion Problems.

The team verified the corrective actions described in the licensee's response letters, dated June 19 and July 17, 1996, to be reasonable and complete.

During this inspection, one similar foreign material exclusion problem was identified. The team discovered a 3-inch diameter hole in a screen covering an opening in the crib house floor above the safety-related intake bay. To address this issue, the licensee promptly generated an action request and placed additional screening material over this opening. The problem was considered to be minor in nature.

E8.5 (Closed) Violation 50-237/96009-05; 50-249/96009-05: Ineffective Corrective Actions for Battery Calculation.

The team verified the corrective actions described in the licensee's response letter, dated November 13, 1996, to be reasonable and complete. No similar problems were identified.

E8.6 (Closed) Violation 50-237/96014-02; 50-249/96014-02: Failure to Enter Diesel Generator Information Into the Vendor Technical Manual Improvement Program.

The team verified the corrective actions described in the licensee's response letter, dated March 6, 1997, to be reasonable and complete. No similar problems were identified.

E8.7 (Closed) Violation 50-237/97007-05; 50-249/97007-05; Licensee Event Report 50-249/97005-00/01: Failure to Test 250 VDC Battery in As-Found Condition.

The team verified the corrective actions described in the licensee's response letter, dated July 1, 1997, to be reasonable and complete. No similar problems were identified.

E8.8 (Closed) Violation 50-237/97019-01; 50-249/97019-01: Inadequate Diesel Generator Operating Procedures.

The team verified the corrective actions described in the licensee's response letter, dated July 9, 1998, to be reasonable and complete. No similar problems were identified.

E8.9 (Closed) Violation 50-237/97028-01; 50-249/97028-01: Failure to Follow Problem Identification Process.

The team verified the corrective actions described in the licensee's response letter, dated March 11, 1998, to be reasonable and complete. No similar problems were identified.

E8.10 (Open) Unresolved Item (URI) 50-237/96012-03; 50-249/96012-03: Cable Ampacity Concerns.

As discussed in inspection report 50-237/96012; 50-249/96012, the licensee was unable to provide the NRC with documentation to demonstrate that safety-related cables were properly sized. In response to this issue, the licensee planned to evaluate cable ampacity utilizing the Sargent and Lundy Interactive Cable Engineering program to identify potentially overloaded cables.

During this inspection, the team determined that the licensee selected one routing point in the plant that contained the highest number of power and control cables routed together for a worst-case review. Preliminary calculations indicated that the conductor temperatures with all the cables energized was below the cable temperature design limit. Field testing was scheduled to begin in October 1998 and was expected to be completed by March 1999. This item will remain open pending NRC review of the field testing results.

E8.11 (Closed) URI 50-237/96014-03; 50-249/96014-03: Adequacy of Evaluation for Battery Ventilation System Temporary Alteration.

The inspector identified that a technical evaluation of the temporary installation of portable heaters to maintain the safety-related 125 VDC and 250 VDC batteries above the minimum required temperature was questionable. Specifically, the technical

evaluation checklist indicated that the affected equipment was not safety-related, although the batteries were safety-related.

During this inspection, the team determined that the 10 CFR 50.59 safety evaluation for the subject temporary alterations identified that the auxiliary power system, which included the safety-related batteries, was affected. Therefore, the team concluded that the checklist was completed incorrectly. However, the team also concluded that this error was inconsequential since the as-built heating and ventilation system was nonsafety-related.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The failure to complete the temporary alteration checklist correctly was an example where the requirements of 10 CFR 50, Appendix B, Criterion V, were not met and was a violation. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

E8.12 (Open) URI 50-237/98007-02; 50-249/98007-02: Containment Cooling Service Water Pump Operability - Post Dam Failure.

The inspector identified that there were no high point vent valves to vent trapped air in containment cooling service water piping following a Loss-of-Coolant-Accident coincident with a Loss-of-Offsite-Power and a dam failure.

During this inspection, the team reviewed a proposed change to the Updated Final Safety Analysis Report to state that a postulated dam failure concurrent with a Loss-of-Offsite-Power and a Loss-of-Coolant-Accident was beyond the plant design basis. This item will remain open pending a review of the plant design basis by the Office of Nuclear Reactor Regulation.

E8.13 (Closed) IFI 50-237/96201-18; 50-249/96201-18: Failure to Test Unit 2 125 VDC Battery as Required.

As discussed in inspection report 50-237/96201; 50-249/96201, the licensee failed to test the Unit 2 125 VDC battery as specified in Dresden Engineering Surveillance (DES) 8300-28, "Unit 2, 125 Volt Station Main Battery Service Test."

During this inspection, the team verified that a revised battery load profile was developed. The team reviewed the results of the latest service test utilizing DES 8300-28 and verified that the battery was tested to this updated load profile.

The NRC issued a letter dated September 23, 1998, which granted enforcement discretion for this issue in accordance with Section VII.B.6, "Violations Involving Special Circumstances," of the "General Statement of Policy and Procedures for NRC

Enforcement Actions (Enforcement Policy)," NUREG-1600. As a result, the NRC will not issue a violation or propose a civil penalty in this case.

