U. S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-254/96002(DRP); 50-265/96002(DRP)

FACILITY
Quad Cities Nuclear Power Station, Units 1 and 2

License Nos. DRP-29; DPR-30

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

DATES
January 19 through March 4, 1996

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4/17/96
Date

AREAS INSPECTED

The inspectors performed a routine, unannounced inspection of operations, engineering, maintenance, and plant support while routinely evaluating safety assessment and quality verification activities. Inspectors performed follow-up inspection for non-routine events and for certain previously identified items.

Executive Summary

Operations

Problems in personnel performance and oversight resulted in breaching primary containment at power, improperly changing a procedure, and decreasing the availability of safety equipment. The inspectors noted strengths in panel monitoring and control room communications. Licensing basis issues were identified regarding spent fuel pool cooling and intake bay level assumptions.

- Operators vented primary containment with the reactor at about 15 percent power. Problems included voluntary entry into TS 3.0.A. and an improper procedure change. Notices of Violation were issued for an improper procedure change and for breaching primary containment integrity (Section 1.2.).
- Problems maintaining proper operation of the trash rake concurrent with debris build up on the traveling screens resulted in a power reduction on both units (Section 1.3.).
- Poor log review resulted in decreased control building chiller availability (Section 1.4.).
- Panel monitoring and communications were good, but the quality of control room turnover and housekeeping declined (Section 1.5. and 1.6.).
- Previous refueling outages with full core offloads had not been controlled to ensure all licensing basis requirements were met. ComEd addressed spent fuel pool cooling concerns relating to a full core off load in order to meet licensing basis requirements (Section 1.7.).

Maintenance and Surveillance

Personnel errors caused several minor events involving safety equipment. These errors were indicative of inattention to detail, and a lack of control and oversight of the maintenance activities.

- Failure to remove a jumper after VOTES testing which resulted in repeated valve cycling for approximately eight minutes was considered a Non-Cited Violation (Section 2.1.).
- Human errors included improper out of service verification and a wrong component mistake (Section 2.2.).
- A procedure problem and insufficient review led to a standby diesel generator (SBDG) automatic start (Section 2.3.).

Engineering and Technical Support

The inspectors noted improvements in some recent root cause evaluations and in follow-up of industry experience. However, initial root cause evaluation for the Unit 2 SBDG events was poor. Inservice testing program weaknesses included testing criteria for Residual Heat Removal Service Water (RHRSW) pumps not verified by calculations, Unit 2 SBDG fuel transfer pump check valve missed surveillance, and Unit 1 standby liquid control check valves surveillance failures (Section 3.0.). Numerous materiel condition deficiencies continued to affect facility operation, challenge operators and cause increased unavailability times of safety related equipment.

- Inspectors observed recently improved root cause evaluations but noted that some as-found information was not captured for troubleshooting (Section 3.1.).
- Numerous material condition issues challenged equipment and unit performance (Section 3.4.).
- Inservice testing (IST) weaknesses were identified (Section 3.5.).
- Several problems challenged control room ventilation operability (Section 3.6.).
- Weaknesses in the followup of the 1995 Unit 2 SBDG failures to start included poor root cause evaluation and poor materiel storage controls. One Violation was issued for materiel storage problems (Section 3.7.).
- The Unit 1 High Pressure Coolant Injection (HPCI) system was declared inoperable due to unexpected system anomalies that occurred during testing. The anomalies were indicative of preventative and corrective maintenance which have not yet effected sustained reliability. (Section 3.8.)
- Structural steel beams and connections for Units 1 and 2 residual heat removal corner rooms were determined as not meeting Updated Final Safety Analysis Report (UFSAR) allowable stress limits (Section 3.9).
- Local leak rate testing procedure violations were identified (Section 3.10.).

Plant Support

A chemistry-related personnel error resulted in an excessive sodium bisulfite discharge. Outage ALARA planning efforts were good and several actions were planned to reduce radiological source term. Failure to plan and monitor for increased reactor coolant activity resulted in increased refuel floor exposure.

- A technician failure to follow procedure resulted in a sodium bisulfite discharge in excess of discharge limits. A Non-Cited Violation was issued (Section 4.1.).
- Early isolation of the reactor water cleanup system resulted in higher than expected radiation levels on the refuel floor. This problem resulted in higher personnel exposure and a delay in critical path activities (Section 4.2.)

Safety Assessment/Quality Assurance

- The inspectors identified some weaknesses in Plant Operations Review Committee practices (Section 5.1.).

Summary of Open Items

Violations: identified in Sections 1.2. and 3.7.

Unresolved Items: identified in Sections 1.9., 3.5., and 3.9.

Inspector Follow-up Items (IFI): identified in Sections 1.3., 1.4., 1.7 and 3.6.

Non-Cited Violations: identified in Sections 2.1., 3.10., and 4.1.

INSPECTION DETAILS

1.0 OPERATIONS:

The inspectors used NRC Inspection Procedures 71707 and 93702 to evaluate plant operations. Strengths in panel monitoring and control room communications were evident. Weaknesses in operator understanding of technical specification and reporting requirements were seen.

1.1 Operating Summary and Follow-up of Events (93702)

Unit 2 operated at or about full power, during the period with power reductions for surveillances, and inoperable fire protection pumps. At the end of the period, the Unit 2 generator was removed from service due to problems with turbine control valve No. 2. Unit 1 was operating at or about maximum available power through most of the period, in coastdown to refueling. Power reductions were necessary for inoperable fire pumps and a feedwater heater relief valve failure. On February 10, Unit 1 was shutdown in preparation for a 78-day scheduled refuel outage, Q1R14. Major outage activities included: inspection of the reactor vessel interior welds, installation of a core shroud support modification, reactor water cleanup pipe replacement, main turbine disassembly and inspection, upgrade to feedwater regulating valves, and a scram discharge volume level switch modification. Unit 1 startup was scheduled for the end of April 1996.

During this inspection period several events occurred, some of which required a prompt notification of the NRC pursuant to 10 CFR 50.72. The following events were reviewed for reporting timeliness and immediate response. Root cause investigation and corrective actions will be reviewed in this or follow-up inspections.

January 23	Emergency Notification System (ENS) call. A trash rake failure concurrent with debris built up on the intake structure resulted in both units reducing power due to low water level downstream of the traveling screens.
January 24	ENS call. A quantity of sodium bisulfite greater than that allowed by the discharge permit was discharged.
January 30	ENS call. Operators declared Unit 1 HPCI inoperable due to auxiliary oil pump cycling.
February 3	ENS call. The "B" train of control room ventilation was declared inoperable due to outside air temperature dropping below -28 degrees F.
February 6	A feedwater heater relief valve opened and overflowed a downstream funnel. Operators reduced Unit 1 power to 60 percent to bypass a string of feedwater heaters.
February 10	Unit 1 was shut down for refuel outage Q1R14.
February 21	ENS call. ComEd identified that a control room emergency filtration unit would not start or run since the toxic gas analyzer would become inoperable during a loss of power to Unit 1.

February 22	ENS call. During testing, Unit 1 SBDG unexpectedly started after operators reset overspeed trip.
February 25	ENS call. The "B" control room emergency filtration system was declared inoperable after identifying low freon pressure in compressor.
March 3	ComEd identified the shared SBDG was inoperable to Unit 2 due to an out of service error.
March 4	ENS call. "B" control room ventilation inoperable due to both inlet and outlet dampers failing to close.
March 4	Unit 2 Turbine Control Valve No. 2. malfunction necessitated taking Unit 2 off-line.

