U.S. NUCLEAR REGULATORY COMMISSION

REGION III

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Report No:	50-249/97021(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Dresden Generating Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450-9765
Dates:	October 21, 1997 through January 27, 1998
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EXECUTIVE SUMMARY

Dresden Generating Station, Units 2 and 3
NRC Inspection Report 50-237/97021(DRS); 50-249/97021(DRS)

An announced core inspection that reviewed the engineering and technical support (E&TS) organization's effectiveness in the performance of routine and reactive site activities including identification and resolution of technical issues and problems. As a result of the inspection, three violations (VIOs) of Nuclear Regulatory Commission (NRC) requirements were identified and one unresolved item (URI) was issued.

- Overall the inspection concluded that the engineering staff was effective in the identification and resolution of technical issues. Self-assessments exhibited a pro-active trend in the attempt to disclose performance problems within the engineering organization. The quality of engineering activities was in most cases technically sound. (Section All)

- The team had concerns that the UFSAR did not accurately characterize the plant's design-basis or the plant's capability to respond to a potential Dresden Lock and Dam failure. As a result, the team concluded that further review by the licensee and NRC was required. An NRC URI was initiated to document these concerns. (Section E3.4; URI 50-237/249-97021-01(DRS))

- The team concluded that all commitments and corrective actions identified by Confirmatory Action Letter (CAL) No. RIII-96-016, dated November 21, 1996, including those activities associated with the Dresden Engineering Assurance Group (DEAG) have satisfied NRC requirements. The CAL was closed. (Section E6).

- In November 1994, the licensee identified that a prior inadvertent change to the Dresden Station's control room ventilation system design deleted the automatic smoke purge mode transfer capability. From November 1994 to March 1996, the licensee failed to perform a written safety evaluation to provide the bases for the determination that the change did not involve an unreviewed safety question. (Section F2; VIO 50-237/249-97021-02(DRS))

- From November 1994 through November 21, 1997, the Fire Protection Report, referenced as part of the UFSAR, had not been updated and the required revision updates submitted to the NRC. (Section F3; VIO 50-237/249-97021-03(DRS))

- As of November 21, 1997, the fire pre-plans had not been updated since September 1992. (Section F3; VIO 50-237/249-97021-04(DRS))

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 Performance and Effectiveness

a. Inspection Scope (IP37550; IP40500)

The purpose of the inspection was to evaluate the effectiveness of the E&TS organization in the performance of routine and reactive site activities including identification and resolution of technical issues and problems. The inspection focused on system engineering functions, modifications, technical problem resolution, and engineering support to other plant organizations. In addition, the licensee's corrective action process was evaluated.

The criteria used to assess the E&TS performance was quality of technical work produced, understanding of plant design, and active involvement in preventing and solving plant problems.

b. Observations and Findings

Overall, the engineering staff was effective in the identification and resolution of technical issues. The inspection showed engineers to be knowledgeable and involved with the work conducted in their respective areas of responsibility. Engineers and immediate supervisors were cognizant of the current status of assigned systems and components, as well as, recent problems and deficiencies that had been identified. The quality of the reviews conducted by the engineering staff was in most cases technically sound. However, minor discrepancies were observed in many of the engineering products and activities. These discrepancies indicated that the licensee's engineering staff should be thorough and exhibit more attention to detail. The DEAG reviews were in most cases thorough and technically sound. However, the team was concerned that many DEAG members were no longer employed at the Dresden site and such loss of experienced personnel might degrade the licensee's ability to maintain an improving trend in engineering performance.

c. Conclusions

The inspection team concluded that conduct of engineering was satisfactory.

E1.2 Problem Identification and Root Cause Determination

a. Inspection Scope (IP37550; IP40500)

The team reviewed several PIFs generated by the plant staff and verified whether the PIFs were properly processed for root cause determination and corrective actions.

b. Observations and Findings

The team reviewed selected PIFs for adequate description of the problem and to verify whether the PIFs were properly prioritized and followed up as necessary. The team also reviewed whether the PIFs were reviewed for root cause determination and corrective actions when required. A nuclear tracking system (NTS) number was assigned to follow up PIFs. The team reviewed a few NTS items to verify whether they were adequately followed up by the licensee for completion. The team noticed that the reasons for NTS due date extensions were not always adequately justified. An example was PIF 97-12037 dated January 30, 1997, regarding allowable battery temperatures. This PIF was tracked by NTS Item 237-201-97-12001. The reason for extending this NTS item for about five months was "to provide new DC system engineer time to evaluate other options."

The team attended a PIF screening meeting on November 3, 1997. The team noted that the department managers/supervisors were present as necessary. The PIFs received were adequately discussed and assigned to the responsible departments for further follow up.

c. Conclusion

The team concluded that a low threshold exists for generation of PIFs. The team observed that the PIFs were promptly processed and assigned to a department for follow up. The root causes for important PIFs were identified for further corrective actions. However, adequate justification was not always provided for extending corrective action due dates.

