U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-456/92023(DRP); 50-457/92023(DRP)

Docket Nos. 50-456; 50-457 Licenses No. NPF-72; NPF-77

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

Facility Name: Braidwood Station, Units 1 and 2

Inspection At: Braidwood Site, Braidwood, Illinois

Inspection Conducted: October 13 through November 30, 1992

Inspectors: S. G. Du Pont
J. R. Roton
G. M. Hausman

Approved By: M. Farber, Chief
Reactor Projects Section 1A

12/17/92
Date

Inspection Summary

Inspection from October 13 through November 30, 1992 (Reports No. 50-456/92023(DRP); 50-457/92023(DRP))
Areas Inspected: Routine, unannounced safety inspection by the resident and regional inspectors of licensee action on previously identified items; licensee event report review; outages; radiation protection; operational safety verification; monthly surveillance observation; and report review.
Results: Three violations were identified in one of the six areas inspected. In the remaining areas, no violations were identified.

The following is a summary of the licensee's performance during this inspection period:

Plant Operations

The licensee's performance in this area for this inspection period was good. Shift briefings continued to provide sufficient information for planned evolutions to be performed during the shift. The inspectors have raised several questions involving operability determinations associated with the Main Steam Line Code Safety Valves.

Radiological Controls

Three violations were issued due to the licensee's failure to adequately control the addition of SF_6 to the steam generators (two violations) and the failure to adhere to the posting requirements of Radiologically Controlled Areas. Additionally, the report discusses activities associated with safety evaluations for the SF_6 and a chloride excursion. One was an example of good efforts producing a detailed evaluation and the other was an example of a failure to perform an evaluation.

Safety Assessment/Quality Verification

The one LER reviewed during this inspection period appears to have appropriate corrective actions to preclude similar events. The licensee's evaluation of the Unit 1 chloride excursion is a good example of a detailed and comprehensive safety assessment. However, the failure to conduct a similarly comprehensive evaluation for the sulfur hexafluoride addition indicates that the sensitivity to and understanding of the need for safety assessments is not uniform throughout the licensee's organization.

Engineering and Technical Support

Due to the inspectors limited review in this area, the licensee's performance was not assessed for this inspection period.

Maintenance and Surveillance

The licensee's performance in maintenance and surveillance during this inspection period was good.

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

*K. L. Kofron, Station Manager
G. R. Masters, Project Manager
G. E. Groth, Production Superintendent
D. E. O'Brien, Technical Superintendent
D. E. Cooper, Assistant Superintendent - Operations
R. J. Legner, Services Director
A. D Antonio, Nuclear Quality Program Superintendent
*R. Byers, Assistant Superintendent Work Planning
G. Vanderheyden, Technical Staff Supervisor
S. Roth, Security Administrator
K. G. Bartes, Nuclear Safety Supervisor
A. Haeger, Regulatory Assurance Supervisor
*J. Lewand, Regulatory Assurance
S. Hunsader, EQ Supervisor Design Support - Nuclear Engineering
K. C. Radke, Technical Staff System Engineer

*Denotes those attending the exit interview conducted on November 30, 1992.

The inspectors also interviewed several other licensee employees.

2. Licensee Action on Previously Identified Items (92701, 92702)

a. Open Item

(Closed 50-456/92017-02(DRP); 50-457/92017-02(DRP): **Failure to Follow Posting Requirements of a Radiologically Controlled Area.** Inspection Report 92017 details the failure of a Radiological Protection Technician (RPT) to adhere to the posting requirement to conduct a whole body frisk prior to exiting a radiologically controlled area (RCA). In their followup review, the inspectors discovered that two weeks prior to this incident, a RPT had failed to verify the decontamination of the 1A letdown heat exchanger room before removing the posting. As a result, the RPT and one Electrical Maintenance Department person were contaminated when they entered the room to replace light bulbs. Additionally, there has been one other incident since the open item was identified. In this incident, two Mechanical Maintenance Department personnel failed to adhere to the posted requirements for entry into a RCA and were subsequently contaminated. These failures to adhere to the posted requirements for conducting work within a RCA are violations of Braidwood Technical Specification 6.11, "Radiation Protection Program," as detailed in Braidwood Radiation Protection Procedure 1110-3, "Radiological Postings, Labels, and Controls," (50-456/92023-01(DRP); 50-457/92023-01(DRP)).