E8.14 (Closed) IFI 50-237/96201-24; 50-249/96201-24: Adequacy of Replacement Breaker.

During a modification review to replace the Unit 2 high pressure coolant injection (HPCI) condenser hotwell drain pump motor, the team questioned the replacement of a Westinghouse Type FA breaker with a Westinghouse Type HFD breaker.

During this inspection, the team reviewed the work package, calculations, post modification testing and the 10 CFR 50.59 safety evaluation associated with the modification. In particular, the team noted that calculation DRE96-0040 verified adequate protection for the pump motor. No deficiencies were identified.

E8.15 (Closed) Licensee Event Report (LER) 50-237/87012-02: HPCI Turbine Trips Due to Hydraulic Control System Problems.

As discussed in the original LER, the licensee identified that a number of HPCI turbine trips were due to premature tripping of the auxiliary oil pump. Supplement 2 to this LER was recently issued to revise corrective actions associated with the original event to address concerns regarding inaccurate 250 VDC battery load profiles.

During this inspection, the team reviewed the revised corrective actions and had no additional concerns. This LER is closed.

E8.16 (Closed) LER 50-237/94021-02: HPCI Turbine Vendor Test Specification Change.

As discussed in the original LER, the Unit 2 HPCI turbine tripped during performance of the monthly operability surveillance due to high exhaust pressure. Supplement 2 to this LER was recently issued to correct technical information regarding the root cause of the event and to revise planned corrective actions to include additional changes to the HPCI monthly surveillance procedure. In addition, the HPCI turbine exhaust check valves were inspected to verify that the repairs to the check valves and the revised HPCI procedure were effective.

During this inspection, the team reviewed the revised corrective actions and had no additional concerns. This LER is closed.

E8.17 (Closed) LER 50-237/96011-00/01: Inservice Tests Not Performed As Required.

During a self-assessment of the inservice testing (IST) program, the licensee identified four IST Code non-compliances: two related to check valves, one related to pressure isolation valve leakage testing, and one related to flow instrument ranges. In addition, during a followup assessment, three additional IST Code non-compliance issues were identified: two related to check valve inspections, and one related to seat leakage testing. The licensee determined that the root cause of the IST non-compliance issues was that due to personnel error, the IST program did not fully incorporate the Code requirements.

As part of the licensee's immediate corrective actions, testing and inspections for the affected components was accomplished. In addition, the licensee planned to revise applicable procedures, including the IST basis document; conduct additional training of engineering personnel; and perform additional self-assessments of the IST program.

During this inspection, the team reviewed the licensee's corrective actions, including the results of a recent self-assessment. The team verified the actions taken for the specific issues identified in the LER and the self-assessment were adequate.

10 CFR 50, Appendix B, Criterion XI, "Test Control," requires that a test program be established to assure that all testing required to demonstrate that structures, systems, and components, will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. In addition, Dresden 2/3 Technical Specification 4.0.E.1 states that inservice testing of American Society of Mechanical Engineers (ASME) Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the Commission. The issues discussed above are examples where these requirements were not met and was a violation. However, this licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-237/98023-03; 50-249/98023-03).

E8.18 (Closed) LER 50-237/96015-00: HPCI Inoperable Due to High Pump Discharge Line Temperature.

The Unit 2 HPCI pump discharge line temperature was observed to be 112 degrees Fahrenheit (°F) and increasing at about 10°F per hour. The licensee determined that the root cause of the event was reactor feedwater backleakage through HPCI discharge testable check valve 2-2301-7. Corrective actions included valve inspection and repair, and procedure enhancements to require cycling of the HPCI pump discharge valves whenever the HPCI discharge line temperature increased toward 150°F.

During this inspection, the team reviewed the work requests associated with the valve repair and verified that applicable procedures were revised to incorporate HPCI valve cycling. No deficiencies were identified.

E8.19 (Closed) LER 50-237/96019-00/01: 250 VDC Battery Not Tested Although Required Due to Battery Loading Changes.

The licensee identified that the Unit 2 and Unit 3 250 VDC system design load profile was incorrectly modeled for the actuation of several HPCI motor-operated valves (MOVs). Specifically, although the valves were modeled as actuating at separate times, testing revealed that the MOV starting currents overlapped. The licensee subsequently determined that the root cause of this event was an inaccurate electrical load profile due to differences in the actual versus calculated inrush current time delays. The licensee

also determined that the capacity of the 250 VDC station batteries was adequate for the maximum load sequence and therefore the safety significance of the event was minimal.

As part of the licensee's immediate corrective actions, loads were removed from the batteries to bring the analytical load profile to a value within the values tested during the last service test. In addition, a modification was installed to avoid overlap of the MOV inrush current.

During this inspection, the team verified that the modification that added the time delay to the MOV initiation logic was adequately tested. In addition, the team verified that the most recent service test incorporated the revised electrical load profile and was adequately performed.