E2 Engineering Support of Facilities and Equipment

E2.1 4kV Breaker Auxiliary Switch Failures

a. Inspection Scope (IP 37550; IP40500)

The team reviewed the licensee's corrective actions for Merlin-Gerin 4.1kV breaker auxiliary switch failures.

b. Observations and Findings

The licensee's corrective actions involved the installation of nylon tie-wraps around the breaker's auxiliary switches. The auxiliary switches on the breakers were made of a phenolic material and were observed to develop cracks at the Dresden and Quad Cities Stations.

The manufacturer and the local distributor of the breakers, Pacific Breaker Systems, Inc. and Golden Gate Switchboard Co., were informed of the defects. A 10 CFR Part 21 notification was issued by Golden Gate Switchboard Co. on April 11, 1997, regarding the cracking and breakage of the circuit breaker auxiliary switches in the mounting area.

The cracking and breakage in the mounting area resulted in unacceptable contact resistance readings.

The licensee developed a temporary fix that used nylon tie-wraps around the two auxiliary switches on each breaker and qualified the fix for a period of 18 months. The qualification test was performed by testing the breaker for 225 cycles and performing a seismic test at Wyle Laboratories. The Plant Operations Review Committee approved the modification for only one plant operating cycle.

The team noted that the root cause(s) for the failure of the auxiliary switches had not been identified by the manufacturer. Potential corrective actions, such as a change in the type of switch material, had not been provided to the licensee.

However, the licensee performed a root cause evaluation during August 1997, which concluded that the primary root cause(s) for the failures were a design weakness in the auxiliary switch mounting and inappropriate torque values for the mounting T-bolts. The evaluation led to the licensee's immediate corrective action of using nylon tie-wraps around the auxiliary switches.

For a semi-permanent fix, the licensee intended to qualify the nylon tie-wraps for a period of six years. The breakers were tested with the nylon tie-wraps for 750 cycles at Commonwealth Edison Company's (ComEd's) C-Team facility and seismically qualified at the Wyle Lab for six years. The licensee intended to use stainless steel U-bolts (in place of the tie-wraps) as a permanent fix.

The licensee's root cause evaluation indicated that the original design created tensile forces where the phenolic material was not sufficiently strong. The team noted that the tie-wraps had reduced the tensile forces to some extent; however, the licensee's root cause effort did not address the weakness of the switch material and the potential need to change to an alternate (stronger) material that could withstand the higher tensile forces.

The team expressed concern that the tie-wrapped auxiliary switches were considered for extended use, prior to the completion of the manufacturer's root cause evaluation and without considering an alternate material. The team considered the potential for cracking the auxiliary switches during operation remained even with the tie-wraps or U-bolts in place.

c. <u>Conclusion</u>

The licensee's actions to temporarily extend the life of the auxiliary switches with nylon tie-wraps were acceptable. However, the licensee's and vendor's failure to address the weakness of the phenolic material and not considering an alternate (stronger) material for the auxiliary switches was considered a weakness.

E2.2 Plant Walkdowns

a. Inspection Scope (IP 37550)

The team walked down several areas of the plant to assess the material condition of equipment and general plant condition.

b. Observations and Findings

The team walked down the intake structure and some electrical areas, such as the diesel generators, switchgear areas, and battery rooms.

The areas walked down were generally kept clean. The equipment observed, such as safety-related batteries, diesel generators and safety-related electrical switchgear were maintained in good condition.

c. Conclusions

The team concluded that the plant areas walked down were well maintained and no deficiencies were observed.

E3 Engineering Procedures and Documentation

E3.1 Design Change Packages, Modifications and Temporary Alterations

a. Inspection Scope (IP 37550)

The team reviewed the following design change packages (DCPs), modifications and temporary alteration (Temp Alt):

- DCP 9700202 — Install 70 Amp Breaker in Cubicle 39-2-C3
- DCP 9700207 — Change out of Control Transformers in Turbine Oil Tank Vapor Extractor Breaker
- E12-3-95-224 — Limit Switch Replacement on Motor Operated Valve (MOV) 3-205-24
- M12-0-97-001A — Auxiliary Electrical Equipment Room Heating, Ventilation, and Air Conditioning (HVAC) Modification
- M12-2-85-302 — Unit 2 - 125 Volt DC Charger Upgrades
- M12-3-96-008 — Time Delay Addition on Valve 3-2301-15
- P12-3-94-284 — Gearset Replacement on MOV 3-1501-28B
- Temp Alt III-09-97 — Install Portable Air Compressor Outside Crib House

b. Observations and Findings

The team observed that the above DCPs and modifications clearly described the proposed alterations and justifications. Each design change contained an adequate 10 CFR 50.59 screening or safety evaluation. The design issues worksheets considered several additional issues. Adequate interdepartmental reviews were performed as necessary.

The team reviewed several calculations made in support of the design changes. The calculations included acceptable assumptions and were adequately reviewed and approved. No problems were identified with the calculations.

Several work requests were reviewed that implemented the design changes. The team found that the design changes did not always include the results of post-modification testing (PMT). An example was the PMT performed for DCP E12-3-95-224 (level switch replacement on MOV 3-205-24) that was completed on June 11, 1997. The team had to obtain a copy of the completed procedure from central files to verify whether the PMT was completed.