<u>(Closed) Open Item (456/88010-01(DRS);457/88011-01(DRS))</u>: **Adequacy of fire protection for several unprotected structural steel columns and auxiliary steel attachments.** The columns were located in the fuel building between V and W at coordinates 17, 18, and 19, and the attachments were in the auxiliary building on column P-21 at elevation 401'-0". The licensee was to provide the methodology used for selection and identification of the fire protected structural steel components and the technical justification that column P-21 met the specified fire rating.

For the columns, the licensee stated that because of the low fire loading and the large open volume of the area, a credible fire would pose no hazard to the structural steel columns, therefore, fire proofing of the columns was not required. Justification was provided in Sargent & Lundy Engineers letter dated May 20, 1988. The columns support the slab at elevation 451' 0", a portion of which carries a fire rating. The calculated fire loading for the area, which includes an allowance for transient combustibles, is 5000 Btu/ft^2 (Fire Protection Report, Subsection 2.3.12.1). This equates to a fire severity of under four minutes duration (NFPA Fire Protection Handbook, Chapter 9, Section 7). Therefore, a credible fire would pose no hazard to the structural steel columns.

For the auxiliary steel attachments, the licensee stated that the additional heat transfer into the fire protected column from the unprotected auxiliary steel attachments did not degrade the fire rating for column P-21 below the specified three hour rating. Justification was supplied in Sargent & Lundy Engineers letter dated May 20, 1988. The column was protected by a fire-proof material, Pyrocrete 102 (7/8" thick), in accordance with applicable installation drawings, which designated a three hour fire rating according to Underwriters Laboratory (UL) Detail X-719. The UL rating was based on tests conducted on a W10x49 column. P-21 was a W14x342 column, which had a cross section seven times as massive as the UL tested column. The American Iron and Steel Institute (AISI) had performed extensive research and tests on a wide range of column sizes including sections which were more massive than the UL tested W10x49 column. These tests were summarized in AISI publication "Design Fire Protection for Steel Columns," Third Edition, March 1980, which indicated that the effective fire rating of the W14x342 column was more than twice that for the W10x49 column. Therefore, ample margin was provided to compensate for the additional heat input from a potential fire due to the unprotected auxiliary steel attachments.

Based upon the above, the inspectors concluded that the methodology and technical justification provided were acceptable and the inspectors had no further concerns. This item is closed.

b. Unresolved Items

(Closed) Unresolved Item (456/90019-02(DRS): Physical separation between fuel oil overflow, supply, and vent lines associated with the opposite train emergency diesels. The licensee conducted a detailed review and analysis that determined the installed piping arrangement, although, not consistent with the configuration described in the fire hazard analysis (FHA), posed no immediate operational concerns for the opposite train diesel. The FHA stated that all equipment, cables, and piping in the diesel generator room, the diesel oil day tank room, and the diesel oil storage tank room would be associated with only one ESF division. However, physical inspections revealed that some of the diesel oil piping on each train was routed through the three rooms of the opposite train.

A detailed review of the Byron/Braidwood Fire Protection Report (FPR) was made to identify all other inconsistences related to physical separation of safety related equipment, cables, and piping. This consisted of a zone-by-zone review of the FHA (Section 2.3 of the FPR), a review of the safe shutdown analysis (Section 2.4 of the FPR), and a review of FPR Section 3.0, which addressed conformance to the Standard Review Plan. No other inconsistences from the FPR were identified. Minor changes were completed to address the inconsistences as described in Sargent & Lundy Engineers Letter dated November 29, 1990, and Transmittal DIT-BB-EXT-0124, "Assessment of Diesel Oil Piping Routed in Opposite Train Diesel Generator Rooms" dated February 25, 1992.