1.2 Breaching Primary Containment With Reactor Critical

On February 10, 1996, with the Unit 1 reactor operating at about 15 percent power, operators opened a vent path from primary containment to reactor building atmosphere. Problem Identification Form (PIF) 96-0407 indicated that TS 3.0.A. had been entered and possibly required NRC notification. The inspector review identified several problems with meeting the requirements of Technical Specification 3.7.A.2., which required primary containment integrity be maintained at all times when the reactor is critical or above 212 degrees F. Problems included not meeting TS 3.7.A.2., poor operator understanding of TS 3.0.A., and using the temporary procedure change process inappropriately.

While the licensee was shutting down Unit 1 for a refuel outage, engineers performed a local leak rate test (LLRT) on the RHR test return line to the torus. Engineers used procedure QCTS 600-18 "RHRS Suppression Chamber Spray Local Leak Rate Test (MO-1(2)-1001-34A/B, 36A/B, and 37A/B)." Operators used a temporary procedure field change process to allow the procedure to be used in an operating condition rather than in shutdown or refueling conditions as was the original intent of the procedure. Use of a temporary procedure change process which changed the original intent of the procedure without the proper pre-implementation review was a **Violation (50-254/265-96002-01)** of TS 6.2.D.1.

The test required opening the 1001-36 & 37 valves with instrument line root valve 1001-151 open. This action allowed the torus to communicate with the reactor building basement atmosphere for the short times the root valve was open. Operators were knowledgeable of TS 3.7.A.2. and the requirements of primary containment integrity. However, Operations management believed that voluntary entry into TS 3.0.A. would be acceptable since the reactor shutdown was already in progress. Unit 1 T.S. 3.0.A described required actions in the event a Limiting Condition for Operation could not be satisfied.

Nuclear Regulatory Commission guidance in Generic Letter 87-09 "Section 3.0. and 4.0. of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements" indicated that T.S. 3.0.A (generically called 3.0.3) was not intended to be used as an operational convenience which permits voluntary removal of components from service in lieu of other alternatives that would not result in components being inoperable. The inspectors ascertained that this testing was performed to more rapidly complete refueling outage work. The operator's log revealed 4 separate entries into TS 3.0.A. The longest entry into 3.0.A. was 3 minutes. During performance of the "A" loop test, the reactor was critical. For both "A" and "B" loop tests, the reactor temperature was above 212 degrees F. Opening of a primary containment boundary with the reactor above 212 degrees F constituted a **Violation (50-254/265-96002-02)** of Technical Specification 3.7.2.A.

In addition, the inspectors also noted poor understanding of reporting requirements for entry into TS 3.0.A., even though these requirements were spelled out in the ComEd reportability manual, 10 CFR 50.72, and 10 CFR 50.73.

The 1001-36 was the RHR test return line to the torus. The 1001-37 valve supplied the torus spray header. Per the UFSAR Table 6.2-7, neither valve would be actuated by the primary containment isolation system (PCIS). However, both the 1001-36 and 37 valves were interlocked closed during a Loss of Coolant Accident (LOCA) signal. The 1001-151 valve was a 3/4" manually operated instrument line root valve, where a test tap capped connection was replaced with an LLRT rig. During the test, an operator with a radio at the 151 valve was able to communicate with the control room.

The inspectors noted that the safety significance of breaching primary containment in this situation was low. However, poor operator understanding of use of TS 3.0.A. and the improper procedure change were considered important deficiencies.

1.3 Failure of Trash Rake Resulted in Dual Unit Entry Into 24 Hour LCO

Failure of the intake structure trash rake concurrent with buildup of debris on the intake structure resulted in low water level inside the intake structure. This resulted in both fire pumps becoming inoperable. Operators reduced power in both units in accordance with an administrative limiting condition for operation (LCO). The issue revealed the following weaknesses:

- Operations did not have a requirement nor method for determining water levels inside the intake structure.
- Preventive and corrective maintenance activities for the trash rake system were inadequate.
- Operations was not prepared to respond in a timely manner to reduced water levels inside the intake structure,
- In-service testing (IST) water level requirements used to determine net positive suction head (NPSH) for safety-related pumps were not supported by calculations.

With the intake structure trash rake broken, concurrent with a large accumulation of debris at the intake structure, water levels inside the intake structure decreased. Operations requested engineering to determine at what water levels pumps would become unavailable. Engineering required water level to be at least 568 feet 6 inches inside the intake structure to maintain fire diesel operability. Water levels inside the intake structure reached a minimum of about 568 feet above sea level. On January 22, operations entered into a 24-hour administrative LCO to shut down both units due to both diesel fire pumps being inoperable.

On January 23, ComEd started to reduce power on both units. Workers repaired the trash rake and cleaned the racks on the intake structure. Water level inside the intake structure increased slowly due to the cleaning efforts. With both units at about 70 percent power, operations exited the LCO with water level inside the intake structure at 571 feet 2 inches and rising. The lowest water level reached inside the intake structure was 568 feet 1 inch.

Engineering calculations concluded that a river level of 561 feet provided adequate NPSH to allow operation of other safety related pumps drawing from the intake structure, including residual heat removal service water (RHRSW) and diesel generator cooling water (DGCW) pumps. This water level was within the limits specified in Section 2.4.4. of the updated final safety analysis report (UFSAR).

The UFSAR indicated river level at the station would be maintained not lower than 561 feet if Lock and Dam No. 14 on the Mississippi River downstream of the station were to fail. However, engineering determined that for IST purposes, safety-related cooling water would become inoperable with water level inside the screens of 566.4 feet. The inspectors questioned the discrepancy between the IST numbers and the engineering calculations used to support the loss of Dam 14. Engineering determined there was no calculational basis to support the IST intake level limitation of 566.4 feet.

ComEd had not completed the investigation into this event and had not approved long term corrective actions yet. Short term corrective action included repairs and daily operation of the trash rake. This was documented the event on PIF 96-0168. The inspectors considered this an **Inspector Follow-up Item (50-254/265-96002-03)** pending review of the investigation, corrective actions and revised river water level calculations for safety related equipment operability.

1.4 Weak Log Reviews Resulted in Decreased Safety System Availability

Weak supervisory reviews of equipment operator logs resulted in a safety system being inoperable without the shift engineers knowledge and increased the time the system was inoperable.

Several non-licensed operators (NLO) identified low freon pressure in the "B" control room emergency ventilation (CREV) system refrigeration condensing unit during shiftly rounds on February 24. The NLOs identified the discrepant condition by circling the low discharge pressure condition on the round sheets. The round sheets were reviewed and signed by Senior Reactor Operator (SRO) qualified supervisors at the end of shifts 1, 2, and 3 on February 24. An oncoming NLO on February 25 elevated the discrepant condition to the shift engineer. After reviewing the previous days logs, the unit supervisor declared the CREV system inoperable as of the previous day.

This was documented on PIF 95-0644. Maintenance personnel later identified a mechanical joint and several packing leaks as the cause of system leakage. The log sheets did not show compressor discharge pressure as a Technical Specification parameter. Also, the shiftly SRO reviews of log sheets did not detect the discrepant parameter as indicative of a problem, and took no compensatory actions.

The inspectors noted loss of freon from the compressor was similar to an event documented in Licensee Event Report (LER) 50-254/95005. The LER documented "B" train CREVs becoming inoperable due to freon leaks in the compressor. The inspectors consider this item an **Inspector Follow-up Item (50-254/265-96002-04)** pending review of corrective actions to improve system performance.