The NRC issued a letter dated September 23, 1998, which granted enforcement discretion for this issue in accordance with Section VII.B.6, "Violations Involving Special Circumstances," of the "General Statement of Policy and Procedures for NRC Enforcement Actions (Enforcement Policy)," NUREG-1600. As a result, the NRC will not issue a violation or propose a civil penalty in this case.

E8.20 (Closed) LER 50-237/96020-00/01: CCSW Temperature Outside Design Basis.

On November 12, 1996, the licensee identified that the CCSW inlet temperature must be maintained below 84°F to prevent exceeding the design peak suppression pool temperature of 170°F and stay within the bounds of the existing containment analysis. The licensee determined that the root cause of this event was inadequate design documentation which led to confusion regarding the original design basis. In addition, inadequate management oversight and design control led to low expectations which resulted in poor identification and resolution of the design basis issue, and inadequate implementation of compensatory actions to address the operability issues.

As part of the licensee's immediate corrective actions, station procedures were revised to reflect the 84°F temperature limit and operators were trained on this issue. In addition, a license amendment was submitted, and Dresden Operating Surveillance (DOS) 1500-12, "CCSW Loop Flow Verification," was created to verify minimum required CCSW flow.

During this inspection, the team reviewed license amendments 157 (Unit 2) and 152 (Unit 3), dated April 30, 1997, which restored the ultimate heat sink temperature limit to 95°F based on a revised system analysis. No deficiencies were identified. In addition, since the revised analysis demonstrated that all emergency core cooling systems would have fulfilled their safety function in the event of a design basis accident, the safety significance of this event was minimal.

10 CFR 50, Appendix B, Criterion III, "Design Control," requires that measures shall be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The issue discussed above is an example where this requirement was not met and was a violation.

However, this licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-237/98023-04; 50-249/98023-04).

E8.21 (Closed) LER 50-237/97005-00: HPCI Low Flow Setpoint Found Outside Technical Specification Limit.

During routine surveillance testing, the licensee identified that the Unit 2 HPCI low flow switch trip setpoint was below the technical specification limit of 600 gallons per minute. The root cause of this event was determined to be setpoint drift.

As part of the immediate corrective actions, the switches were re-calibrated to within tolerance. In addition, a review of calibration data did not reveal an adverse trend. As part of the long-term actions, more frequent switch calibrations were scheduled.

During this inspection, the team verified that the HPCI low flow switch had not been found out-of-tolerance during switch calibrations following the event.

Technical Specification 3.2.b required that the HPCI low flow switch setpoint be greater than 600 gallons per minute. The event discussed above is an example where this requirement was not met and was a violation. However, this licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-237/98023-05; 50-249/98023-05).

E8.22 (Closed) LER 50-237/97016-00: Autostart of the 2/3 Diesel Generator (DG) Due to Knowledge Deficiency.

An unqualified toggle switch was found installed on the Unit 2/3 DG feeder breaker which supplied the Unit 3 safety-related bus. Following installation of a replacement breaker, the 2/3 DG was started for an operability run. After waiting 3 to 5 minutes, the high voltage operator (HVO) raised DG speed locally and the DG tripped unexpectedly. The HVO reset the local annunciators and depressed the alarm reset pushbutton. The HVO believed that the pushbutton would only reset the low cooling water pressure trip indicator. However, the pushbutton also reset the sealed-in trip logic for the DG (as designed), and caused the DG to restart. The root cause of the event was determined to be a knowledge deficiency in the operation of the DG local control panel alarm reset pushbutton.

Corrective actions included discussion of this event with the other operating teams by the involved operating team. In addition, applicable procedures were revised.

During this inspection, the team verified that appropriate personnel had been trained on this event and that applicable procedures were revised. No deficiencies were identified.

E8.23 (Closed) LER 50-237/97018-00/01: Unexpected HPCI Steam Supply Isolation During Testing.

A HPCI primary containment isolation signal was received while testing the HPCI steam line low pressure isolation circuitry. The licensee identified the root cause of the event as the spurious actuation of the HPCI low reactor pressure isolation circuitry since no component failures were identified. However, suspect relays were replaced and successfully tested.

During this inspection, the team reviewed the root cause analysis. No deficiencies were identified.

E8.24 (Closed) LER 50-249/96016-00/01: Redundant DGs Inoperable While in Mode 1.

During preparations for running the Unit 3 DG, about 13 gallons of coolant was found inside the DG airbox. The licensee identified the root cause of the coolant leak as the failure of the gasket between the cylinder and discharge elbow on DG cylinder 7.

As part of the licensee's corrective actions, the cylinder and piston assembly was replaced. Subsequently, the DG was re-tested satisfactorily.

During this inspection, the team verified that the licensee's corrective actions had been properly implemented. No deficiencies were identified.

E8.25 (Closed) LER 50-249/97006-00: Preconditioning 125 VDC Battery Prior to Service Discharge Test.

