U. S. NUCLEAR REGULATORY COMMISSION

REGION III

REPORT NO. 50-454/95009(DRP); 50-455/95009(DRP)

FACILITY

Byron Station, Units 1 & 2
License No. NPF-37; NPF-66

LICENSEE

Commonwealth Edison Company
Opus West III
1400 Opus Place
Downers Grove, IL 60515

DATES

September 19 through November 6, 1995

INSPECTORS

H. Peterson, Senior Resident Inspector
C. H. Brown, Resident Inspector
N. D. Hilton, Resident Inspector
D. L. Schrum, Fire Protection Specialist
E. R. Duncan, Resident Inspector, Braidwood
A. McMurtray, Resident Inspector, Point Beach

APPROVED BY

Martin J. Farber, Chief
Reactor Projects Branch 4

12/7/95
Date

AREAS INSPECTED

A routine, unannounced inspection of operations, engineering, maintenance, and plant support was performed. Safety assessment and quality verification activities were routinely evaluated. A routine, unannounced fire protection inspection of surveillances, equipment, impairments, control of combustibles, fire brigade training and drills, and fire protection audits was also performed.

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RESULTS

Assessment of Performance

OPERATIONS: The inspectors routinely observed professional control room operations and operators demonstrated a strong questioning attitude. The questioning attitude was demonstrated by repeated questioning of the leak detection capability requirements which eventually identified two inadvertent entries into Technical Specification 3.0.3 (paragraph 1.1.1). On one occasion, an operator missed a step in a procedure that caused the control rods to step in and resulted in a few degrees cooling of reactor coolant (paragraph 1.1.2). Unit 1 was shutdown October 22 for a mid-cycle outage to inspect the steam generator tubes. During reactor coolant loop drain, nitrogen used during the blowdown of the loop leaked past an A Loop Stop Isolation Valve and displaced approximately 1400 gallons of water from the reactor vessel to the pressurizer (paragraph 1.2.1).

MAINTENANCE: Involvement and coordination of routine surveillance and maintenance activities were reviewed by the inspectors, and no major concerns were noted. Early maintenance activities during the mid-cycle outage appeared to be well performed. A rotated heat exchanger end bell on the Unit 2 A Safety Injection pump was identified during this period, however, the end bell was most likely installed incorrectly in 1993. Inadequate corrective action for a similar condition on the Unit 2B Chemical and Volume Control Pump resulted in a Non-cited Violation (paragraph 2.1). Additionally, missed surveillances on 125 volt battery terminal connections resulted in a Non-cited Violation for inadequate procedures (paragraph 2.2).

ENGINEERING: Several issues required engineering support this period. Proactive operations support was demonstrated by identification of a degrading feedwater flow venturi transmitter. A violation was identified for failure to have integrated leak tests to identify leakage from systems that could contain highly radioactive fluids outside the containment after an accident (paragraph 3.1).

PLANT SUPPORT: Fire protection performance was good with the exception of certain impaired fire barrier procedures not being followed. A violation was identified for a failure to authorize an impairment for fire doors blocked open to support flushing activities (paragraph 4.1.3). Radiation Protection prepared for the mid-cycle outage well and performed early outage activities without incident. The Emergency Protection group demonstrated very good performance during the pre-exercise drill, the drill was well thought out, and the performance of the personnel was considered to be very good.

SUMMARY OF OPEN ITEMS
Violations: identified in Sections 3.1 and 4.1.3
Unresolved Items: None
Inspector Follow-up Items: identified in Sections 3.4 and 3.5
Non-cited Violations: identified in Sections 2.1 and 2.2

INSPECTION DETAILS

1.0 OPERATIONS

NRC Inspection Procedure 71707 was used in the performance of an inspection of ongoing plant operations. The inspectors determined that the licensee effectively carried out its responsibility to oversee and direct plant operations. No violations were identified.

1.1 Performance of Operations at Power

The inspectors routinely observed professional control room operations and operators demonstrated a strong questioning attitude. The questioning attitude was demonstrated by repeated questioning of the leak detection capability requirements which eventually identified two inadvertent entries into Technical Specification 3.0.3 (paragraph 1.1.1). Additionally, on one occasion, an operator missed a step in a procedure that caused the reactor control rods to step in and resulted in a few degrees cooling of reactor coolant (paragraph 1.1.2).