1.5 Conduct of Operations

During control room tours, the inspectors noted good communications, panel checks, and pre-evaluation briefings. The inspectors attended several control room turnover briefings and found them to be less effective than previous turnover meetings held outside the control room. Problems included background noise from over 30 people in the control room, inability to hear the speaker, and limitations arising from two separate turnovers to include all operators. Management was in the process of developing solutions to these problems to improve the control room turnover process.

On three separate occasions during the Unit 1 refueling outage, poorly made or maintained hose fittings separated, causing water leaks in different areas of the reactor building. One incident involved spraying water from the demineralized water system onto the 4160 volt Bus 13 and other switchgear. ComEd took immediate corrective action to clean up the water and minimize damage. Operations management was evaluating methods to ensure hoses were kept away from risk significant switchgear.

1.6 Engineered Safety Features (ESF) System Walkdown (71707)

From inspections in the reactor building, the inspectors noted declining housekeeping conditions. The inspectors found that operators had not brought these declining conditions to the attention of management, which was a weakness in operator rounds.

The inspectors performed system walkdowns of the Unit 2 Core Spray system and the 2B, 1A, and 1B RHR subsystems. In the core spray rooms, the inspectors noted one minor discrepancy between the drawing and the as-built configuration of the low pressure injection keep filled piping and found an electronic dosimeter used for trending area radiation readings mounted improperly. The inspector briefed the system engineer on the walkdown findings.

The inspectors noted that the housekeeping in the 1B RHR room was poor. Materials and tools related to outage work in the room were not properly staged and were found throughout the room. The inspectors also noted poor lighting on the 2B RHR mezzanine level. The inspectors informed plant management of the condition.

1.7 Full Core Offloads into Spent Fuel Pool

The inspectors reviewed the evaluation of acceptability of full core offloads into the spent fuel pool. The UFSAR stated the Spent Fuel Pool Cooling and Cleanup System (SFPCCS) pool water temperature be maintained at or below 125 degrees F under a "maximum normal heat load." A maximum normal heat load was defined as the heat load from an average spent fuel batch (30 percent of core) plus the decay heat by the batch from the previous refueling. To handle the "maximum heat load" (defined as the decay heat of the full core offload plus the decay heat from two previous refuelings), the SFPCCS was designed to be supported by the RHR system in the RHR Fuel Pool Assist Mode to maintain the pool temperature at or below 150 degrees F. ComEd was complying with the UFSAR in this area. However, the assumptions and commitments from the safety evaluation dated June 9, 1982, "Reracking of the Pools With High Density Fuel Racks" were not addressed in plant procedures. These assumptions were also part of the basis in support of the UFSAR. ComEd intended to update the UFSAR with the assumptions.

Upon review of NRC Information Notice (IN) 95-54 "Decay Heat Management Practices During Refueling Outages," ComEd determined that several plant procedure changes and the UFSAR needed to be revised to reflect the frequency of a full core offload and previous licensing commitments. Procedure QCOP 1000-11, "RHR Fuel Pool Cooling Assist" was in place and available to be used. ComEd found the following assumptions and commitments from the 1982 safety evaluation required further consideration:

- A required delay of 100 hours for start of fuel offload and 6-day full core offload time was needed. These times were later incorporated into the master refuel procedure.
- RHR system availability was required for the fuel pool assist mode when the decay heat generated by the offloaded fuel was calculated to be greater than the capacity of the SFPCCS. The Master Refuel Procedure was revised to assure the spool pieces for the RHR system were installed prior to any fuel moves to the spent fuel pool.
- The need for spent fuel pool transfer canal gates to be removed for full core offload required additional information.

- The need for spent fuel pool temperature monitoring increased with loss of SFPCCS. This was addressed in a revised procedure.
- The UFSAR needed to be revised to more clearly establish whether a full core offload evolution was considered a normal or infrequent refueling evolution, and to reflect storage of other than GE 8X8R fuel.

Several changes were made in their procedures in addition to those mentioned above. The revised procedures required stopping fuel moves and instituting corrective measures when the spent fuel pool temperature reached 125 degrees F instead of 145 degrees F. The revised procedures also required fuel pool temperatures be less than 115 degrees F before fuel moves could recommence. The criteria for a full core offload regarding spent fuel pool temperature had been less conservative than for a "normal" evolution. Therefore, the spent fuel pool operating criteria for a full core offload was revised to reflect a "normal" refueling evolution. The spent fuel pool temperature limit for the full core offload was changed from 150 degrees F to 140 degrees F. The spent fuel pool of the opposite unit was intended to be maintained at or near 90 degrees F. A procedure was developed to crosstie Unit 1 and Unit 2 SFPCCS. ComEd had addressed the concerns of IN 95-54 and made procedure changes before entering refueling outage Q1R14. The status of UFSAR changes will be tracked as **Inspector Followup Item 50-254/265-96002-12.**

1.8 Strike Contingency Plan (92709)

The inspectors reviewed the ComEd strike contingency plan in accordance with Inspection Procedure 92709. During the initial review, the inspectors found some weaknesses with respect to the training of personnel to perform certain non-licensed operator and maintenance duties. A more thorough plan was developed, which was subsequently approved by the station Plant Operations Review Committee (PORC) on February 26.

1.9 RHRSW Pump Start Anomalies

During a review of PIF 95-3088, ComEd determined that technical information regarding anomalies observed during a December 22, 1995, RHRSW pump start did not correspond to the description of the event given by the operator who started the pump. The licensee took disciplinary action against the operator after the operator had apparently failed to notify management of an inadvertent start of an RHR pump vice RHRSW pump. The inspectors will review the details of this event as **Unresolved Item (50-254/265-96002-05)** following licensee investigation and review.

1.10 Follow-up on Non-Routine Events and Previously Opened Items. The inspectors used NRC Inspection Procedures 92701 and 92702 to review previously identified items and to ensure that corrective actions were accomplished in accordance with the technical specifications. This included reviewing the responses to notices of violation, IFIs, and LERs.

(Closed) Unresolved Item 50-254/265-94026-01: Out of Service Errors. The corrective actions to this item will be addressed as part of Notice of Violation 50-254/265-94029-01. This item is closed.

(Closed) LER 50-265/95002-00: Unplanned Start of Unit 2 SBDG. The unplanned start was caused by operator error due to poor self-check. Management took appropriate actions. This item is closed.

(Closed) Violation 50-254/95002-01: Human Error Events. Three separate human error events occurred with operators performance. A control room operator started two RHR pumps in lieu of two RHRSW pumps. Another control room operator failed to reduce load prior to performing turbine generator combined intermediate valve testing. A radwaste operator overfilled a clean up phase separator tank resulting in increased radiation dose to workers. The corrective actions were reviewed. This item is closed.

2.0 MAINTENANCE:

The inspectors used NRC Inspection Procedures 62703 and 61726 to evaluate maintenance and testing activities. Human errors caused minor events with little safety significance but important potential safety significance. Outage risk management for some equipment was not well planned early in the period. This resulted in Unit 1 SBDG being unavailable for longer than necessary while only being worked during one shift, and venting primary containment during power operations as described in Section 1.2.

2.1 Failure to Remove Jumper After VOTES Test

After VOTES testing of the Reactor Core Isolation Cooling (RCIC) pump suction valve, MOV 1-1301-22, workers failed to remove a jumper at the motor control center (MCC), as required by the procedure. This resulted in repeated valve cycling for approximately eight minutes. Poor communication among the VOTES technicians and the electricians supporting the test led to the event. A PIF 96-0431 was written to document the event and corrective actions.