The team observed that Temp Alt III-09-97 provided the reasons for the alteration, an adequate safety evaluation and a date for the expected removal of the alteration (five months after installation). The team's walk down of the temporary alteration found the Temp ALT installation in good condition.

c. Conclusion

The team concluded that the modifications, DCPs and temporary alteration reviewed were adequately implemented. However, some DCPs did not include PMT results.

E3.3 Calculations/Evaluation

a. Inspection Scope (IP 37550)

The team reviewed the following calculations/evaluation and associated DEAG reviews:

- Calculation DRE 97-0171, "Determination of Acceptance Criteria for CCSW One and Two Pump NPSH Testing - Units 2 & 3," Revision 0
- Calculation DRE 97-0172, "Vortexing at CCSW Intakes - Units 2 & 3," Revision 0
- Document ID # 5543459, "Evaluation, Re: Low Pressure Coolant Injection (LPCI) System, Hydraulic Calculation for Containment Cooling and Containment Cooling Spray Modes," dated October 29, 1997

b. Observations and Findings

Calculation DRE 97-0171:

The team observed that the calculation used the pump suction centerline as the pump datum plane.

The team determined that this method of calculation was non-conservative and introduced an error into the calculation. The team's assessment of the DEAG review identified that the DEAG did not detect this error, but did note conservatism in the calculation. The team determined that the conservatism compensated for the non-conservative error.

The team observed that not all of the logic thought processes and equation derivations were documented in the calculation, making the methodology more difficult to understand (e.g., the gage error effect was not bounded). These weaknesses indicated a need for more attention to detail. The DEAG review recommended similar clarifications to make the calculation a better source of information for future users.

Calculation DRE 97-0172:

The Vortexing calculation stated the maximum CCSW intake flow rate was 7,200 gpm. The calculation's design input reference was the Hydraulic Institute Standards, ANSI/HI 1.3.3.6.1-1.3.3.6.3, American National Standard for Centrifugal Pumps, approved May 23, 1994.

As flow rates increase the distance between intake centerlines must be increased to prevent vortexing. The calculation identified the actual distance between the CCSW intake centerlines as 42 inches. The design input reference stated the minimum distance between the intake centerlines should be 52 inches for a 7,200 gpm flow rate and that at 42 inches the flow rate should be limited to 5,400 gpm.

Although the actual distance did not meet the design input reference's recommendation for a 7200 gpm flow rate, the calculation concluded that the distance was acceptable because the CCSW system was required to be maintained at 20 psid higher than the LPCI system. The 20 psid differential was maintained by throttling the CCSW flow rate below 7,200 gpm.

The team was concerned that the amount of throttling was not specified and given the right operating configuration, vortexing might occur due to insufficient distance between intake centerlines. In response, the licensee obtained and documented in Nuclear Design Information Transmittal (NDIT) S040-DH-0513 the vendor's confirmation that a 42 inch distance was acceptable for flows as high as 7,200 gpm. The team determined that the specific vendor statement took precedence over the general recommendation in the design input reference. Therefore, the calculation's conclusion that CCSW pump intake bay dimensions were adequate was correct. The DEAG reviewer stated the reason he did not comment on the absence of a specified maximum flow rate was that it was common knowledge within Dresden Engineering that the 20 psid restriction required throttling the CCSW flow.

Document ID # 5543459:

The 12 System Key Parameter Verification Program (LPCI System Discrepancy #4) identified that no formal hydraulic calculation existed which demonstrated that the LPCI system could provide the required 5,000 gpm flow through the containment cooling heat exchanger to ensure adequate containment cooling.

This evaluation documented that the LPCI system could provide the required flow. The capability was demonstrated primarily by Dresden Operating Surveillance (DOS) 1500-10, "LPCI System Pump Operability Test with Torus Available and Inservice Testing (IST) Program," Revision 30 and NFS-BSA-D-97-03, "Sensitivity Analysis Post-LOCA Containment Performance for Dresden Units 2/3," dated March 12, 1997. The team determined that the evaluation was technically sound.

c. Conclusions

The team concluded that the calculations and evaluation were technically sound. However, the documentation of logic employed and the common site specific knowledge used was not always evident and could have been improved with more attention to detail.

E3.4 Updated Final Safety Analysis Report

a. Inspection Scope (IP 37550)

The team reviewed sections of the UFSAR and the licensee's corrective action documentation associated with a potential Dresden Lock and Dam failure.

b. Observations and Findings

The team expressed a number of concerns with regards to the validity of some UFSAR statements contained within Section 9.2.5.3.1, "Dam Failure during Normal Operations," and Section 9.2.5.3.2, "Dam Failure Coincident with a LOCA."

The team observed that the UFSAR did not accurately characterize the plant's design-basis or the plant's capability to respond to a potential Dresden Lock and Dam failure. As a result, the team had concerns with the ability of the plant to respond to a dam failure as stated in the UFSAR.