1.1.1 Leak Detection Capability Lost

The licensee identified that reactor coolant leak detection capability as required by Technical Specifications (TS) was inoperable on two occasions, September 13 and September 15, 1995 respectively. Each example was an inadvertent entry into TS 3.0.3. Licensee Event Report (LER) 454: 95-0003 was issued on October 11, 1995.

TS 3.4.6.1 required containment particulate radioactivity monitoring, floor drain and reactor cavity flow monitoring, and containment gaseous radioactivity monitoring systems to be operable. A radiation monitor system, PR11J, contained both the particulate and gaseous radioactivity functions. Both the containment floor drain and reactor cavity drain functions were performed by the containment drain monitoring system, RF008.

The licensee identified both PR11J and RF008 were inoperable on Unit 1 for a period of 2 hours and 47 minutes on September 13, 1995 due to calibrations in progress on each instrument. Operating personnel were aware of the concurrent entry into two TS action statements but believed concurrent action statement entries were acceptable.

The evening of September 14, the Containment Leak Detect Flow High alarm (from RF008) was received on the main control board for Unit 1. Indicated leak rate was 1.6 gallons per minute. A primary leak rate determination was performed to verify the leakage was from a non-primary source. Following verification that the leak was not reactor coolant, a TS action statement was conservatively entered to reset the RF008 alarm setpoint. The alarm reset was to regain the alarm function on the main control board. On September 15, following another review of the TS, the calibration of PR11J was allowed to resume. With RF008 still declared inoperable waiting for the alarm setpoint change, TS 3.0.3 was again inadvertently entered. The second event lasted 5 hours and 30 minutes, concluding when the calibration of PR11J was completed and the associated action statement exited.

The licensee documented corrective action, including a review to identify other TS where the action statement requirements may not be obvious, in the LER. The inspectors will review and close the LER in a future report.

1.1.2 Rods Step In During Power Instrument Adjustment

A secondary heat balance had been completed and the Unit 1 operator was completing the power range instrument gain adjustments when the rods started to step in. The operator responded to the panel and noted that rod control was in automatic. The operator placed rod control in manual to stop the rod motion. The evaluation found that the locking tab on the gain adjustment potentiometer for N-44, the last power range drawer to be adjusted, had slipped when the locking tab was pushed, increasing the gain setting. The operator had not placed rod control in manual per the step in the heat balance procedure and was counseled to be attentive to procedures. The rods stepping in caused only a small drop in temperature since the rods were quickly returned to the original position.

1.2 Performance of Operations while Shutdown

Unit 1 was shutdown October 22, 1995 for a planned 32 day outage to inspect the steam generator U-tubes. Other major work planned; the second phase of the Natural Draft Cooling Tower modification, 10-year inspection of 1A Reactor Coolant Pump motor and inspection of Unit 1 Main Condenser. During reactor coolant loop A draining, nitrogen used during the blowdown of the loop leaked past an A Loop Stop Isolation Valve and displaced approximately 1400 gallons of water from the reactor vessel to the pressurizer (paragraph 1.2.1).

1.2.1 Nitrogen Leak Past Loop Stop Isolation Valves

On October 26, 1995, the licensee was draining the Unit 1 A Reactor Coolant System (RCS) loop for Steam Generator (SG) eddy current testing. Nitrogen (N_2) was connected to the loop to assist in the draining. With the pressurizer at 30 percent initially, an operator noticed pressurizer level increasing. Soon after pressurizer level was noted as increasing, the Reactor Vessel Level Indication System (RVLIS) alarm was received in the Main Control Room. The operators immediately secured the N_2. Maximum pressurizer level reached 42 percent. Approximately 1400 gallons of water was transferred from the reactor vessel to the pressurizer.

The operators vented the reactor vessel head and RVLIS indication returned to 100 percent and pressurizer level returned to 30 percent.

The licensee concluded the cold leg loop stop isolation valve (LSIV) allowed the N_2 to enter the reactor vessel. The LSIVs are parallel disk gate valves designed to seat using differential pressure across the valve. During the N_2 pressurization, the differential pressure across the LSIV was reduced to minimal. Springs between the disks should allow for seating the valves, but the disks may shift as the differential pressure across the valve changes. The licensee believed with the N_2 applied, the disks were shifting back and forth from seating on the vessel side to the loop side. This shifting would allow N_2 to enter the vessel.