On June 10, 1997, a service discharge test on the Unit 3 125 VDC battery was performed. Following that test, the licensee identified that the battery was pre-conditioned since an equalization charge and maintenance activities to remove and clean intercell connectors were completed just prior to the test. The root cause of the problem was an inadequate technical review of the modified performance test prerequisites because the technical reviewers did not identify or adequately question the as-found requirements for service discharge test performance.

As part of the immediate corrective actions, the service discharge test documentation was reviewed to assure that an acceptable test had been performed despite the issues identified above. In addition, personnel were trained on this event and applicable procedures were revised.

During this inspection, the team verified that the revised testing procedure contained appropriate caution statements and that cognizant engineering personnel were aware of the issue and activities that would constitute pre-conditioning.

Technical Specification (TS) 4.9.C.4 required that at least every 18 months verify that the 125 VDC batteries are operable by verifying that the battery capacity is adequate to supply all of the emergency loads when subjected to a battery service test. The TS

4.9.C bases stated that the battery service test is a test of the battery "as found" capacity to satisfy the design requirements of the system. Therefore, the circumstances in which the battery was pre-conditioned through the performance of maintenance just prior to the service test is an example where the requirements of TS 4.9.C.4 were not satisfied and was a violation. However, this licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-237/98023-06; 50-249/98023-06).

V. Management Meetings

X1 Exit Meeting Summary

The team presented the inspection results to members of licensee management at an exit meeting on September 23, 1998. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Swafford, Station Manager, Dresden
P. Chabot, Engineering Manager, Dresden
R. Peak, Design Engineering Manager, Dresden
D. Willus, System Engineering Manager, Dresden
P. Planning, Engineering Programs Supervisor, Dresden
D. Winchester, Nuclear Oversight Manager, Dresden
R. Fisher, Maintenance Manager, Dresden
F. Spangenberg, Regulatory Assurance Manager, Dresden
R. Whalen, Primary Group Leader, System Engineering
G. Abrell, NRC Coordinator, Regulatory Assurance, Dresden

LIST OF INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 37551	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 37001:	10 CFR 50.59 Safety Evaluation Program
IP 92903:	Followup - Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/98023-01;50-249/98023-01	VIO	Temporary Alteration Program Procedure Violations
50-237/98023-02;50-249/98023-02	VIO	Inservice Testing Deficiencies
50-237/98023-03;50-249/98023-03	NCV	Inservice Tests Not Performed As Required
50-237/98023-04;50-249/98023-04	NCV	CCSW Temperature Outside Design Basis
50-237/98023-05;50-249/98023-05	NCV	HPCI Low Flow Setpoint Found Outside TS Limit
50-237/98023-06;50-249/98023-06	NCV	Preconditioning 125 VDC Battery for Discharge Test

Closed

50-237/93019-01;50-249/93019-01	EEI	FSAR Change Lacked Prior NRC Approval
50-237/93019-02;50-249/93019-02	EEI	Inadequate Safety Evaluation for FSAR Change
50-237/94014-03;50-249/94014-03	VIO	Untimely Corrective Actions for HPCI Cooler Fans
50-237/96002-06;50-249/96002-06	VIO	4kV Breaker Maintenance and FME Problems
50-237/96009-05;50-249/96009-05	VIO	Ineffective Corrective Actions for Battery Calculation
50-237/96014-02;50-249/96014-02	VIO	Failure to Enter DG Test Valve Information Into VETIP
50-237/97007-05;50-249/97007-05	VIO	Failure to Test 250 VDC Battery in As-Found Condition
50-237/97019-01;50-249/97019-01	VIO	Inadequate DG Operating Procedures
50-237/97028-01;50-249/97028-01	VIO	Failure to Follow PIF Process
50-237/96014-03;50-249/96014-03	URI	TALT Adequacy for Battery Room Heater Installation
50-237/96201-18;50-249/96201-18	IFI	Failure to Test Unit 2 125 VDC Battery
50-237/96201-24;50-249/96201-24	IFI	Adequacy of Replacement Breaker
50-237/87012-02	LER	HPCI Turbine Trips Due to Hydraulic System Problems
50-237/94021-02	LER	HPCI Turbine Vendor Test Specification Change
50-237/96011-00/01	LER	Inservice Tests Not Performed As Required
50-237/96015-00	LER	HPCI Inoperable Due to High Discharge Temperature.
50-237/96019-00/01	LER	250 VDC Battery Not Tested Although Required
50-237/96020-00/01	LER	CCSW Temperature Outside Design Basis
50-237/97005-00	LER	HPCI Low Flow Setpoint Found Outside TS Limit
50-237/97016-00	LER	Autostart of the 2/3 DG Due to Knowledge Deficiency
50-237/97018-00/01	LER	Unexpected HPCI Steam Supply Isolation During Testing
50-249/96016-00/01	LER	Redundant DGs Inoperable While in Mode 1
50-249/97005-00/01	LER	Preconditioning 250 VDC Battery for Performance Test
50-249/97006-00	LER	Preconditioning 125 VDC Battery for Discharge Test

Discussed

50-237/96012-03;50-249/96012-03	URI	Cable Ampacity Concerns
50-237/98007-02;50-249/98007-02	URI	CCSW Pump Operability - Post Dam Failure