The inspectors reviewed the licensee's detailed analysis and discussed the results with NRR. The discussion concluded that the licensee's analysis was acceptable and that Appendix R concerns were adequately addressed, since offsite power would be available and the diesel oil piping would remain intact during a postulated fire. Corrective actions taken by the licensee indicated prompt actions were performed including an expanded scope of review and analysis which included the Byron Station, as-well-as, evaluating the probability of missile affects on the opposite train diesel piping. The inspectors noted, however, that during the performance of two minor changes numerous field problem reports (FPRs) were generated relating to interference and clearance problems indicating that planning was not effective. The inspectors had no further concerns and considered this item closed.

c. Violation

(Closed) 50-456/92017-03(DRP); 50-457/92017-03(DRP): Technical Specification 6.8.1 was violated when the licensee failed to convert Nuclear Work Requests to Temporary Alterations in accordance with Braidwood Administrative Procedure 2321-18. The licensee's response to this violation was prompt and thorough.

Actions taken appear adequate to preclude recurrence of this or similar events. This item is closed.

One violation was identified.

3. Licensee Event Report (LER) Review (92700)

LERs were reviewed and closed based on the following criteria:

- Reportability requirements were met.
- Immediate corrective actions were accomplished.
- Corrective actions to prevent recurrence has been or will be initiated per technical specifications.

No violations or deviations were identified.

(Closed) 457/92006: **Reactor Trip Due to Valve Mispositioning.** This event is discussed, in detail, in Inspection Report 50-456/92020; 50-457/92020, Paragraph 4. In addition to those corrective actions previously identified, the inspector also notes the involvement of GN Venture contractor personnel in the assessment of root cause determination and corrective action. This item is closed.

4. Outages (86700)

No violations or deviations were identified.

On November 3, 1992, Unit 1 Main Generator was synchronized to the grid, ending the licensee's planned 66-day refueling outage seven days ahead of schedule. In addition to completing ahead of schedule, the 185.0 Rem of exposure and 88 personnel contaminations were well under the ALARA goals of 208.5 Rem and 126 personnel contaminations established for the outage. From a budgetary standpoint, initial estimates show the outage to be $1.8 million under the approved budget. The outage was accomplished without major incidents and difficulties. The lessons-learned must be carried forward into the refueling outage for Unit 2, which begins in March 1993.

5. Radiation Protection (93702)

Two violations were identified pertaining to radiological work practices, performing safety analysis, and preparing written procedures for chemistry related evolutions.

- Addition of sulfur hexafluoride (SF_6).
- Evaluation of planning and implementation (SF_6).
- Inspectors conclusion (SF_6).
- Apparent violations (SF_6).
- November 6, 1992, chloride excursion.
- Chloride excursion safety evaluation.

Addition of sulfur hexafluoride (SF_6) causes unexpected variations in steam generator chemistry. On November 6, 1992 the Braidwood Chemistry Department inadvertently caused a chemical variation on the Unit 1 steam

generators. Technicians injected a large amount, about 15 standard cubic feet, of sulfur hexafluoride (SF_6) gas into the condensate system. The injection of SF_6 was part of troubleshooting efforts on Unit 2 steam generators. Probable leakage in the Unit 2 steam generators resulted in unexpected chemistry levels. Since the condensate system contains several connections between the units, the licensee suspected that these connections were the leakage source.

Shortly after injecting SF_6, the technicians noticed unexpected responses in the Unit 1 steam generators' chemistry. A chemistry sample confirmed that a large variation occurred. This required entry into the action level of the station chemistry procedures. The sample displayed elevated cation conductivity, phosphates, fluorine, and sulfonates. The technicians determined that the SF_6 gas unexpectedly broke down into sulfonates and fluorine.