Disk pressurization was a process that used the Safety Injection (SI) Accumulators to apply pressure between the disks of the LSIVs. Disk pressurization would not always prevent leakage into the reactor vessel or the RCS loop, however, the amount of leakage could be measured by calculating frequency of filling the SI accumulators. The consequences of using disk pressurization included additional dissolved oxygen in the RCS, additional SI pump run time, additional dose required to install the system, and the possibility of adding larger amounts of water to the loops. The licensee originally planned not to use disk pressurization during the loop draining during B1P02. Following the leakage past the A LSIVs, the licensee applied disk pressurization to the LSIVs and reduced the leakage in the A loop.

The response taken by management was a strong positive attribute. The inspector observed careful critiques and conservative operations following the N_2 leak past the LSIV. Strong teamwork was observed with virtually every department present and actively participating.

1.2.2 RCS Pressure Control Transient

During the mid-cycle outage, panel 1PA06J was de-energized for modifications. 1PA06J provided electrical power for numerous circuits in the main control room, primarily indications on the main control board. When the panel was de-energized, the automatic portion of the controller for the centrifugal charging pump flow control valve, 1CV121, was de-energized and the valve went full open, allowing the full flow of the charging pump to enter the RCS, increasing the pressure. The operators did not realize that automatic valve control would be de-energized. The operator attempted to close the next two valves in-line with 1CV121 resulting in one valve shutting (1CV8106) and the second valve failing approximately one-half closed (1CV8105). 1PA06J work was stopped and panel re-energized. The operator regained control of 1CV121 by placing the valve in manual and the shut isolation valve, 1CV8106, was then opened. The licensee has initiated an investigation of the failure of 1CV8105.

The pre-work review of documents indicated 1CV121 would be de-energized but was not specific about whether indication, control power, or both would be lost. The licensee assumed that only indication would be lost, based on other components de-energized from 1PA06J. The inspectors considered this a personnel error prior to de-energization of 1PA06J in that the preparation of the de-energization list did not specify the effects of de-energizing 1PA06J on 1CV121. The inspectors also noted good performance by the shift operators in preparation for the de-energizing of 1PA06J. Operators closely monitoring panels were able to immediately identify the beginning of the transient and stop the transient after only a 20 psig increase in Reactor Coolant System pressure.

2.0 MAINTENANCE

NRC Inspection Procedures 62703 and 61726 were used to perform an inspection of maintenance and surveillance activities. Early maintenance activities during the mid-cycle outage appeared to be well performed. A rotated heat exchanger end bell on the Unit 2 A Safety Injection pump was identified during this period, however, the end bell was most likely installed incorrectly in 1993. Inadequate corrective action for a similar condition on the Unit 2 B Chemical and Volume Control Pump resulted in a Non-cited Violation (paragraph 2.1). Additionally, missed surveillances on 125 volt battery terminal connections resulted in a Non-cited Violation for inadequate procedures (paragraph 2.2).

2.1 Rotated Heat Exchanger End Bells

On October 11, the Unit 2 A Safety Injection pump oil cooler heat exchanger (HX) end bell was found rotated 90 degrees from the design orientation. Braidwood had notified Byron the previous day that Braidwood had discovered several HX end bells rotated 90 degrees. Initial inspection on October 10 by the Essential Service Water (SX) system engineer verified the SX pump oil coolers were properly configured. The engineer notified the other safety related pump system engineers by e-mail. The following morning, prior to the system engineers checking their systems, the NRC identified the Unit 2A Safety Injection (SI) pump oil cooler appeared to have an end bell rotated 90 degrees.

The safety related pumps have horizontal coolers orientated with oil supply and return lines (the shell side) vertical. The end bells contain two plugs for cathodic protection inserts. The plugs should be orientated vertical also. The 2A SI plugs were horizontal. The 2A SI cooler was the only safety related cooler identified with the rotated end bell. Investigation by the licensee indicated the end bell was most likely rotated during or before the most recent maintenance period for the cooler in June 1993.