The team's review of the licensee's "Summary of Dresden NRC Requirements for 1997," dated September 30, 1997, indicated that the licensee was aware of similar concerns, although not identical to the team's. The licensee stated that several PIFs related to this issue were in the corrective action process. The PIFs identified were:

- PIF 227A-12-1997-012788, "UFSAR Implied One CCSW Pump Operation After a Dam Failure Coincident With a LOCA," dated February 25, 1997

- PIF D1997-05554, "UFSAR CCSW Piping Statement Discrepancy" dated June 25, 1997

- PIF D1997-05955, "UFSAR LPCI Flow Timing Discrepancy," dated June 24, 1997

- PIF D1997-06487, "Incorrect Source Document Referenced for Diesel Generator Cooling Water Pump in a Calculation," dated August 27, 1997

- PIF D1997-08290, "NRC Concerns About CCSW System Performance After a Dam Failure Coincident With a LOCA," dated November 25, 1997

  This PIF was issued as a result of the team's concern that no high-point vent valves were installed to vent trapped air during the reflood of the CCSW intake bay, which was not considered by DOA-0010-01, "Dresden Lock and Dam Failure," Revision 6.

In addition, the licensee stated that an evaluation had not been completed to determine whether the Dresden Nuclear Plant Design Basis required the plant to be capable of a safe shutdown after a dam failure coincident with a Unit 2 or 3 LOCA and a loss of offsite power (LOOP)..

c. Conclusions

The team had concerns that the UFSAR did not accurately characterize the plant's design-basis or the plant's capability to respond to a potential Dresden Lock and Dam failure. As a result, the team concluded that further review by the licensee and NRC was required. An NRC URI was initiated to document these concerns. (URI 50-237/249-97021-01(DRS))

E4 Engineering Staff Knowledge and Performance

a. Inspection Scope (IP 37550)

The team observed the performance of the engineering staff, interviewed both system and design engineering personnel, and walked down plant systems with some system engineers.

b. Observations and Findings

All engineers interviewed appeared to be experienced and well qualified. However, the turnover rate for some system engineers appeared to be high. The system engineer for

DC systems was only on the job for about six months. The system engineers for several other systems were only on the job for about six months to 1½ years. However, the team did not identify any specific problems directly linked to the lack of experience on the part of the system engineers.

The team noted that the system engineers interviewed maintained good system notebooks. The system engineers were required to walk down their systems periodically. The team walked down selected plant systems with the system engineers, and considered them knowledgeable on their assigned systems.

The team observed a surveillance test on the Unit 2 125 Volt alternate battery. The test was modified performance test per procedure Dresden Engineering Surveillance (DES) 8300-52. As the DC system engineer at Dresden was relatively new to this test, it was performed under the supervision of a system engineer from Braidwood. The battery testing was done smoothly and no major problems were observed. The team noted good communications with operations and maintenance during these tests.

c. Conclusion

The team concluded that the system engineering department was adequately staffed. The team determined that the engineers interviewed were qualified and experienced in the areas assigned. Good inter-departmental communications were noted between system engineering, operations and maintenance during the special test observed.

E6 Engineering Organization and Administration

a. Inspection Scope (IP 37550; IP 92703)

The team evaluated the performance and effectiveness of the DEAG to determine if the CAL commitments and corrective actions were completed and had satisfied NRC requirements.

b. Observations and Findings

On November 21, 1996, CAL No. RIII-96-016, was issued by the NRC as a result of significant concerns with the station's control of calculations and with the overall performance of site and corporate engineering activities. The CAL identified various planned corrective actions to improve the performance of the engineering organization. One of the planned activities was the formation of an engineering assurance group or DEAG that was composed of senior ComEd engineering personnel and experienced outside experts. The function of the group was to provide oversight of key engineering activities until normal engineering functions had improved to the point where the reviews were no longer necessary.

In NRC Inspection Report 50-237/249-97008(DRS), the NRC evaluated the CAL activities and determined that the CAL commitments and corrective actions were completed, except for those activities associated with the DEAG. The inspection identified that initial DEAG implementation was not effective as an oversight organization. As a result, the CAL remained open until effective DEAG performance was demonstrated.

The team reviewed most of the DEAG review sheets for the period between June 1997 and October 1997, and determined that the DEAG reviews had in most cases, documented relevant significant problems and appropriately required those documents to be corrected. As a result, the DEAG reviews have improved the quality of the engineering products. The DEAG reviews provided good recommendations for improvements in methodology, technical content, and clarification and documentation improvements that would make the engineering products a better source of information for future users.

Since June 1997, the DEAG provided monthly reports to engineering management that summarized the scope of the DEAG activities and the results of the DEAG reviews. The DEAG observations were consistent through November 1997, in identifying areas that needed improvement. The improvement areas were identified as follows:

- Understanding of Regulatory or Design-Basis Requirements on Work Performed
- Attention to Detail
- Interdiscipline Reviews

The team observed that the DEAG reviews were generally thorough and technically sound and produced similar observations with other licensee self-assessment efforts, as described in Section E7. The DEAG efforts showed that the quality of the engineering documentation has improved. However, the team was concerned that many of the DEAG members, who were engineering contractors, were no longer employed at the Dresden site and such loss of experienced personnel might degrade the licensee's ability to maintain an improving trend in engineering performance. Full staffing of qualified personnel in the DEAG was a continuing problem.

c. <u>Conclusions</u>

The team concluded that all commitments and corrective actions identified by Confirmatory Action Letter (CAL) No. RIII-96-016, dated November 21, 1996, including those activities associated with the Dresden Engineering Assurance Group (DEAG) have satisfied NRC requirements. The CAL was closed.