After the completion of VOTES testing, procedure QCTP 0730-04, "Motor Operated Valve Testing Using the VOTES 100 Diagnostic Procedure," called for the removal of the jumper (step F.8) at the MCC and removal of the local test equipment (step F.9) at the valve. The VOTES technician, acting as test director, instructed the work group to begin removing the test equipment, but failed to inform the electricians of the need to remove the jumper at the MCC. A second VOTES technician subsequently took over completion of the procedure and assumed that the jumper was removed. The technician requested operations to re-energize the breaker for the motor operated valve. With the breaker energized and the jumper installed, the valve cycled continuously.

Shortly after exiting the work area, the second VOTES technician realized the jumper at the MCC was still installed. The technician contacted the control room and proceeded with the electrician to the MCC. The technician and electrician reached the MCC, heard the valve cycling, and turned the breaker off.

The work group supervisor immediately stopped all VOTES testing and began an investigation. Electricians recorded the valve operator motor temperature at 91 degrees F, and concluded that it had sustained no damage as a result of the continuous cycling

Engineers held briefings with personnel involved with MOV testing to discuss causes and solutions to the event. A temporary procedure change was implemented to ensure jumpers were removed and personnel were clear of equipment before energizing the equipment. Increased supervisory oversight was implemented for a short period to ensure the expectations of valve testing were being implemented. Failure to follow the testing procedure was a violation of TS 6.2.A. This licensee-identified and corrected violation is being treated as a **Non-Cited Violation (50-254/265-96002-06)** consistent with Section VII.B.1 of the NRC Enforcement Policy.

2.2 Human Error Events

During performance of a weekly power functional test, an instrument maintenance (IM) technician tested intermediate range monitor (IRM) 15 instead of IRM 13. The technicians stopped testing immediately after realizing they were testing the incorrect IRM. The Quad Cities Instrument Surveillance procedure, QCIS 700-7, allowed technicians to test all IRMs and in no specific order. The inspectors consider this error to be caused by a lack of self-check.

Other errors included failure of mechanical maintenance supervisors to properly ensure a diesel generator cooling water pump was safe for starting before operations implemented an out of service temporary lift. The pump was coupled to the motor, and pump work was not complete when a supervisor approved the temporary lift to run the motor. An electrical maintenance worker identified the error before equipment or personnel damage occurred. This event was described in PIF 96-536. Another error involving a chemistry discharge event is discussed in Section 4.1.

2.3 Unanticipated Start of the Unit 1 SBDG

During post maintenance testing, a procedure deficiency led to a Unit 1 SBDG automatic start. Mechanical maintenance completed over speed testing of the Unit 1 SBDG per procedure QCMMS 6600-3, "EDG Periodic Preventive Maintenance Inspection." In accordance with the procedure, maintenance personnel mechanically reset the SBDG. When personnel locally reset the annunciator, the SBDG automatically restarted. Operations personnel in the control room quickly identified the SBDG control switch was in the RUN position. The procedure did not specify the required position of the SBDG control switch. This event was reported to the NRC as an unanticipated start of a train of safety equipment.

A procedure deficiency and a lack of understanding of the SBDG start circuitry resulted in the SBDG restarting. Although there was a briefing prior to performing the evolution, the procedural error was not identified. ComEd planned to change QCMMS 6600-3 to prevent recurrence.

2.4 Follow-up on Non-Routine Events and Previously Opened Items. The inspectors used NRC Inspection Procedures 92701 and 92702 to review previously identified items and to ensure that corrective actions were accomplished in accordance with the technical specifications. This included reviewing the responses to notices of violation, inspection follow up items (IFIs), and LERs.

(Closed) LER 50-265/94008 and Rev 1: Inboard and Outboard Reactor Recirculation System Valves 2-220-44 and 45 Failed to Close during Surveillance Testing. This item was identical to IFI 50-254/265-94010-3 addressed below. The nuclear tracking data base was corrected to reflect added testing and root cause analysis performed after subsequent valve failure on October 16. The frequency of testing Units 1 and 2 220-44 and 45 valves was increased from quarterly to monthly. The inspectors verified this with the surveillance schedule. This LER is closed.

(Closed) Inspector Follow-up Item 50-254/265-94010-02: Poor Personnel Safety Practices. The inspectors identified poor safety practices which resulted in worker injuries. Management attention was increased on personnel safety issues, and the hard hat policy for contaminated areas was changed. The inspectors noted improved safety performance during Q1R14 outage, and good management attention to other personnel safety problems. This item is closed.

(Closed) Inspector Follow-up Item 50-254/265-94010-03: Solenoid Air Operated Valve (AOV) Problems. ComEd identified that Unit 2 primary containment isolation AOVs 2-220-44 and 45 would not stroke properly. The AOVs were considered inoperable, and the failures were attributed to excessive stem packing friction. As corrective actions, the valves were disassembled and reassembled and then placed on an increased testing frequency. However, the 2-220-44 valve failed to close again on October 16. The valves were declared inoperable and AOV diagnostic equipment was utilized to determine that the 2-220-44 operator required adjustment. After the adjustments were made, the valve failed the local leak rate test. The failure was attributed to the valve disc not being properly centered on the seat. The valve was repaired again, and declared operable. The inspectors noted the valve performance had improved. The licensee purchased two AOV diagnostic machines for future AOV testing. This item is closed.

3.0 ENGINEERING AND TECHNICAL SUPPORT:

The inspectors used NRC Inspection Procedure 37551 to evaluate the engineering area. The inspectors noted improvements on some recent root cause evaluations and in follow-up of industry experience. Initial root cause evaluation for the Unit 2 SBDG was poor. Inservice testing program weaknesses included testing criteria for RHRSW pumps not verified by calculations, Unit 2 SBDG fuel transfer pump check valve missed surveillance, and Unit 1 standby liquid control check valves failed surveillance similar to failures identified in Inspection Report 50-254/265-94020.

3.1 Improved Root Cause Evaluations

Root cause evaluation had been weak in past inspection period (see Section 3.7.). The inspectors noted some improvement in recent root cause evaluations being required by plant management. Improvements in investigative techniques were evident in the HPCI auxiliary oil pump trip evaluation. One continued weakness was that as-found evidence continued to be eliminated prior to evaluation of root cause. In two separate investigations, as-found breaker condition (for 125 VDC battery charger and Unit 1 HPCI gland exhauster) were not checked before technicians altered breaker conditions. The inspectors identified this problem previously during Unit 2 SBDG troubleshooting. This time, however, management identified the problem with maintaining as-found condition data, and were pursuing a solution. Engineering recently formed a team of experienced engineers and consultants to improve the overall root cause process. The team was in the development stages at the close of the period.

3.2 Improved Assessment of Scram Solenoid Pilot Valve (SSPV) Problems

Engineering was proactive in assessing problems with SSPV timing issues this period. After replacing all Unit 2 and most of Unit 1 SSPVs with new solenoids containing Viton diaphragms, engineers noted increased scram timing for some rods of up to 30 milliseconds (MS) for the first 5 percent of rod motion. Vendor (General Electric) information indicated possible problems with Viton diaphragms sticking to seating surfaces. While seeking guidance from NRC Information Notice 96-07, "Slow Five Percent Scram Insertion Times Caused by Viton Diaphragms in Scram Solenoid Pilot Valves," and the Boiling Water Reactor Owners Group (BWROG), ComEd implemented interim corrective actions including, increased frequency scram testing for Technical Specification required sample groups plus testing on the 10 slowest control rods. Some thermal limit setpoint parameters were adjusted to add conservative margin while SSPV performance was in question. An operability determination was performed for the control rods in conjunction with PIF 95-3031.