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
CCSW	Containment Cooling Service Water
CFR	Code of Federal Regulations
ComEd	Commonwealth Edison
CST	Condensate Storage Tank
DAP	Dresden Administrative Procedure
DCP	Design Change Package
DEP	Dresden Engineering Procedure
DES	Dresden Engineering Surveillance
DG	Diesel Generator
DISACM	Dresden IST Surveillance Acceptance Criteria Manual
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DTS	Dresden Technical Surveillance
EEI	Escalated Enforcement Item
ER	Engineering Request
FME	Foreign Material Exclusion
gpm	Gallons Per Minute
GSLO	Gland Seal Leakoff
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Coolant Injection
HVO	High Voltage Operator
IFI	Inspection Followup Item
IP	Inspection Procedure
IST	Inservice Testing
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NSWP	Nuclear Station Work Procedure
NTS	Nuclear Tracking System
OM-6	Operations and Maintenance Standard Part 6
OPEX	Operating Experience
PDR	Public Document Room
PIF	Problem Identification Form
SBGT	Standby Gas Treatment
TALT	Temporary Alteration
TS	Technical Specification
TSUP	Technical Specification Upgrade Program
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VDC	Volts Direct Current
VIO	Violation
°F	Degrees Fahrenheit

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC team reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

Procedures

CAP-1	Root Cause Investigation and report Instructional Guide
CAP-2	Coding and Trending Instructional Guide
CAP-3	Problem Identification Form Threshold Information Instructional Guide
CWPI-NSP-AP-1-1	Event Response Guidelines
CWPI-NSP-AP-1-2	Quarantine of Areas, Equipment and Records
CWPI-NSP-AP-1-8	Licensee Event Report/Security Event Report
CWPI-NSP-AP-1-10	Operating Experience
CWPI-NSP-AP-1-12	Action Tracking Program
DAP 05-08	Control of Temporary System Alterations
DAP 07-05	Operating Charts, Logs, and Records
DEP 40-06	Safety-Related Motor Valve Data and Settings
DMS 4100-05	Unit 2/3 Fire Pump Bay Intake Screen and Bay Quarterly Inspection
DMS 5700-02	Roll-O-Matic Air Filter, Quarterly Surveillance and Preventive Maintenance
DMS 5700-01	Ventilating System Fan Damper Surveillance
DOS 1500-02	Containment Cooling Service Water Pump Test and Inservice Test
DOS 1500-12	Containment Cooling Service Water Loop Flow Verification
DOS 2300-03	HPCI System Operability Verification
DOS 6600-11	Diesel Generator Cooling Water Pump Test
DTS 7500-11	DOP Testing of Unit 2/3 SBGT HEPA Filters
NO-08	Use of Field Monitoring Program
NOD-OA-26	Intra-ComEd OPEX/Lessons Learned Program
NSP-AP-1002	Plant Operations Review Committee
NSP-AP-1004	Corrective Action Program Process
NSP-AP-2004	Corrective Action Program Process Roles and Responsibilities
NSP-AP-3004	Corrective Action Process Manual
NSP-AP-3009	Self-Assessment Program
NSP-AP-4004	Corrective Action Program Procedure
NSWP A-04	10 CFR 50.59 Safety Evaluation Process

Audits and Self-Assessments

NGG 98-05	Engineering Audit
NTS 237-251-98-00100	Compliance with the Standardized Corrective Actions Program
NTS 237-251-98-00600	Conduct of Regulatory Assurance
NTS 237-251-98-01900	Corrective Action Records Timeliness of Corrective Action Completion
NTS 237-251-98-02100	Engineering Rapid Response Team Effectiveness

NTS 237-251-98-02500	Engineering Effectiveness
QAA 12-96-14	Dresden Site Quality Verification Audit, Engineering
QAA 12-96-11	Site Quality Verification Audit, Nuclear Engineering Procedures
QAS 12-97-07	Surveillance of Operability Determination Timeliness
QAS 12-98-10	Pre-Conditioning Review
QAS 12-98-20	Assessment of Response and Problem Resolution of Electrical System and Component Failures and Abnormalities
QAS 12-98-19	Assessment of Dresden Engineering Assurance Group

OPEX Documentation

NTS 237-455-97-60700	SIL 607: T-Type Scram Solenoid Pilot Valve Inadvertent Scram
NTS 237-455-97-61000	SIL 610: Incorrect Core Disc Material in Scram Solenoid Pilot Valves
NTS 237-404-98-1800	Potential for Loss of High Pressure Safety Injection During Testing
NTS 2374049715000	Potential Common Cause 4-Kilovolt Breaker Failures
NTS 237-455-95 58401S1	SIL 584 Supplement 1: Scram Solenoid Pilot Valve Delayed Response

Root Cause Reports

RCR 249-200-98-00200	Unit 3 Shutdown Due to Scram Discharge Volume Level Switch Issue
RCR 237-200-98-00800	Failure of Pump to Produce Adequate Flow