E7 Quality Assurance in Engineering Activities

a. <u>Inspection Scope (IP37550; IP40500)</u>

The team reviewed the following self-assessment documents to assess quality and proposed corrective actions:

- Report Number 237-230-97-00300, "Common Cause Analysis and Investigation of an Adverse Trend in Human Performance Error-Related Licensee Event Report (LER) Rate for the First Two Quarters of 1997 Which Resulted in Exceeding the Dresden 50.54(f) Performance Criterion Action Level, Caused by Failure to Make Timely Change and Inadequate Work Practices," Revision 0

- Report Number 237-251-97-05000, "Plant Engineering Work Management and Support Responsiveness," dated November 18, 1997

- DOC ID# 5549414, "Assessment of Engineering Department Safety Evaluation," Revision 0

b. <u>Observation and Findings</u>

The team's review of the documents identified above indicated that the licensee had taken a pro-active position in an attempt to disclose the performance problems within the organization. Many of the weaknesses identified described similar problems previously identified by the NRC, but the make-up and the openness of the licensee's conclusions indicated a positive trend. For example, the LER common cause analysis investigation identified that the most prevalent problems were associated with personnel acceptance of insufficient time to perform consistent quality technical reviews due to shortcuts taken and inaccurate assumptions made during validation and verification activities. The licensee stated that the same type of errors were occurring station wide and in a variety of processes. In addition, as discussed in Section E6, the DEAG consistently identified that the problems associated with engineering rework were predominately due to inattention to detail as a result of not taking the time to perform an adequate detailed review.

The team observed that the self-assessment documents identified above were focused, provided detailed and relevant observations, and provided a quality product. The self-assessment corrective action recommendations were appropriate for the identified weaknesses. For example, the insufficient time pressure problem was addressed by the LER common cause analysis investigation by the implementation of an Engineering Rapid Response Team (ERRT) to remove short duration emergent work activities from the system engineer's responsibility. In addition, an engineering reporting system (ERS) was developed and implemented to provide a workload scheduling and tracking tool to assist engineering personnel in managing workload.

The team observed that the proposed self-assessment corrective actions have not been fully effective for all proposed recommendations. For example, the ERRT was effective in reducing some of the reactive workload; however, the ERS was too complex and not user friendly to effectively prioritize and manage the engineers workload. The DEAG, as discussed in Section E6, provided quality reviews that contributed to the overall effectiveness of the licensee's self-assessments activities.

c. Conclusions

The team concluded that the licensee's self-assessment activities were pro-active and for the most part effective.

IV. Plant Support Areas

F2 Status of Fire Protection Facilities and Equipment

a. Inspection Scope (IP40500; IP92904)

The team reviewed the licensee's corrective actions concerning problems associated with the control room's HVAC system automatic smoke purge mode.

b. Observations and Findings

During testing of the control room's HVAC system exhaust ducts in November 1994, the licensee discovered that a prior inadvertent change to the control room's HVAC system deleted the automatic smoke purge mode transfer capability as described in UFSAR, Section 6.4.4.3. A URI 50-237/249-96002-07 was generated to track the concern and is discussed further in Section F8.2.

The UFSAR stated that the control room's HVAC system was designed to isolate and maintain design conditions within the control room during fires. In the event of smoke in the control room, the smoke purge mode would allow 100% outside air intake with no recirculation of exhaust air into the control room HVAC zone (envelope). The UFSAR further stated that smoke detectors automatically switched the control room's HVAC system (Train A) to the smoke purge mode.

The licensee concluded that the problem occurred as a result of control room modifications M12-2/3-82-1, M12-0-87-005, and M12-0-86-006. The smoke detectors were inadvertently isolated as a result of modifications to the control room's envelope, which deleted the automatic smoke purge mode transfer capability. As a result, control room operators were required to take manual action to initiate the HVAC smoke purge mode. A safety evaluation to ascertain whether the problem was an unreviewed safety question was not initially performed by the licensee. Following NRC concerns, the licensee performed a safety evaluation prior to startup from the 1996 Unit 2 refuel outage. The licensee concluded that an unreviewed safety question did not exist.

A recent modification, M12-0-96-001, "Control Room HVAC Fire Protection System Modification" corrected the deleted automatic smoke purge mode transfer capability by installing smoke detectors in the remaining ventilation system. However, the team identified that the description of the system's automatic initiation capability had been removed from the UFSAR. Removal of the UFSAR's reference to the control room's automatic transfer to the smoke purge mode was made during the performance of the safety evaluation made in March 1996, just prior to the Unit 2 startup. The UFSAR change was made to accommodate the inadvertent change to the control room HVAC system by only referencing the manual mode. The licensee stated that as a result of two engineers not communicating, one engineer had taken the description for the automatic initiation of the smoke purge mode out of the UFSAR.