The inspectors evaluated the planning and implementation of the troubleshooting efforts involving injection of SF_6. Previously, SF_6 injection was used to find leaks in the condenser water boxes. The success of this usage influenced the chemistry department to use SF_6 in the condensate.

However, they did not evaluate the possibility of SF_6 going in the steam generators. They also did not evaluate the effects of the steam generator water chemistry on SF_6. SF_6 gas is only slightly soluble in water and soluble in alkaline solutions. Since water chemistry is alkaline, SF_6 broke down into fluorine and sulfonates. This condition is not desirable since fluorine is corrosive to the heat transfer surfaces and sulfonates is basic. The technicians injected SF_6 into the condensate system without considering these effects. They also injected the SF_6 without a procedure.

The inspectors concluded that the chemistry department did not follow several procedures and requirements before and during the injection of the SF_6 gas. 10 CFR 50.59, "Changes, Tests, and Experiments," requires performance of an evaluation of tests or experiments not described in the safety analysis report. The regulation also requires the evaluation for changes in the facility. This evaluation is to determine the possibility of an unreviewed safety question or a change in the technical specification.

The Braidwood safety analysis report describes the methods for maintaining water chemistry in the steam generators. The addition of ammonia in the form of ammonium hydroxide, or an equivalent amine, and hydrozine to the condensate is in Section 10.3.5.1. The addition of SF_6 gas to the condensate is not in the safety analysis report.

The regulation 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be done by documented procedures. These procedures shall be of a type appropriate to the circumstances of the activity and followed. Additions of chemicals to the condensate affects the steam generator and can affect the quality of the heat transfer surface.

The Braidwood Technical Specification, Section 6, "Administrative Controls," requires the establishing of procedures for activities prescribed in Regulatory Guide 1.33, Quality Assurance Program Requirements. The regulatory guide requires the establishing of procedures for controlling water quality. These procedures will contain limits for concentrations of agents that are corrosive to heat transfer surfaces. The addition of SF_6 into the condensate was performed without establishing a procedure.

The inspectors concluded that the following activities were violations of NRC requirements. The injection of SF_6 without a procedure was an apparent violation of Technical Specifications (50-456/92023-02(DRP); 50-457/92023-02(DRP)). The failure to perform an evaluation for unreviewed safety condition is an apparent violation of 10 CFR 50.59 (50-456/92023-03(DRP); 50-457/92023-03(DRP)).

The inspectors reviewed the activities associated to the September 6, 1992, Unit 1 Steam Generator chemistry variation (chloride excursion). The 10 CFR 50.59 Safety Evaluation of Unit 1 chloride excursion concludes there is no unreviewed safety question. Inspection Report 50-456/92020; 50-457/92020, details the chloride excursion experienced by Unit 1 during its shutdown to commence a refueling outage. As required by Technical Specification 3.4.7, a Safety Evaluation was completed which addressed the potential effect of this excursion on the Reactor Coolant System (RCS) austenitic stainless steel, Alloy 600 materials, and Zircaloy fuel cladding.

The 10 CFR 50.59 safety evaluation was completed by Westinghouse and concluded there was no unreviewed safety question resulting from the excursion. The technical basis for this conclusion was:

a. Austenitic stainless steels are potentially susceptible to chloride stress corrosion cracking (SCC) in aqueous solutions under certain conditions. Oxygen is necessary for chloride cracking in austenitic steels at temperatures below boiling. No detectable oxygen was reported in the Unit 1 RCS during the excursion. Hydrogen peroxide was not added to the RCS during the excursion. Moreover, of the 47-hour duration of the excursion, the RCS average temperature was above 150°F (the acceleration temperature for chloride SCC) for only 35-hours. Existing Westinghouse test data for sensitized 304 stainless steel indicates that the crack initiation time for chloride concentrations of 390 ppb (the peak concentration seen during the excursion) is on the order of 12-13 months, well beyond the 35-hour exposure for Unit 1. The test data was conservatively based on exposure in a fully aerated chloride solution of 150°F. As previously stated, no detectable oxygen was reported in the RCS during the excursion. Therefore, under the conditions for Unit 1, it was judged that the elevated chloride concentration would not have a negative impact on the performance of the austenitic stainless steel present in the RCS.