The SI oil coolers are four pass coolers. The rotated end bell caused two of the four passes to be blocked, reducing the capacity of the cooler. Engineering calculations demonstrated the pump remained operable due to the design margin of the cooler. The pump bearings had not demonstrated elevated temperatures during surveillance runs, supporting the engineering calculation.

In addition, the licensee identified the potential for the Condensate/Condensate Booster (CD/CB) pumps to have an oil cooler end bell problem. The coolers on the CD/CB are not all oriented the same direction and the orientation of the end bell is not as easily identified. During work periods on each pump, the end bells will be removed and verified. The licensee suspects that some of the end bells are rotated.

During September 1993, the licensee identified the Unit 2B Chemical and Volume Control (CV) pump gear box oil cooler had an end bell rotated 45 degrees. The 2B CV pump had significant run time with no overheating problems noted. The licensee identified the issue as an inadequate procedure and committed to provide more effective cooler disassembly/reassembly instructions. After the commitment to revise procedures was made, the procedure used for the CV pump work was reviewed and identified to contain adequate instructions. No additional corrective action was performed as a result of the CV pump end bell rotation issue. Inspection of the remaining safety related pumps would have revealed the rotated end bell on the 2A SI pump.

The licensee identified the inadequate corrective action taken for the 2B CV pump and inspected all the safety related pumps, along with non-safety related pumps with similar configurations, after the identification of the 2A SI pump issue. The inspector concluded the safety significance to be minor due to both engineering calculations and operating experience supporting operability. This licensee-identified and corrected violation for inadequate corrective action is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy.

2.2 <u>125 Volt Battery Missed Surveillance</u>

On September 28, 1995, the licensee discovered the Electrical Maintenance Surveillances 1/2 BHS 4.8.2.1.2.C-1 "125 Volt Battery Bank 18 Month Surveillance," did not include a step for measuring the resistance of the terminal connection on the positive post for cell 1 or of the terminal connection on the negative post for cell 58. The surveillance was written to fulfill the Technical Specification requirements 4.8.2.1.2c.1), 2), and 3). The requirement 4.8.2.1.2.c.3) states that at least once per 18 months each 125 volt battery bank and its associated charger shall be demonstrated to be Operable by verifying that: "3) The resistance of each cell to cell and terminal connections is less than or equal to 150 E(-6) ohm."

The licensee review identified the engineering surveillances 1BVS 8.2.1.2.d-1, "Unit 1 125 Volt Battery Bank 111 (and d-2 for division 112) Service Test" and 2BVS 8.2.1.2.d-1, "Unit 2 125 Volt Battery Bank 211 (and d-2 for division 212) Service Test" measured the resistance of the terminal connections at cells 1 and 58, but did not have an acceptance criteria. The measurement of terminal resistances was added to the Surveillances in Revision 6 dated March 2, 1988 for Unit 1 and Revision 2 dated December 28, 1987 for Unit 2. The surveillances measure the cell to cell and terminal resistances except for the two identified terminal connections. The performance tests completed on the battery banks have not indicated a problem with any of the four banks.

This licensee-identified and corrected violation for inadequate procedures is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy.

3.0 ENGINEERING

NRC Inspection Procedure 37551 was used to perform an on-site inspection of the engineering function. Several issues required engineering support this period. Proactive operations support was demonstrated by identification of a degrading feedwater flow venturi transmitter. A violation was identified for failure to have a program to identify and quantify leakage from systems that could contain highly radioactive fluids outside the containment after an accident (paragraph 3.1).

3.1 Reactor Coolant Outside Containment Leak Testing

Technical Specification 6.8.4.a.2 required an integrated leak test of portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The leak test was to be conducted at refueling cycle intervals or less. The inspectors identified the leak test did not exist.

The integrated leak test was part of the post Three Mile Island actions discussed in the Byron/Braidwood Updated Final Safety Analysis Report (UFSAR). In Appendix E of the UFSAR, the licensee committed to monitor the leak testing of piping so that the appropriate lines are examined at the required intervals. Table 15.6-13 provided the "Maximum Recirculation Loop Leakage External to Containment" values assumed during Loss of Coolant Accident analysis. The values listed were from Emergency Core Cooling System (ECCS) loops only. Total leakage from outside containment from ECCS recirculation loops was assumed to be less than 3910 cc/hr.