The safety evaluation performed in June 1996, for Modification M12-0-96-001, neglected to identify that a change to the UFSAR was required. As a result, during this inspection, the licensee issued PIF# D1997-08239 to correct the affected UFSAR sections concerning the control room HVAC system's automatic initiation.

c. <u>Conclusions</u>

The failure to perform a safety evaluation from November 1994 until March 1996, until identified by the NRC, was a violation of 10 CFR 50.59.
(VIO 50-237/249-97021-02(DRS))

F3 Fire Protection Procedures and Documentation

a. <u>Inspection Scope (IP40500; IP92904)</u>

The team reviewed the licensee's corrective actions concerning problems associated with the Fire Protection Report (FPR).

b. <u>Observations and Findings</u>

The NRC previously identified that polyvinyl chloride (PVC) drain piping was installed during a 1986 control rod drive modification and that the licensee had not performed a safety evaluation nor added the increased combustible fire loading to the FPR's Fire Hazards Analysis (FHA). In addition, the NRC also identified that the construction of a turbine deck concrete building, which was another combustible fire load, had not been added to the FHA. The licensee committed to perform a safety evaluation, investigate/identify other unevaluated plant PVC usage, and specifically evaluate PVC usage during the modification process and to include the identified combustible fire loads in the next update to the FHA. A URI 50-237/249-96002-09(DRS) was generated to track the concern and is discussed further in Section F8.3.

Branch Technical Position Auxiliary Power Conversion System Branch (BTP APCSB) 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants," dated May 1976, was an FPR requirement, which required the minimization of PVC usage in the plant. The team determined that the safety evaluation completed as part of the licensee's corrective action was acceptable. During the licensee's investigation, additional in-plant PVC usage was identified. In addition, the licensee had changed the modification process to ensure that PVC usage was minimized in the plant.

The team observed, however, that the combustible fire load items were never added to the FHA, which included the PVC usage and turbine deck concrete building previously identified. The reason that the combustible fire load items had not been incorporated into the FHA was that the FPR had not been updated since 1994. The FHA is part of the FPR and the FPR was considered part of the UFSAR.

Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," dated April 24, 1986, stated that fire protection plans and programs shall be incorporated as part of the UFSAR and therefore, would be updated and submitted to the NRC in accordance with the requirements of 10 CFR 50.71(e). GL 86-10 also stated, "All changes to the approved program shall be reported annually to the Director of the Office of Nuclear Reactor Regulation, along with the UFSAR revisions required by 10 CFR 50.71(e)." The failure to submit revised portions of the FPR to the NRC was a violation of 10 CFR 50.71(e). (VIO 50-237/249-97021-03(DRS))

The team also observed a weakness within the licensee's corrective action process concerning these earlier identified FPR problems. Following NRC Inspection Report 96002 (February 14, 1996, through March 29, 1996), Quality and Safety Assessment (Q&SA) wrote Corrective Action Record (CAR) 12-96-151 "Fixed Combustible Loading." The CAR identified that, contrary to the requirements of GL 86-10 and Engineering Procedure ENC-QE-85, "Control and Revision of the Fire Protection Program Documentation," updates to the FHA Report, which was part of the FPR, had not been submitted to the NRC. A PIF and NTS item were generated on December 12, 1996, 10 months after the identification of the earlier FPR problems. NTS history indicated that completion of the FPR update was extended from June 30, 1997, to September 1, 1997, and then to December 18, 1998. In addition, on November 7, 1997, Q&SA identified that there was no process to receive, evaluate, track, and update FPR information.

On November 19, 1997, the licensee opened NTS Item #237-225-97R12-97242 to track the development of a procedure to control updating of the FPR and provide interim tracking of FHA changes. A due date of September 4, 1998, was assigned to the NTS item. Currently, the FPR does not represent plant conditions. The identification and corrective actions for FPR problems were not timely.

The team further identified that fire risks associated with the additional combustible fire loading had not been incorporated into the fire pre-plans. Technical Specification (TS) 6.2.A stated that written procedures shall be established and implemented covering these activities. Dresden Fire Protection Procedure (DFPP) 4100-01, Revision 1, "Fire Protection Program," required that Fire Pre-Plans be updated annually. The Dresden "Fire Pre-Plans," Revision 2, had not been updated since September 1992. The licensee's failure to comply with these requirements was a violation of TS 6.2.A. (VIO 50-237/249-97021-04(DRS))

c. <u>Conclusions</u>

Failure to update and submit the revised portions of the FPR to the NRC was a violation of 10 CFR 50.71(e). (VIO 50-237/249-97021-03(DRS)) Failure to update the fire pre-plans was a violation of TS 6.2.A. (VIO 50-237/249/97021-04(DRS))

F8 Miscellaneous Fire Protection Issues (IP92904)

F8.1 (Closed) VIO 50-237/249-96002-05B(DRS): This violation was issued for not performing a full 8 hour discharge test on 47 Appendix R emergency lighting units as required by DES 4153-04, "Emergency Lighting Discharge Test," Revision 0. The licensee changed the procedure/surveillance to ensure that the batteries were discharge tested for the full 8 hours. The team reviewed two years of surveillance data and determined that the licensee's corrective actions were effective. This item was closed.