3.3 125 VDC Battery Charger

On January 25, 1996, the DC output breaker on Unit 1 Battery Charger No. 1 tripped unexpectedly. System Engineering determined the most likely cause to be high resistance at the cable connection resulting in thermal overload. This cause could not be confirmed because the breaker was removed from service before the resistance at the connection could be measured. Aside from the inability to confirm the root cause, the inspectors noted that the investigation appeared to be thorough.

The 125 VDC system had been put on equalizing charge following a successful four-hour battery discharge test. The system engineer observed the charger operating in a current limiting mode (227 amps) during the first one-half hour of charging. Approximately one hour later, an operator found the DC output breaker in a tripped condition, and reset it. The breaker functioned normally after the reset. Initially system engineering suspected problems with the current limiter. The breaker was bench tested and four-hour load tested at an elevated current limit setting (240 amps) without duplicating the tripped condition. Consultations with the vendor helped engineers narrow the root cause to a loose cable connection resulting in high resistance, localized heating, and a premature thermal overload trip.

No load test of the battery charger had been performed on a regular basis. However, the technical specification upgrade program (TSUP) will require such a test. Using the lessons learned from this event, ComEd was developing a test procedure to conduct a four hour battery charger load test once every refueling outage. The procedure is expected to include a step to take temperature readings at the connections before the test is conducted.

3.4 Material Condition Issues

During the inspection period, equipment problems continued to affect facility operation, challenge operators and cause increased unavailability times of safety related equipment. The following are examples of equipment problems for the period:

- Reactor building 2A closed cooling water pump power supply breaker tripped.
- Intake structure trash rake failed and resulted in both fire pumps becoming inoperable.
- 2B service water pump motor developed a fault resulting in other non-essential loads tripping from the undervoltage condition.
- Shared SBDG immersion heater failed resulting in operations declaring shared SBDG inoperable.
- Unit 1 HPCI auxiliary pump oil pump relief setpoint was found set too high with pump cutout pressure switch set too low.
- Turbine Control Valve No. 2 oscillated uncontrollably. This required operators to remove Unit 2 from service to repair a failed servomotor.

3.5 Safety Equipment Inoperable Due to Missed Surveillance

On January 29, 1996, ComEd identified a failure to inspect the Unit 2 SBDG fuel oil transfer pump discharge relief inlet check valve (2-5299-3). This failure was reported under PIF 96-221. Another IST testing requirement problem was documented in PIF 95-2851. The inspectors will review the corrective actions for these PIFs, once complete, and track them as **Unresolved Item (50-254/265-96002-07)**.

3.6 Problems With Safety Related CREV System

The inspectors noted a high incidence of CREV equipment deficiencies. A 14-day limiting condition for operation (LCO) was entered five times during the inspection period. Four entries were unplanned, requiring ENS notifications, and one entry was for planned corrective maintenance. The dates and reasons for the four unplanned entries were:

Feb. 3 - Outside air temperature dropped below minimum design temperature of -28.1 degrees F.
Feb. 21 - CREVs would not start or operate after a loss of power to MCC 16-3.
Feb. 25 - Low freon pressure in compressor due to system leaks.

March 4 - Both suction and discharge dampers on the air handling unit fan failed to close due to a failed relay.

Section 9.4. of the UFSAR stated that the ventilation system for Control Room Area, Turbine Building and Reactor Building were designed to maintain adequate inside air temperatures with outside air temperatures between -6 degrees F and 93 degrees F. The inspectors questioned the operability of plant equipment when outside air temperature dropped below -6 degrees F. Temperatures during the inspection period went below -28 degrees F. ComEd was reviewing which components were sensitive to building temperatures outside of the UFSAR design basis.

On January 22 maintenance and engineering staff identified during reassembly of the "B" CREV refrigerant compressor, the compressor head was improperly aligned with the cooling channels. The condition existed since initial construction and resulted in less efficient compressor operation and contributed to the erosion found in the condenser. A previous opportunity to identify this misalignment occurred during an inspection of the unit in 1992. The inspector reviewed the maintenance work history which indicated that the misalignment was detected but the vendor provided incorrect information which led the maintenance personnel to reassemble the unit in the same incorrect alignment. Engineering performed an operability screening and concluded that even with reduced efficiency of the refrigerant condensing unit, the system would meet its design function. The resolution to the problem was to properly align the condenser head and attached piping. The questioning attitude of the individuals prevented reassembling the compressor in the degraded condition.

Due to good communications with another licensee, Quad Cities engineering identified a design deficiency which would prevent the CREV filtration fans from starting and operating after a loss of power to Unit 1. The toxic gas analyzer was designed to stop the CREV booster fans and close dampers upon detection of high concentrations of ammonia at the booster fan intake. The safety function of the CREV system was to maintain radiological protection for personnel in the control room zone after a design basis accident. Protection was designed to be provided by pressurizing the control room volume with filtered air from the booster fan intake. A loss of power to MCC 16-3 (Unit 1 non-safety related bus powering the toxic gas analyzer), concurrent with a loss of coolant accident, would prevent the CREV system from performing its intended safety function.

ComEd determined manual removal of a non-safety related relay in the toxic gas analyzer circuit during a loss of power to MCC 16-3 would allow for proper operation of the CREV system. The procedure QCOP 5750-09, "Control Room Ventilation," was changed to remove the relay during the CREV starting sequence. This action was intended to ensure loss of power to MCC 16-3 would not inhibit the safety function of the CREV system.

The operators were trained on the new procedure revision. The inspectors reviewed the revision to QCOP 5750-09 and verified accessibility and labeling of the affected relay.

During testing of the service water relief valve on the inlet side of the compressor condenser, engineers noted relief valve (RV-1/2-5741-345) was set at 300 psi. Engineering determined the relief valve was required to be set at 150 psig. The incorrect relief valve setting had existed since initial system installation in 1984. In August 1995, during a system walkdown to update drawings to match as-built configuration, engineers noted the discrepancy between the installed valve setpoint (300 psig) and the drawing setpoint (150 psig), assumed that the installed valve was correct, and requested a drawing change. Upon discovery of the incorrect setpoint in January 1996, maintenance installed a new relief valve set at 150 psi and engineering requested a second drawing change. An **Inspector Followup Item (50-254/265-96002-13)** was issued to track resolution of control building ventilation discrepancies.

3.7 Unit 2 Standby Diesel Generator Followup

The licensee concluded the investigation of the 1995 Unit 2 SBDG failures and attributed the failures to a combination of erratic operation of the governor shutdown solenoid and a binding air start motor. The inspectors identified the following weaknesses with the troubleshooting effort and corrective actions:

- Initial troubleshooting efforts were weak, leading to a prolonged period of substandard system reliability. Problems related to the start failures were first indicated in August 1995 and continued into November 1995.

- The investigation identified that one of the failed air start motors had been improperly stored. The licensee did not investigate the storage controls of the air start motors in a timely manner.

- The licensee failed to re-open the investigation of root cause after the initial failure, when materials testing showed that the initial root cause given could not be supported.

- The issue of pre-conditioning while performing tests to declare operability was not adequately considered. The initial conditions of the intermittent failures included a significant time period prior to a start attempt and subsequent failure to start. However, earlier tests for operability were done after short time periods between start attempts or following a previous run. These runs tended to mask problems which arose after the engine had been sitting for some time. Later operability tests were run with consideration of "preconditioning" effects.