The licensee was performing the visual inspections, VT-2, as required in the American Society of Mechanical Engineers (ASME) Code, Section XI. The VT-2 inspection was described as "Other Leak Testing" in the UFSAR following the integrated leak rate program. The ASME inspections were similar in scope to the integrated leak rate determination described in TS 6.8.4.a, however, the ASME inspections were limited to Class 2 and Class 3 systems as described in ASME Section XI. Therefore, all potential leakage points were not included in the VT-2 inspections.

The inspectors concluded that while a leak rate test of potentially highly radioactive systems outside containment did not exist, the safety significance was minor. The VT-2 inspections covered essentially all of the subsystems required in TS 6.8.4 and the UFSAR. Actual identified leakage was virtually zero in the sampling of VT-2 inspections reviewed by the inspector. The most significant was wet boric acid with the majority of the indications being dry boric acid, indicating inactive leaks. The failure to have an integrated leak test of potentially highly radioactive system outside containment was considered a violation of technical specifications (50-454/455-95009-01(DRP)).

3.2 Feedwater Flow Transmitter Drift Identified

System Engineering Department (SED) identified a minor feedwater flow venturi discrepancy on Unit 2 during the period. Each of the four feedwater venturi tubes contained two transmitters. SED routinely tracked the performance of the transmitters by comparing the two transmitters for each venturi. This method identified a transmitter that started to drift from the calibration setting. The instrument drift was not visible to the operators. The transmitter was replaced. The inspectors considered the identification a demonstration of strong proactive plant operation.

3.3 "Boraflex" in the Fuel Storage Rack

The Boraflex on the fuel storage racks for the spent fuel storage pool has been noted to degrade under gamma irradiation and long term exposure to the spent fuel pool environment. As the boraflex degrades, the polymer matrix degrades and silica filler and boron carbide are released into the spent fuel pool (SFP). Presence of silica in the SFP water samples was an indication of depletion of the boron carbide, the neutron absorber, in the racks. The licensee has found silica in SFP water and has reduced silica to less than 3 ppm during refueling when the SFP was connected to the reactor cavity. After receipt of NRC Information Notice 95-38, the licensee has developed plans including not reducing the silica in the SFP water and using several methods to minimize the amount of SFP water containing silica from mixing with reactor cavity water. The irradiation of the Boraflex was reduced by planning the storage according to the out-of-core time for each fuel element. The licensee was reviewing the necessary actions and the system capacities to maintain the pool at a reduced temperature, which also increases the useful lifetime of the boraflex panels.

3.4 Diesel Generator Jacket Water Cooling Standpipe

The diesel generator jacket water standpipe volume was determined to be less than described in the UFSAR by the licensee. The standpipe served two purposes: accommodate thermal expansion during heat up and provide a reservoir of make up water. System Engineering performed an operability assessment and found the EDGs operable. The initial evaluation did not state that the current standpipe volume was sufficient by itself. However, the evaluation did state "alternate means" to supply make up water were available. A temporary procedure change discussing alternate methods of make up water was referenced in the operability assessment. This issue is an Inspection Follow-up Item (50-454/455-95009-02(DRP).

3.5 Refueling Water Storage Tank Switchover Calculations

During a safety injection, suction for the Emergency Core Cooling System (ECCS) pumps was designed to switch from the Refueling Water Storage Tank (RWST) to the containment sump prior to completely draining the RWST. The switchover occurred at 46 percent level in the RWST. Original analysis by the licensee assumed the entire volume of the RWST was in the containment sump prior to switchover to the sump. Initial calculations bounded the concerns, including adequate net positive suction head for the ECCS pumps and a slightly elevated water temperature for core cooling earlier in an accident sequence. Detailed calculations were being performed at the close of this inspection period. This issue is an Inspection Follow-up Item (50-454/455-95009-03(DRP).

4.0 **PLANT SUPPORT**

NRC Inspection Procedure 71750 was used to perform an inspection of Plant Support Activities. Fire protection performance was good with the exception of certain impaired fire barrier procedures not being followed. A violation was identified for a failure to authorize an impairment for fire doors blocked open to support flushing activities (paragraph 4.1.3). Radiation Protection prepared for the mid-cycle outage well and performed early outage activities without incident. The Emergency Protection group demonstrated very good performance during the pre-exercise drill, the drill was well thought out, and the performance of the personnel was considered to be very good.