Modifications

DCP 9500058 Standby Gas Treatment System "A" Valve Positioner
DCP E-12-0-96-213 Unit 2/3 250 VDC Battery Charger Anchorage
DCP E-12-3-96-225 Unit 3 125 VDC Battery Charger Anchorage
DCP 9100070 Unit 2/3 Diesel Generator Fuel Oil Filter Assembly Replacement
DCP E12-0-96-211 Diesel Generator Air Start Pressure Switch Replacement
DCP E12-3-96-228 Upgrade Unit 3 Low Pressure Coolant Injection Heat Exchanger Partition Plate
DCP 9700245 Emergency Core Cooling System Room Cooler Cooling Water Supply

Temporary Alterations

TALT 1-02-98	Install Pipe Repair Clamp on Fire Protection Header
TALT 2-33-96	Install Portable Heaters Outside 125/250 VDC Battery Rooms
TALT 2-01-97	Install Portable Heaters Outside 125 VDC Battery Rooms
TALT 2-12-97	Install Equipment to Monitor 2A Reactor Feed Pump
TALT 2-15-97	Bypass Flow Switch on 2A Reactor Feed Pump Ventilation Fan
TALT 2-07-98	Install Blind Flange on Unit 2C CCSW Vault Room Cooler
TALT 2-14-98	Lock Open Reactor Feed Pump Ventilation Intake Damper
TALT 2-19-98	Supply Air to Unit 2 Fuel Pool Cooling Reject Air-Operated Valve 2-1901-57
TALT 3-04-97	Open Unit 3 Station Blackout Diesel Generator Intake Damper
TALT 3-12-97	Lock Open Reactor Feed Pump Ventilation Intake and Exhaust Dampers and Lock Shut Recirculation Damper
TALT 3-06-98	Install Recorder in Reactor Protection System
TALT 3-08-98	Install Plug on Leaking Indexing Tube C-9

10 CFR 50.59 Safety Evaluations

1996-04-247 Unit 2/3 Diesel Generator Ventilation Fan Wiring Temporary Alteration
1997-01-006 Revise UFSAR to Address 125 VDC Battery Qualified Life
1997-01-056 Standby Gas Treatment System Damper Thermal Overload Setpoint Change
1997-01-065 Place and Maintain 24/48 VDC Battery Charger in Equalize Mode
1997-01-077 Upgrade HPCI Gland Seal Leakoff to Safety-Related
1997-02-078 Permanent Change to Reactor Building Volume
1997-02-108 Replace Heat Exchanger 2A-1503 Partition Plate
1997-02-152 Revise UFSAR to Correct CCSW Piping Elevation Error
1997-04-222 Upgrade HPCI Gland Seal Leakoff to Safety-Related
1997-04-233 Install Additional Valves/Piping in CCSW System
1997-04-242 HPCI Pump Suction Alignment to Condensate Storage Tank
1998-01-011 Temporary Alteration to Install Blank Flanges in Unit 2 CCSW Pump Cooler
1998-01-058 Change TS Bases to Reflect Unit Auxiliary Transformers as Sources of Power
DES 8300-28, Revision 4, dated 3/20/98
DES 8300-33, Revision 5, dated 7/22/98
DOP 2300-01, Revision 15, dated 6/13/97
DOP 2300-03, Revision 24, dated 6/11/97
DOP 2300-04, Revision 9, dated 8/6/98
DOS 1500-02, Revision 28, dated 7/7/98
DOS 1500-12, Revision 6, dated 4/3/98
DOS 2300-03, Revision 51, dated 4/30/98
DTS 1500-05, Revision 2, dated 4/7/98
DTS 2300-02, Revision 1, dated 4/30/98
DSSP 0200-L, Revision 7, dated 4/10/98

Operability Determinations

96-051 Upgrade of HPCI Gland Seal Leakoff and Exhauster System to Safety-Related
97-081 Minimum Water Level in CCSW Intake Bay
97-102 Cracked Motor Mounting Rail on Unit 2 Diesel Generator Air Start Compressor
97-107 CST Level For Transfer of HPCI From CSTs to Torus
98-009 HPCI Drain Pot Discharge Piping Design Basis Analysis
98-024 Ability of Standby Gas Treatment System to Remove Iodine

Calculations

DRE96-0026 Flange Calculation for HPCI Discharge Line No. 2-2304-12-DX
DRE96-0031 Standby Gas Treatment System Support Anchor Thread Engagement
DRE96-0040 Breaker and Thermal Overload Heater Sizing for Replacement Breaker
DRE96-0126 Motor Terminal Voltage Calculation for Dresden 250 VDC MOVs
DRE96-0214R3 Differential Pressure Between LPCI/CCSW Heat Exchangers
DRE97-0059 HPCI Leakoff Piping Analysis
DRE97-0161 HPCI Gland Seal and Exhauster Analysis
DRE97-0252 Sizing of CCSW Pipe & Flow Limiting Orifices for ECCS Room Coolers