- A 10 CFR Part 21 notification, (10 CFR 21-0045, dated April 28, 1989), was issued identifying adverse effects of moisture on the carbon vanes of the SBDG air start motors. The engineering resolution recommended implementation of a moisture controlled storage environment for the SBDG air start motors and also for the spare rotor vanes for these motors. Following the Unit 2 SBDG failure to start events on September 26 1995, and October 24, 1995, ComEd determined that one of the root causes of the start failure events was a faulty air start motor. This was documented in PIR 265-200-95-161. On November 22, 1995, following the Unit 2 SBDG start failure events, the inspector questioned the system engineer about storage controls of the SBDG air start motors. These air start motors were installed on the Unit 2 SBDG on March 29, 1995. The system engineer responded on December 5, 1995, that storage controls were not implemented. On December 7, 1995, PIF 95-2982 was generated, identifying that, contrary to the Part 21 notification and engineering's resolution, these air start motors had been improperly stored.

Failure to store the air start motors in a moisture controlled environment was a **Violation of 10 CFR 50, Appendix B, Criteria XIII (50-254/265-96002-08)**.

3.8 Unit 1 HPCI System Inoperable

Measures necessary to assure sustained HPCI system reliability have not yet been achieved. On January 30, 1996, the plant operators declared Unit 1 HPCI inoperable due to unexpected system anomalies that occurred during performance of the HPCI manual initiation test. The Unit 1 HPCI system has been inoperable due to system malfunctions numerous times in the past year.

PIF 96-0246 and LER 1-96-004 were issued to address anomalous conditions and subsequent problems with the HPCI system gland exhauster. The following anomalies occurred during the test:

- The Auxiliary Oil Pump (AOP) cycled on and off a number of times.
- The Emergency Oil Pump (EOP) started and continued to run.

- HPCI AOP motor overload annunciated.
- The 250V DC Battery Bus undervoltage alarm annunciated.

The auxiliary oil pressure regulating valve (PRV) No. 3 was designed to open to regulate oil pressure at about 50 psig until the HPCI turbine reached sufficient speed for the attached oil pump to maintain system pressure. The control pressure then was meant to increase to above 60 psig, at which point the AOP cutout pressure switch (PS No. 4) would actuate to trip the AOP.

Comed found PRV No. 3 setpoint was controlling too high (58 psig), and PS-4 was set at the low end of its setpoint tolerance (58 psig). Before the pump could control on PRV No. 3, PS No. 4 would actuate to open the breaker for the Aux. Oil Pump. As pressure decreased, PS No. 4 reset causing the AOP to restart, and the cycle was repeated. The root cause of this problem was determined to be inadequate preventive maintenance and trending of the HPCI oil system. The PRV No. 3 was reset to the correct setpoint, PS No. 4 re-calibrated to the middle of its tolerance range, and improvements initiated to the HPCI oil preventive maintenance and trending program. The anomaly of the EOP starting when the AOP started had been a common occurrence during HPCI system testing. ComEd determined that this deficiency is not a HPCI system operability issue and generated PIF 96-0246 to address the corrective actions.

ComEd determined that the AOP motor overload alarm was probably due to a sticky time delay relay in the starting circuit which caused the untimely bypass of a starting resistor. This would have allowed higher than normal in-rush current, causing the HPCI AOP motor overload alarm and also the 250 V DC battery bus undervoltage alarm. The battery bus undervoltage alarm was found set at 253 volts versus 250 volts. The AOP time delay bypass switches were replaced in the starting circuit, and the 250 VDC undervoltage alarm setpoint was reset to the proper value. The problem did not repeat during subsequent operability testing.

The existing preventative maintenance program had not been effective in assuring that system control components are adequately tested, calibrated, and maintained. The following issues were identified:

- The pressure regulator (PRV No. 3) for the Aux. oil pump had not been adjusted since 1988.
- The gauge (PG No. 5) used to adjust the regulator (PRV No. 3), installed on the HPCI turbine front standard, was found to be reading about 4 psig low. It is possible that this gage was inaccurate at the time of the regulator adjustment in 1988, causing the high regulator pressure.
- The 250 V DC undervoltage alarm had not been calibrated since 1991.

Cycling of the AOP caused intermittent loss of control oil pressure to the HPCI turbine steam stop valve. Consequently, the steam stop valve cycled, interrupting steam flow to the HPCI turbine enough to cause a slow initiation time of 43 seconds. The design requirement is for the system to provide 5000 gpm to the reactor within 45 seconds of initiation which is about twice nominal. A slow time of about 40 seconds in April 1995 could have been an indicator of degrading system condition. ComEd did not initially classify the slow time of 43 seconds for the HPCI system to reach 5000 gpm as a significantly degraded condition, although the start time had increased significantly.

On February 5 during Unit 1 HPCI system troubleshooting runs, the gland exhauster breaker tripped. The breaker trip settings for the gland exhauster breaker had drifted and were found to be too low. This malfunction also would have rendered the system inoperable. Additional troubleshooting of the gland exhauster problem extended the duration of the Unit 1 HPCI system inoperability. The Unit 1 HPCI gland exhauster breaker was replaced and tested for proper operation, the Unit 2 gland exhauster breaker checked for proper operation, and an operability test performed to verify integrated system response following all repairs associated with this event.

Operations response to the recent system anomalies was positive. Although detailed, the engineering staff's initial resolution failed to address all possible causes of the problems. The Station Manager directed the engineering staff to investigate all possible root causes and to assess reliability related corrective actions that go beyond the elements of this event.

On February 9, 1996, Operations ran the monthly test on the Unit 1 HPCI system. Operators found that the limit switch providing indication that the turning gear was disengaged had failed. The operators verified that the turning gear was disengaged according to the procedure. A work request was initiated to change the switch or adjust the linkage during Q1R14.

Additional HPCI system analyses, evaluation for reliability assurance and performance improvements were initiated. The resultant recommendations from these analyses were documented in the licensee's Nuclear Tracking System (NTS) and in LER 1-96-004 and PIR 254-180-96-04. Satisfactory operability testing of the system was performed and the system was declared operable on February 9, 1996, prior to the Unit 1 shutdown for refuel outage Q1R14.

3.9 Units 1 and 2 Residual Heat Removal (RHR) Corner Room Structural Steel

Some of the structural steel beams and connections supporting the RHR heat exchangers were determined to be overstressed relative to UFSAR allowable stress limits. ComEd completed an operability determination (PIF 95-2256 dated August 18, 1995) with supporting functionality evaluation (calculation QDC-0020-S-0055 dated October 2, 1995). The operability determination showed that the analyzed beams meet functional criteria, but did not meet UFSAR allowable stress limits. Initial discovery was made during contractor reviews and walkdowns of associated piping supports, where a number of pipe supports were not accounted for in existing calculations. The inspectors consider this an **Unresolved Item (50-254/265-96002-09)** pending further inspector review.

3.10 Follow-up on Non-Routine Events and Previously Opened Items. The inspectors used NRC Inspection Procedures 92701 and 92702 to review previously identified items and to ensure that corrective actions were accomplished in accordance with the technical specifications. This included reviewing the responses to notices of violation, IFIs, and LERs.

(Closed) Violation 254/265-93024-04A and Inspection Follow-up Item 254/265-94004-48: ComEd Failed To Evaluate and Identify Appropriate Corrective Action For Vendor Service Information Letters (SILs). A diagnostic evaluation team (DET) report discussed the failure to review industry information.