F8.2 (Closed) URI 50-237/249-96002-07(DRP): This unresolved item was issued for inadvertently deleting the control room HVAC system automatic smoke purge mode transfer capability as described in UFSAR, Section 6.4.4.3. The change to the automatic smoke purge mode had been made as a result of a control room modification. A recent modification corrected the control room's HVAC system automatic smoke purge mode problem. However, a violation was issued for not performing a safety evaluation as discussed in Section F2. This item was closed.

F8.3 (Closed) URI 50-237/249-96002-09(DRS): This unresolved item was issued for using PVC during a modification without performing a safety evaluation. The licensee completed the safety evaluation and concluded there was no unreviewed safety question. During the team's review, the FPR was observed as not having been updated for PVC usage and the addition of a turbine deck concrete building. As a result, a violation was issued for not having updated the FPR since 1994 as discussed in Section F3. This item was closed.

V. Management Meetings

X1 Exit Meeting Summary

The team presented the final inspection results to members of licensee management at the conclusion of the inspection on January 27, 1998. The team initially met with the licensee's representatives to summarize the scope and findings of the on-site inspection activities on November 26, 1997. During both of these meetings, the team questioned licensee personnel as to the potential for proprietary information being included or retained in the inspection report material as discussed at the exits. No proprietary information was identified as included or retained.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Abrell, NRC Coordinator, Regulatory Assurance
D. Ambler, Regulatory Assurance Supervisor (Acting), Regulatory Assurance
H. Anagnostopoulos, Corrective Action Process (CAP) Supervisor, Quality & Safety Assessment
R. Book, CAP Staff, Quality & Safety Assessment
A. Casillo, Mechanical Lead (M1), Design Engineering
W. Clover, Design Engineer, Design Engineering
J. Dawn, DEAG Supervisor, Plant/Engineering Programs
F. Fink, Business Manager, Dresden
M. Friedmann, HP Technical Lead, Health Physics
R. Freeman, Site Engineering Manager, Dresden
W.Halcott, Auxiliary System Lead, Systems Engineering
M.Heffley, Site Vice President, Dresden
K. Housh, ISEG Engineer, Quality & Safety Assessment
L. Jordan, Training Manager (Acting), Training
A. Khanna, Design Lead, Design Engineering
J. Kish, CCSW System Engineer, Systems Engineering
W. Lipscomb, Assessor, Site Vice President Staff
R. Mahendranathan, Mechanical Engineer, Design Engineering
T. McGowan, DC System Engineer, Electrical System & Components
E. Netzel, Director, Supplier Evaluation Services/Nuclear Oversight
K. Peterman, Supervisor, Configuration & Administration Management; DEAG Member
P. Planing, Superintendent, Systems Engineering
P. Racicot, AC System Engineer, Electrical System & Components
C. Richards, Audit Supervisor, Quality & Safety Assessment
E. Salinas, System Engineer, Systems Engineering
B. Shete, Mechanical Engineer, Design Engineering
F. Spangenberg, Regulatory Assurance Manager, Dresden
D. Spencer, Electrical System & Components Lead, Systems Engineering
S. Tutich, Electrical Lead, Design Engineering
L. Weir, Superintendent, Design Engineering
D. Winchester, Manager, Quality & Safety Assessment

ComEd Contractors

H. Campbell, Member, DEAG (Titan)
C. Kinstler, Engineer (Sargent & Lundy)
H. McCullough, Site Lead (Acting), Design Basis Initiative (Sargent & Lundy)

LIST OF INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 92703:	Followup of Confirmatory Action Letters
IP 92904:	Followup - Plant Support

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

50-237/249-97021-01(DRS)	URI	UFSAR Dam Failure Discrepancies
50-237/249-97021-02(DRS)	VIO	Failure to Perform 50.59 Evaluation
50-237/249-97021-03(DRS)	VIO	Failure to Update FPR and Submit to NRC
50-237/249-97021-04(DRS)	VIO	Failure to Update Fire Pre-plans

<u>Closed</u>

50-237/249-96002-05B(DRS)	VIO	Failure to Adequately Test Emergency Lighting
50-237/249/96002-07(DRP)	URI	Untimely Resolution of Operability Evaluations
50-237/249-96002-09(DRS)	URI	Polyvinyl Chloride (PCV) Usage Not Well Controlled

LIST OF ACRONYMS

ATTN	Attention
BWR	Boiling Water Reactor
CAL	Confirmatory Action Letter
CAR	Corrective Action Record
CCSW	Containment Cooling Service Water
CFR	Code of Federal Regulations
ComEd	Commonwealth Edison
DAP	Dresden Administrative Procedure
DEAG	Dresden Engineering Assurance Group
DES	Dresden Engineering Surveillance
DFPP	Dresden Fire Protection Procedure
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DTI	Desk Top Instruction
E&TS	Engineering and Technical Support
GL	Generic Letter
HVAC	Heating, Ventilation, and Air Conditioning
ISEG	Independent Safety Engineering Group
JSPLTR	ComEd (J.S. Perry) Letter
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPM	Licensing Project Manager
MSL	Mean Sea Level
NEP	Nuclear Engineering Procedure
NOC-BOD	Nuclear Operating Committee-Board of Directors
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NTS	Nuclear Tracking System
PDR	Public Document Room
PIF	Problem Identification Form
PVC	Polyvinyl Chloride
Q&SA	Quality and Safety Assessment
RBCCW	Reactor Building Closed Cooling Water
RG	Regulatory Guide
SRI	Senior Resident Inspector
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USQ	Unreviewed Safety Question
VIO	Violation