DRE98-0077	Dresden HPCI Room Thermal Response With Reduced Room Cooler Capability
D3-HPCI-08B(C)	HPCI Steam Supply Line Piping Analysis
E-001-2301	Motor-Operated Valve Terminal Voltage Calculations
NED-E-EIC-0065	Thermal Overload Heater Sizing for Direct Current Motor-Operated Valves
NED-EIC-DR0007	Valve Actuator Motor Terminal Voltage Calculation
29.0202.1233.018	Addition of Local Leak Rate Test Flange to HPCI Turbine Exhaust System

Completed Surveillance Documentation

DES 8300-28	Unit 2, 125 Volt Station Main Battery Service Test
DOS 6600-01	Diesel Generator Surveillance Tests
DOS 6600-02	Reversal of Diesel Generator Cooling Water Flow
DOS 6600-04	Bus Undervoltage and ECCS Integrated Flow Test for Unit 3 DG
DOS 6600-05	Bus Undervoltage and ECCS Integrated Flow Test for Unit 2 DG
DOS 6600-06	Bus Undervoltage and ECCS Integrated Flow Test for Unit 2/3 DG
DOS 6600-07	Testing Swing Bus Protective Relays and Auto Transfer Function
DOS 6600-08	Quarterly DG Cooling Water Pump Test for Inservice Testing Program
DOS 6600-09	Testing of ECCS Undervoltage and Degraded Voltage Relays
DOS 6600-11	Diesel Generator Cooling Water Pump Test
DOS 6600-12	Diesel Generator Endurance and Full Load Rejection Test
DOS 6600-14	Quarterly Fuel Oil Transfer Test for Inservice Testing Program
DTS 7500-11	DOP Testing of 2/3 SBGT HEPA Filters

Licensee Event Reports

LER 50-237/87012-02	HPCI Turbine Trips Due to Hydraulic System Problems
LER 50-237/94021-02	HPCI Turbine Vendor Test Specification Change
LER 50-237/96011-00/01	Inservice Tests Not Performed As Required
LER 50-237/96019-00/01	250 VDC Battery Not Tested Although Required
LER 50-237/96020-00/01	CCSW Temperature Outside Design Basis
LER 50-237/97005-00	HPCI Low Flow Setpoint Found Outside Technical Specification Limit
LER 50-237/97016-00	Autostart of Unit 2/3 Diesel Generator Due to Knowledge Deficiency
LER 50-237/97018-00	Unexpected HPCI Steam Supply Isolation During Testing
LER 50-249/96016-00/01	Redundant Diesel Generators Inoperable While in Mode 1
LER 50-249/97005-00/01	Preconditioning 250 VDC Battery for Modified Performance Test
LER 50-249/97006-00	Preconditioning 125 VDC Battery for Service Discharge Test

Engineering Requests

ER9500105	Evaluate Standby Gas Treatment System Bolt Mounting
ER9604771	Safety Function of 2(3)-2301-28/31
ER9700096	Calibrate Repair or Replace Humidity Indicator
ER9700406	Upgrade HPCI Piping to Safety-Related
ER9702718	Determine Standby Gas Treatment System Operating Limits
ER9703211	Unit 2 Scram Investigation
ER9703754	Disassemble/Inspect Valve to Meet Inservice Testing Requirements

ER9704547	Revise Setpoint of 2(3)-4941-6 TO 12
ER9800015	HPCI Minimum Lube Oil Temperature in Standby
ER9800072	Temporary Alteration for 2C CCSW Pump Room Cooler Inlet
ER9800097	Standby Gas Treatment System Crosstie Valve Manual Stops
ER9800471	Calibrate Unit 2 Control Loop and Instrumentation
ER9800524	Feed Pump Ventilation Calibration Data
ER9800675	Repair 2C CCSW Pump Vault Cooler and Replace Standpipe
ER9801140	HPCI Seal Wall In-Leakage Engineering Guidance
ER9801248	Perform Inservice Testing Relief Valve Surveillance on 2-2301-53
ER9801312	Evaluation for 2-2301-8 Valve Disk Guides
ER9801358	Request Justification for Threaded Rod
ER9801374	HPCI Auxiliary Oil Pump Modification
ER9802345	Valve Will Not Close Due to Bent Valve Stem