ComEd acknowledged that SILs were not being reviewed within the 90-day administrative time limit. All SILs were subsequently reviewed and evaluated for which responses could not be found. The inspector verified that there were no SILs where the evaluation or corrective action was past due. This item is closed.

(Closed) Violation 254/265-93024-04B: Poor Corrective Action. ComEd failed to complete corrective actions to identify and replace defective safety related and important to safety 4kV breaker switches. This occurred because of an inadequate review of procedure changes. The original vendor inspection criteria that addressed failure of 4kV breaker switches had been incorporated into procedure QCEPM 200-3, "Inspection and Maintenance of 4kV Vertical Circuit Breakers Type 4.16-350," Revision 3. However, the inspection requirements were inadvertently deleted from a subsequent revision to procedure QCEPM. The inspector noted that the latest revisions to procedures QCEPM 200-1 and 200-3 incorporated the vendor switch inspection criteria. This issue is closed.

(Closed) Violation 254/265-93024-04C: Poor Corrective Action. A number of broken or damaged 4kV breaker parts were identified indicating improper handling. However, corrective actions had not been taken to prevent the recurring problems. Failures of 4KV breakers have occurred which may have been the result of rough handling. Training was held to improve operator handling of the 4kV breakers. In addition, procedure QCAP 200-1, "Conduct of Operations," had incorporated guidance and expectations with respect to the proper handling of 4kV breakers. A review of records did not indicate a high failure rate of 4kV breakers. This issue is closed.

(Closed) Inspection Follow-up Item 254/265-94004-14: Recurring DC Ground. Although DC grounds were being properly located and corrected, the licensee had not determined the root cause or causes for recurring hard grounds. In addition, many of the grounds that alarmed in the control room were either not documented or the information was not communicated to system engineering. The licensee has since revised procedure QCOP 6900-19, "Documenting 125/250 VDC Grounds," that includes documenting grounds associated with the safety related 125 and 250 Vdc battery systems and the 125 Vdc station blackout battery system. In addition, the inspector noted that the procedure required a ground report for each intermittent and hard ground identified. The purpose of the ground report was to provide pertinent plant information to the DC system engineer for trending and evaluating root causes. This issue is closed.

(Open) Violation 254/265-95002-02: Design Control Violations. The inspectors identified RHR pressure switch setpoints were set outside of manufacturer calibration range. In addition, the inspectors identified two examples of the controlled data base not reflecting completion of two safety-related modifications to the facility.

ComEd attributed the RHR pressure switch setpoints set outside the manufacturers calibration range to documentation errors and miscommunication between site and corporate personnel. Exempt changes were issued to replace the affected pressure switches. The inspectors noted the 1(2)-1053 series pressure switches were replaced in both units. However, the control data base for the Unit 1 switches was not yet updated.

The inspectors identified two examples of safety-related modifications having been completed but the controlled data base was not updated in a timely manner. Lessons learned provided by Design Engineering to site engineering personnel stressed the procedural requirements to process design change requests within 30 days of operations acceptance of the change. The way setpoint changes were processed was also modified. Setpoint changes, from procurement to calculation to installation, were controlled on site. The inspectors noted that the control data base had not yet reflected the replacement of Unit 1 1001-88 series, 1001-89 series, 1001-90 series, and 1001-83 series of pressure switches with new model pressure switches. Similarly, the inspectors noted the "I Data Base" had not yet reflected the setpoint changes for Unit 1 263-111 series pressure switches.

The inspectors verified commitments in the response to the Notice of Violation were satisfied, but the problem was not yet resolved. However, ComEd attributed a large backlog of DCRs as the reason for the data base not being updated. The inspectors consider this item open since the controlled data base still had yet to be updated for the above Unit 1 pressure switches. The inspectors will review this item after the controlled data base is updated.

(Closed) Inspector Follow-up Item 50-254/265-95003-01: Intake Screens Removal Without Evaluation. The inspectors identified poor control of intake screen removal. Engineering evaluations determined that removal of the RHRSW bay screens were acceptable if properly controlled by procedure. The inspectors ensured the revised procedure was in place. This item is closed.

(Closed) LER 50-265/95004-01: All Four Condenser Vacuum Scram Switches Out of TS Calibration. With Unit 2 shutdown, ComEd identified during quarterly surveillance testing that all four condenser vacuum switch setpoints were out of TS calibration limits. Due to setpoint drift with the Barksdale model B1T-H18SS pressure switches, these switches were previously replaced with an identical model. After experiencing setpoint drift with the new switches, the vacuum switches were replaced with identical models again and the testing frequency was increased. Engineering did not want to replace the switches with a different model due to cost and the intermittent drift characteristics of the switches setpoints. The switches were submitted for testing and analysis. The switches were found to be prone to setpoint drift from high temperature and humidity environments. Engineering submitted a request to replace the switches with more reliable models. The switch replacement was scheduled for Q1R14 and Q2R14. This item is closed.

(Closed) Unresolved Item (50-265/95004-02): Local Leak Rate Testing (LLRT) Problems. An LLRT director found two standby liquid control valves out of position closed with out of service (OOS) tags attached. The OOS tags required the valves to be open. The LLRT director positioned the valves with the OOS tags attached to the open position. This violated Quad Cities Administrative Procedure (QCAP) 230-4, "Equipment Out of Service," Section C.2.c which stated, "A component shall never be operated with an associate OOS card on it." During the investigation, ComEd identified two other procedural violations attributed to the LLRT director. The LLRT director performed Procedure Field Change (PFC) 1343 eight separate times without terminating the PFC after its performance. This was a violation of QCAP 1100-13, "Processing Procedure Field Change," Section 2.2, which stated, "The PFC is initiated for the performance of a task and is terminated upon completion of the task." Additionally, the individual added a step to PFC 1409 without obtaining approval from the unit supervisor. The inspectors consider this a violation of QCAP 1100-13, "Processing Procedure Field Change," Section D.2.c which required obtaining PFC request approval from the unit supervisor. The above examples were considered violations of Technical Specification 6.2.A.1. which required procedures be established and implemented. This licensee-identified and corrected violation is being treated as a **Non-Cited Violation (50-254/265-96002-10)** consistent with Section VII.B.1 of the NRC Enforcement Policy. The individual involved with these events was disciplined by management. The individual made no attempt to cover up his actions and was aware of the potential consequences of his actions. This item is closed.

4.0 PLANT SUPPORT:

The inspectors used NRC Inspection Procedure 83750 to evaluate plant support activities. A chemistry-related personnel error resulted in excessive sodium bisulfite discharge. Outage ALARA planning efforts were good and several actions were planned to reduce radiological source term. Failure to plan and monitor for higher than anticipated reactor coolant activity resulted in increased refuel floor exposure.

4.1 Chemical Discharge to River above Limits

A human error resulted in the discharge of sodium bisulfite to the Mississippi river in amounts greater than that allowed by discharge limits.

The chlorine-based biocide solution is used for river water systems. The sodium bisulfite solution is also used to neutralize any remaining biocide solution in the water discharged back to the river to ensure chlorine discharged to the river would be within discharge limits. On January 23, chemistry technicians secured the biocide solution for work on the intake structure but failed to secure the neutralizing agent. This resulted in a discharge of about 623 pounds of bisulfite above the discharge limits of 5000 pound in 24 hours.

A chemistry technician secured the biocide injection system and initialed completion of the steps in chemistry procedure, QCCP 700-22, "Operation of Circulation Water Biocide and Sodium Bisulfite Systems." The technician was distracted and did not secure the bisulfite solution. During review of procedures later that morning, a supervisor identified steps for securing the bisulfite injection system were unsigned. The technician, believing he had secured the bisulfite solution, signed the steps as completed. The following day, the same technician identified the bisulfite injection system was still operating and notified management of the error made.