PARTIAL LIST OF DOCUMENTS REVIEWED

DOCUMENT NUMBER	DOCUMENT DESCRIPTION	REVISION OR DATE ISSUED
CAL No. RIII-96-016	Confirmatory Action Letter	November 21, 1996
CAR 12-96-151	Fixed Combustible Loading	December 23, 1996
CAR 12-97-105	Fire Protection Report	November 7, 1997
DAP 02-27	The Integrated Reporting Process (IRP)	Revision 7
DAP 21-03	Processing Plant Design Changes	Revision 13
DEAG Review Sht 8.10	Removal of Description of Acid & Caustic Equipment from UFSAR	August 21, 1997
DEAG Review Sht 8.11	Troubleshooting of a Stator Leak	August 22, 1997
DEAG Review Sht 8.16	Clarification of Information on an Overhead Crane	August 22, 1997
DEAG Review Sht 8.17	Clarification of Spent Fuel Pool Liner Thickness	August 22, 1997
DEAG Review Sht 8.28	Security Position Title Change in the UFSAR	August 28, 1997
DES 4153-04	Emergency Lighting Discharge Test	Revision 0
DFPP 4100-01	Fire Protection Program	Revision 1
-------	Dresden Engineering Assurance Group Activities for May, 1997 (1st DEAG Monthly Report)	June 26, 1997
DOC ID # 0005458065	Dresden Engineering Assurance Group Activities for June, 1997	July 11, 1997
DOC ID # 0005491140	Dresden Engineering Assurance Group Activities for July, 1997	August 18, 1997
DOC ID # 0005503264	Dresden Engineering Assurance Group Activities for August, 1997	September 8, 1997
DOC ID # 0005558157	Dresden Engineering Assurance Group Activities for October, 1997	November 17, 1997

DRE 97-0171	Calculation for Determination of Acceptance Criteria for CCSW One and Two Pump NPSH Testing - Units 2 & 3	Revision 0
DRE 97-0172	Calculation to Determine Submergence for Excessive CCSW Intake Vortexing Prevention.	Revision 0
DTI-DE-15	Roles and Responsibilities of the Dresden Engineering Assurance Group	Revisions 0, 1, 2
ENC-QE-85	Control and Revision of the Fire Protection Program Documentation	
Eval Doc ID #5543459	CAL Action Item Update Report Following First Monthly Status Meeting Held December 19, 1996	December 30, 1996
GL 86-10	Implementation of Fire Protection Requirements	April 24, 1986
JSPLTR: 97-0005	ComEd Interim Response to NRC Independent Safety Inspection Report	January 13, 1997
JSPLTR: 97-0041	ComEd Response to NRC Independent Safety Inspection Report	February 26, 1997
JSPLTR: 97-0043	Verification Screening of Key Parameters for Twelve Risk Significant Systems	Revision 0
M12-0-96-001	Control Room HVAC Fire Protection System Modification	
NEP-04-01DR	Dresden Plant Modification Site Appendix	Revision 2
NEP 10-03	Disposition of Design Basis Discrepancies	Revision 0
NEP 12-01	Preparation, Review, and Approval of Design Input Requirements	Revision 2
NEP 12-02	Preparation, Review, and Approval of Calculations	Revision 4
NSWP-A-15	ComEd Nuclear Division Integrated Reporting Program	Revision 0 & 1
OP EVAL 97-81	Minimum Water Level in CCSW Intake Bay	July 8, 1997
PIF # D1997-05554	UFSAR CCSW Piping Statement Discrepancy	June 25, 1997

PIF # D1997-05556	UFSAR Safety Grade Cold Shutdown Capability Discrepancy	June 25, 1997
PIF # D1997-05955	UFSAR LPCI Flow Timing Discrepancy	June 24, 1997
PIF # D1997-06487	Incorrect Source Document Referenced for Diesel Generator Cooling Water Pump in a Calculation	August 27, 1997
PIF # D1997-08239	UFSAR Deletion/Addition	November 21, 1997
PIF # D1997-08290	NRC Concerns About CCSW System Performance After a Dam Failure Coincident With a LOCA	November 25, 1997
PIF # 227A-12-1997-012788	UFSAR Implied One CCSW Pump Operation After a Dam Failure Coincident With a LOCA	February 25, 1997
Report Base NTS Number: 237-251-97-05000	Plant/Programs Engineering Sel Assessment 3-7 Nov 97	November 18, 1997
_ _ _ _ _ _ _	Fire Protection Report (FPR)	Amendment 10 December 1994