b. Alloy 600 materials have generally exhibited good resistance to chloride induced cracking. Existing test data has demonstrated

that chloride induced cracking is not an issue for Alloy 600 material exposed to the type of environment found in the Unit 1 RCS during the excursion. Alloy 600 c-ring test specimens stressed to 2/3 of yield strength, have been exposed to fully aerated chloride solutions with different chloride concentrations (100-1000 ppm) at 150°F for periods of up to 12 months with no incidents of cracking. The test data far exceeds the chloride level and exposure period that the Alloy 600 materials were exposed to during the chloride excursion. Therefore, future performance of Alloy 600 material will not be adversely affected because of the excursion.

c. Westinghouse Zircaloy-4 material specifications limit the chloride content to 20 ppm maximum, but certifications for the material typically report that values are less than 10 ppm. During reactor operation, the protective oxide that is formed on the Zircaloy-4 fuel components is considered to contain, due to diffusion effects, similar impurity levels to the base material. Assuming therefore, that the oxide film on the Unit 1 Zircaloy-4 fuel components contained approximately 10 ppm chloride prior to the coolant chloride excursion, and that all of the 390 ppb chloride from the coolant was absorbed by the oxide lm, the total resultant chloride content of the oxide film would show an increase of less than 1 ppm. The resulting chloride content would still be considerably below the material specification limit of 10 ppm maximum. Additionally, corrosion studies performed on Zirconium, where Zirconium at 650°F was placed in water containing 200 ppm chlorine gas, have shown Zirconium's corrosion resistance to be much less sensitive to impurities in the water than to those in the metal. Thus, no adverse effects on the Zircaloy fuel cladding, fuel integrity and fuel handling operations are expected.

6. Operational Safety Verification (71707)

The inspectors verified that the facility was being operated in conformance with the licenses and regulatory requirements and that the licensee's management control system was effectively carrying out its responsibilities for safe operation. No violations or deviations were identified.

- Untested code safety valves.
- Inspectors' concerns.
- Inspectors' review of Technical specifications.
- Determination of having proper lift setpoints.
- Unit 1 outage to investigate generator cooling system.
- Westinghouse recommendations for generator cooling syst.
- Unit 1 return to 100% reactor power.

Untested Main Steam Line Code Safety Valves raise questions regarding the ability of Unit 1 to proceed with mode change to Mode 3. On October 29, 1992, Unit 1 entered Mode 3 following completion of its refueling outage. During the outage, five Main Steam Line (MSL) Code Safety valves were modified or repaired. The Restart Onsite Review

identified the need to test the lift setpoint pressures of these valves at nominal operating pressure (NOP) and temperature (NOT), while in Mode 3, to close the Unit 1 outstanding Nuclear Work Requests (NWRs).

In completing the checklist required before entry into Mode 3, the Unit 1 Supervisor noted that four of the MSL Code Safety valves were located on the same MSL. Based on the Unit Supervisor's interpretation of Technical Specification (TS) 3.7.1.1, "Safety Valves," it was questioned whether the mode change could be made with four untested (inoperable) code safeties on one main steam line. Table 3.7-1 of the TS only provides direction for continued power operation with a maximum of three inoperable code safety valves.

Through a series of conference calls, the licensee determined "a review of NWRs associated with these valves provides a reasonable assurance of proper lift setpoint." Therefore, the mode change could occur since the valves, although not tested, were not inoperable.

On October 30, 1992, with Unit 1 at NOP and NOT in Mode 3, the five MSL Code Safety valves were tested per Braidwood TS Procedure BwVS 7.1.1-1, "Main Steam Safety Valves Operability Test." All five valves failed the surveillance, were adjusted and retested satisfactorily.