4.1 Fire Protection

During this inspection period, a special inspection was performed in the area of Fire Protection. Selected portions of NRC inspection procedure 64704 were utilized.

4.1.1 (Closed) Violation (50-454/94004-02; 50-455/94004-02)

This violation pertained to the failure to have a transient combustible authorization issued for approximately 30 gallons of oil which was left unattended in the auxiliary building. Adequate corrective actions appeared to have been taken to resolve this problem. No new problems with transient combustibles were noted during this inspection.

4.1.2 Observation of Plant Areas

The inspector toured the areas of the auxiliary and turbine buildings and the screen house to observe the adequacy and control of combustibles, fire doors, hose stations, suppression and detection equipment, extinguishers, emergency lights, and housekeeping.

The overall material condition of fire protection equipment was adequate. The material condition of the fire suppression and detection equipment was satisfactory, except for fire main valves with packing leaks. The diesel and electric fire pumps appeared to be in good condition during the walkdown. The diesel fire pump maintenance history indicated that very few problems occurred with this pump during the past three years. Fire brigade equipment was in good condition and was stored in locked cages in convenient locations in the plant. Fire hoses and stations were in good condition. All observed emergency lights were operable and were correctly aimed for safe shutdown paths in the plant. However, the licensee did not know the actual availability of emergency lights and was performing a failure rate study to assess emergency lighting problems.

The number of impaired doors in the plant was a concern as the licensee had been dealing with a large number of impaired doors since 1994. A team had been assembled to study the various solutions required to keep most doors operable. The licensee had completed the majority of that study. Impairment data indicated the licensee was making progress in reducing the backlog of impaired doors. However, during the plant tour door 170 (turbine building common door separating the two units) was noted as not shutting and no impairment had been issued. Numerous plant staff passed through the door without noticing the lack of an impairment on this door. In addition, quality assurance personnel had identified this door as impaired the previous day without an impairment issued at the time of the tour. However, roving fire watches were observed making plant tours of the area because of other fire protection impairments.

The control of normal combustibles was good with very few transient combustibles in the plant. Storage cages in the plant contained a limited amount of combustibles. Flammable liquids were stored in fire proof cabinets and in appropriate safety cans. However, the number of oil leaks on rotating equipment was a concern. This increased the plant's fire hazard and resulted in a long-term transient combustible with the oil from leaks being drained into drums. One compressed cylinder of hydrogen was noted as not properly stored. It was located in a helium storage rack and was not properly fastened to the rack. The licensee removed the hydrogen cylinder during the inspection. The low number of fires in the plant during the past three years was an indication of good transient combustible control and effective control of hot work.

4.1.3 Impairments

The inspectors identified that the number of fire protection impairments was high during a review of an impairment list. However, a subsequent licensee review of these impairments showed that many of them had been repaired but the impairment list had not been updated. The licensee determined that plant staff had not returned the impairment sheets so that those items could be closed. The licensee was taking corrective actions during the inspection to resolve this problem. The actual number of impairments not counting thermo-lag and bored holes in fire barriers to reroute cables to resolve thermo-lag concerns was acceptable. Contributing to this improvement was a maintenance person who had assumed ownership of fire door repairs with the number of impaired doors rapidly decreasing. In addition, the licensee was committing more resources to further reduce the number of fire protection impairments.

On September 18, the inspector observed flushing of the 1A/2A essential service water floor drains. To perform the evolution, the licensee blocked open a fire door on one level of the auxiliary building to perform work on another level, also with a blocked open fire door. No plant barrier impairment permit (PBI) was obtained.

When this issue was discussed with the fire marshal, the marshal stated that since the workers were "in the area," a PBI was not required. The fire marshal considered the workers to be in the area although the fire marshal understood that the location of the work was not within visual range of the impaired fire door which was on a different level from where the work was performed. In addition, Byron Administrative Procedure, BAP 1100-3, "Fire Protection Systems, Fire Rated Assemblies, Ventilation Seals, and Flood Seal Impairments," does not consider personnel in the area as a basis for not issuing a PBI.