Problem Identification Forms

PIF 1997-0058	Unit 2 Station Blackout Diesel Generator Battery Room Reached 66°F
PIF 1997-0292	Unit 2 Battery Temperature Dropped Due to Ventilation Problems
PIF 1997-0299	Low Ambient Temperature Caused Low Battery Electrolyte Temperature
PIF 1997-0302	Uninterruptible Battery Cell Found Out of Tolerance
PIF 1997-0383	Fuel Oil Intrusion in Unit 3 Trackway
PIF 1998-0766	Additional Run Time on Unit 2/3 DG for Post-Maintenance Testing
PIF 1997-0811	Interlock Door Alarming Due to Vent Fans
PIF 1997-0936	Quarterly Evaluations Not Performed
PIF 1997-1044	Ambient Temperature in Unit 2 Alternate 125 VDC Battery Area
PIF 1997-1255	Unit 1 Battery Specific Gravity Found Out of Tolerance
PIF 1997-1859	Improper Check Valve Installed
PIF 1997-2217	Lack of Review on Nuclear Operation Notifications
PIF 1997-2381	Unit 3 Diesel Generator Hunting Frequency
PIF 1998-3580	Hydrogen Bottle Mixed With Nitrogen Bottles
PIF 1998-3859	Unit 2 Diesel Generator Frequency Swings and Trouble Alarm
PIF 1997-4046	Filling Trucks From Unit 1 Diesel Generator Fuel Tank
PIF 1997-4230	12/20 Calculations in Error
PIF 1997-4971	Unsampled Fuel Put in Unit 1 Diesel Generator Fuel Tank
PIF 1997-5684	3/9 Air Start Motors Failed Bench Test
PIF 1997-5711	Dipsticks and Methodology for Checking Oil Different
PIF 1997-5884	Six Gallon a Day Fuel Leak
PIF 1997-6215	Drain Backs Up Fuel Oil
PIF 1997-6405	Unit 2/3 Vent Fan Failure to Transfer to Unit 3
PIF 1997-6527	Technical Specification Surveillance Overdue
PIF 1997-6884	Preconditioning of Diesel Generators
PIF 1997-7814	Human Factor Issue - Diesel Generator Voltmeter in Unit 2 Control Room
PIF D1998-0632	HPCI Small Bore lines Do Not Meet UFSAR Design Criteria
PIF D1998-0942	Reportability Screening for HPCI Drain Piping Support Operability Determination
PIF D1998-2111	Unit 3 Shutdown Due to Inoperable Scram Discharge Volume Level Switches
PIF D1998-3024	Iodine Loading of the Standby Gas Treatment System

PIF D1998-4183	Failure of 3C CCSW Pump to Produce Adequate Flow
PIF D1998-4776	Ineffective Corrective Action Associated with Root Cause Investigation 237-900-79-04800
PIF D1998-4784	Nuclear Oversight Identifies Faulty Switch During Receipt Inspection
PIF D1998-4791	Reactor Low Pressure ECCS Permissive Out of Tolerance
PIF D1998-4794	Discrepancies in Engineering Score Card Implementation
PIF D19984807	Locked Isolation Valves Found with Broken Locking Device
PIF D1998-4846	Unit 2 250 VDC Charger Not Picking up Load
PIF D1998-4896	Diesel Generator System Notebook Not Recording System Walkdowns
PIF D1998-4924	250 VDC discovered at 290.4V on Equalize Charge
PIF D1998-4925	Inconsistency with Application of Operability Determinations
PIF D1998-4978	Insufficient Time to Review Nuclear Tracking System Items on Due Date
PIF D1998-4988	Receipt/Shipment Inspection Certification Paperwork Not Onsite
PIF D1998-5045	Silicon Controlled Rectifiers Installed Backwards During Unit 2 250 VDC Battery Charger Maintenance
PIF D1998-5141	Excessive Noise and Vibration With Unit 2 250 VDC Battery Charger
PIF D1998-5151	Unit 2 250 VDC Battery Charger Arcing at Firing Module
PIF D1997-5535	Unit 2/3B Standby Gas Treatment System Inoperable Due to Surveillance
PIF D1997-6056	Standby Gas Treatment System Valve 3-7503 Missing Four Bolts
PIF D1997-8211	Low Level Setpoint Margin Reduced for CST due to HPCI Vortexing
PIF D1997-8329	HPCI Inoperability Due to Vortexing in the Condensate Storage Tanks
PIF D1997-8451	Operability Determination 97-107 Performed Due to HPCI Vortexing

Strategic Reform Initiatives

NGG-3, Action Step 8	Implement the System Health Indicator Program.
NGG-8, Action Step 3	Define and issue common procedures for the IST, Inservice Inspection, Maintenance Rule, Appendix R, and Generic Letter 89-13 programs.
NGG-8, Action Step 4	Implement the engineering work management system.
NGG-13, Action Step 1	Assess effectiveness of the Safety Review Board, Plant Onsite Review Committee, and departmental self-assessment processes. Also assess the effectiveness of other programs such as the Corrective Action Process and OPEX.
NGG-13, Action Step 4	Implement a generic lessons-learned process.

Miscellaneous Documents

NRC Safety Evaluation for License Amendment Nos. 157/152
Inservice Testing Basis Document, Revision 0
Inservice Testing Program, Revision 3
Effectiveness Review: Foreign Material in CCSW Pump
JSPLTR 95-0020, Change of Commitment for Submittal of Proposed License Amendment
Document 0005627080, Engineering's HPCI Low Flow Setpoint Recommendations
Document 0005581078, DE&S Letter 1: HPCI Drain Pot Piping Met Operability Limits
Document 0005581383, DE&S Letter 2: HPCI Drain Pot Piping Met Operability Limits
Plant Engineering Handbook, Revision 1
Dresden IST Surveillance Acceptance Criteria Manual (DISACM)