The event was documented on PIF 95-0191 and was investigated. This human error was attributed to distraction. Additionally, the individual failed to self check both the actions performed and procedure completion at the end of the evolution. The corrective action included installation of a timer to automatically secure the bisulfite pumps, counseling the individual, and reviewing the event with chemistry department personnel. The inspectors reviewed the corrective actions and aquatic toxicity report. The inspectors concluded there was no nuclear safety consequences and minimal environmental significance to this event.

The inspectors consider this event as violation of Technical Specifications 6.2.A, failing to implement procedure QCCP 700-22. This licensee-identified and corrected violation is being treated as a **Non-Cited Violation, (50-254/265-96002-11)**, consistent with Section VII.B.1 of the NRC Enforcement Policy.

4.2 Radiation Protection

4.2.1 Higher than Expected Radiation Levels on Refuel Floor

Early isolation of the reactor water cleanup (RWCU) system after Unit 1 shutdown and cooldown prevented removal of activated wear products from the reactor coolant system (RCS). This resulted in higher radiation levels on the refuel floor after reactor disassembly and reactor cavity fill.

Unit 1 shutdown and cooldown commenced early on February 10. These two evolutions resulted in the release of activated wear products into the RCS. The RWCU system, which removes wear products and reduces shutdown radiation levels from the RCS, was taken out of service on February 12 prior to the removal of the wear products which were released into the RCS during the shutdown. Neither radiological protection or chemistry personnel recognized the implications of removing the RWCU system from service early. After filling the refuel cavity on February 14, radiation protection technicians identified higher than anticipated radiation levels around the reactor cavity. Radiation protection management was notified and decided to continue with some low dose jobs through the night. Later, all work was suspended in and around the reactor cavity and operations filtered the cavity water through the fuel pool demineralizers. This resulted in radiation levels in the area returning to normal levels. ComEd estimated that about one additional man-rem exposure was acquired from the higher radiation levels on the refuel floor and significant scheduling and work delays were induced. The licensee verbally notified other company boiling water reactor sites of this event and planned to submit written notification later.

This issue was documented on PIF 96-0483. The investigation determined the higher radiation levels were due predominantly to cobalt-60 activity which was pushed from the RCS into the reactor cavity during the fill evolution. Chemistry planned to develop a criteria for removing RWCU from service based on RCS activity. This criteria would ensure sufficient activity was removed from the RCS prior to future refuel cavity fill evolutions.

4.2.2 Review of Unit 1 Outage Planning

The outage dose goal was 470 rem. The work scope appeared fixed and included a 30 percent allowance for emergent work. Significant work (and associated dose goals) included:

- Reactor water cleanup (RWCU) modification (68 rem)
- Inservice Inspection (ISI) (47 rem)
- Turbine Work (36 rem)
- Valve Work (75 rem)
- Residual heat removal (RHR) heat exchanger repair (16 rem)
- In vessel work (16 rem)

Several ALARA initiatives were planned for these tasks. An ALARA manager was designated for each project and contingency plans were developed and in place. The use of lead shielding was also increased. Specific initiatives for RWCU included removing a highly contaminated line near the job site, and performing pre-fabrication in low dose rate areas. The work scope was fixed and included lesson's learned from similar modifications on Unit 1 and at Dresden station.

Initiatives for ISI work included decreasing the amount of temporary scaffolding and placing some permanent scaffolding in containment.

The planned turbine work included an overhaul of one low pressure (LP) and one high pressure (HP) turbine. The estimated dose was higher than usual based on initial survey results which identified dose rates of 3-5 mrem/hr (about 2 times higher than in 1994) and possible increased contamination levels on the turbine components. ComEd believed the increase resulted from hydrogen addition (both units). Although a subsequent survey indicated general area dose rates had declined to $\leq$ 1 mrem/hr (similar to 1994), ComEd remained concerned about contamination levels and will use sandblasting and floor control to mitigate dose and contamination spread.

Several source term reduction efforts were planned, including: hydrolasing of high dose piping; installation of low cobalt replacements for the RHR A, B, C pump suction valves and several control rod drive (CRD) blades; removing the CRD sink drain line; performing chemical decontaminations of the RWCU and RR systems; and installing permanent shielding in the RWCU heat exchanger room. ComEd estimated that these efforts would save about 650 rem.

4.2.3 Internal Monitoring

ComEd was considering implementing a passive monitoring program to replace certain types of whole body counting. An analysis indicated that the gamma sensitive portal monitors could detect levels of internally deposited radioactive material at $\leq$ 1 percent of the Annual Limit of Intake (ALI) with excellent reliability. The inspectors reviewed the study results (actual monitor performance was not validated) and concluded that the use of these monitors for passive monitoring was sound.

5.0 **SAFETY ASSESSMENT/QUALITY VERIFICATION** (40500)

5.1 Plant Operations Review Committee

The inspectors observed the Plant Operations Review Committee meetings and determined that reviews were thorough in most cases, but that packages brought to the committee often required revision. Inspectors noted two weaknesses in the functioning of the committee, involving independence of members and preparation of material for review.

In several PORC meetings, the inspector noted the minimum number of committee members present, and one of the voting members also acting as the sponsor for the issue being briefed. After discussing this apparent conflict with members of the committee, the inspector found that some committee members also shared this concern.

The inspectors found that many briefing packages were not submitted for PORC members review two days in advance of the meeting as recommended by QCPP 1001 "PORC." In some cases the packages were submitted at the time of the meeting. This did not allow the committee members to review the material before the meeting.

6.0 REVIEW OF UFSAR COMMITMENTS

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. Some discrepancies with UFSAR commitments were noted and addressed in Sections 1.7. and 3.6.

7.0 EXIT INTERVIEW

The inspectors met with the ComEd representatives denoted below on March 1, 1996, and summarized the scope and results of the inspection and discussed the likely content of this inspection report. ComEd acknowledged the information and did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.

The following management representatives attended the exit meeting conducted on March 1, 1996, along with others.

ComEd

Ed Kraft, Site Vice President
Bill Pearce, Station Manager
Alan Blamey, Station Support Engineering Supervisor
Nick Chrissotimos, Regulatory Assurance Supervisor
Dave Craddick, System Engineering Supervisor
Sharon Eldridge, Design Engineering Supervisor
Frank Tsakeres, Radiation Chemistry Superintendent
Mike Wayland, Maintenance Superintendent

8.0 DEFINITIONS

8.1 Non-Cited Violations

The NRC uses the Notice of Violation as a standard method for formalizing the existence of a violation of a legally binding requirement. However, for a violation of minor safety significance or a licensee identified Severity Level IV violation that meets the criteria described in the NRC Enforcement Manual, the NRC will not generally issue a Notice of Violation. Violations of regulatory requirements identified during this inspection for which Non-Cited Violations were issued are discussed in Sections 2.1., 3.10., and 4.1.

8.2 Unresolved Items

Unresolved Items are matters about which more information is required in order to ascertain whether they are acceptable items violations or deviations. Unresolved Items disclosed during this inspection are discussed in Sections 1.9., 3.5., and 3.9.

8.3 Inspector Follow-up Items

Inspector Follow-up Items are matters which have been discussed with the licensee which will be reviewed further by the inspectors and which involve some action on the part of the NRC or licensee or both. Inspector Follow-up Items disclosed during this inspection are discussed in Sections 1.3., 1.4., 1.7. and 3.6.