BAP 1100-3 states, in part, that: "C.1. A Barrier/Fire Protection Systems Impairment Permit, BAP 1100-3T1, is required for all fire protection equipment, fire detection instrumentation, fire rate assemblies, ventilation seals, and flood seals which are impaired. F.1. The following steps should be completed before a Barrier/Fire Protection System is taken Out-of-Service, impaired, or otherwise rendered inoperable. F.1.b. A Barrier/Fire Protection Systems Impairment Permit, BAP 1100-3T1, shall be initiated by the department in charge of the work." Contrary to these requirements the doors discussed above were impaired and returned to service without a Barrier/Fire Protection Systems Impairment Permit being initiated. This was considered a violation of station procedures and technical specifications (50-454/455-95009-04(DRS)).

4.1.4 Fire Brigade

The plant requirements for the fire brigade were all being met in an effective manner. The inspector reviewed fire brigade qualifications and associated training records. Onsite fire drill requirements had been met by all brigade members who were listed as qualified. The fire brigade training program appeared to be good.

The inspectors observed a fire drill in a chemical storage area. The brigade members responded in a timely manner. The brigade leader made a quick assessment and took control of the team. The response time of the brigade was very good. There was excellent involvement from numerous other plant staff members in the brigade drill. There was good communication between the fire brigade leader and brigade members and the control room. Chemical risks were appropriately evaluated. The overall assessment of the drill was excellent.

A critique was held at the end of the fire brigade drill with all of the participants in the drill present. The participants were allowed to give their insights on what they considered as problems during the drill. The staff's overall assessment was that this was an excellent drill with very few problems. The only problem noted was that drill critiques contained very few documented problems. With a more detailed documentation of fire brigade problems this information can be passed on during training session to other brigade teams.

4.1.5 <u>10 CFR 50.59 Safety Evaluations</u>

The inspector reviewed 10 CFR 50.59 safety evaluations issued to assess changes made to the fire protection program. All of the changes were appropriate and none appeared to be detrimental to fire protection safety.

4.1.6 <u>Audits and Field Monitoring Reports (FMRs)</u>

Audit investigations for fire protection were detailed and thorough with adequate staff hours devoted to each audit. The FMRs were performance based observations of conditions in the plant and were effective in identifying problems in the fire protection program. The licensee had taken timely corrective actions for those fire protection deficiencies found during the audits.

4.2 <u>Security & Safeguards</u>

The inspectors noted satisfactory performance of routine items including proper display of photo-identification badges by station personnel, verification vital areas were locked and alarmed, and personnel and packages entering the protected area were adequately searched by appropriate equipment or by hand.

4.3 <u>Emergency Planning</u>

During this inspection period a pre-exercise drill was conducted. The inspector observed portions of the pre-exercise and concluded the pre-exercise offered positive training results.

5.0 PERSONS CONTACTED AND MANAGEMENT MEETINGS

The inspectors contacted various licensee operations, maintenance, engineering, and plant support personnel throughout the inspection period. Senior personnel are listed below.

At the conclusion of the fire protection portion of the inspection on September 21, 1995, (denoted by #) and at the conclusion of the inspection on November 6, 1995, (denoted by *) the inspectors met with licensee representatives and summarized the scope and findings of the inspection activities and discussed the likely content of the report. The licensee did not identify any of the documents or processes reviewed by the inspectors as proprietary.

K. Graesser, Site Vice President
#*K. Kofron, Station Manager
#*D. Wozniak, Site Engineering Manager
*T. Gierich, Operations Manager
P. Johnson, Technical Service Superintendent
#*E. Campbell, Maintenance Superintendent
#*M. Snow, Work Control Superintendent
#*D. Brindle, Regulatory Assurance Supervisor
#*A. Javorik, Technical Staff Supervisor
K. Passmore, Station Support & Engineering Supervisor
P. Donavin, Site Engineering Mod Design Supervisor
T. Schuster, Site Quality Verification Director
#*R. Colglazier, NRC Coordinator
*R. Wegner, Shift Operations Supervisor
W. Kouba, Long Range Work Control Superintendent
R. Scheidecker, Fire Marshal
D. Popkins, Operation Admin. Engineer
J. Michmershuizen, Assistant Fire Marshal
D. Sanders, TSS Fire Prevention Site Manager
F. Pallak, Fire Prevention System Engineer
*D. Shaw, Shift Control Room Engineer
*L. Bunner, Shift Engineer