In reviewing this event, the inspectors raised the following questions/concerns:

a. Was the interpretation of the TS correct?

Based on the licensee's interpretation, the inspectors then questioned:

b. How was the conclusion that the untested MSL Code Safety valves had a "reasonable assurance" of having proper lift setpoints determined?

c. Why were the MSL Code Safety valves not considered inoperable and an operability determination conducted per Braidwood Administrative Procedure BwAP 330-10, "Operability Determination of Safety-Related Equipment?"

In reviewing TS 3.7.1.1, the inspectors determined that the provisions of TS 3.0.4 applied and the Mode 3 change with four untested/inoperable code safety valves on the same MSL was not prohibited. Although the TS applies to Modes 1, 2, and 3, it is based on the operability of the MSL Code Safety valves to ensure that secondary coolant system will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most sever anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% rater thermal power coincident with an assumed loss of condenser heat sink (i.e., no steam dumps to the condenser). Therefore, since the ability to provide relieving capacity is not an issue in Mode 3, the mode change could be made with all five MSL Code Safety valves, on any single steam line, inoperable. To allow entry into Mode 3, where the code safety valves can be tested and set, the provisions of TS 3.0.4 (the mode change provision) are applicable to TS 3.7.1.1.

The determination that the MSL Code Safety valves had a reasonable assurance of having proper lift setpoints was, in the inspectors opinion, weak. While a review of the NWRs showed the valves were assembled correctly, they had not been bench tested. Additionally, historical data and the engineering judgment of the Technical Staff and Engineering and Nuclear Construction staff did not support the conclusion reached. The inspectors questioned how the available information and engineering judgment was weighed in reaching a determination of "reasonable assurance."

Regarding the issue of operability, the inspectors feel there can be no question that the valves were clearly inoperable and required an operability determination made per BwAP 330-10.

The inspectors have discussed these questions and concerns with the licensee and will follow their resolution closely.

Unit 1 enters planned forced outage to investigate elevated Delta T on the Generator Cooling system. On November 20, 1992, Unit 1 entered Mode 3 to commence a planned forced outage. The outage was required to investigate and repair the cause for the elevated temperature differential (Delta T) of approximately 10°C between the stator-rotor cooling water inlet and outlet temperatures.

A review of the refueling outage activities completed on the tator-rotor cooling system did not identify a possible cause of the high Delta T. The licensee performed a fiber-optic inspection of the inlet manifold, took flow measurements of individual coils while conducting a reverse flow flush on the stator, and inspected the coil hoses for blockage. These efforts failed to identify a root cause for the condition.

Based on Westinghouse recommendations, the alarm setpoints were raised to 14°C for high Delta-T and 16°C for high-high Delta-T. Additionally, Westinghouse performed a balance of the #5 turbine generator bearing. The balancing successfully reduced the vibration form approximately 5.9 mils to approximately 1.1 mils. All other bearing vibrations are less than 2.1 mils. The licensee also repaired various steam, water, and oil leaks which had developed since the unit was returned to service following the refueling outage.

On November 24, 1992, the unit entered Mode 1 and is currently at 100% reactor power. The licensee and the inspectors will continue to monitor the cooling system performance. the inspectors will evaluate the licensee's corrective actions during a subsequent inspection based on the system's continued performance.

7. <u>Monthly Surveillance Observation (61726)</u>

The inspectors observed several of the surveillance testing required by technical specifications during the inspection period and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that results conformed with technical

specifications and procedure requirements and were reviewed, and that any deficiencies identified during the testing were properly resolved.

No violations or deviations were identified.

The following surveillance activities were observed and reviewed:

- Eddy current inspection.
- 174 tubes were plugged.
- 21 tube plugs were replaced.

Braidwood Technical Surveillance 4.5.0-1, Steam Generator Eddy Current Inspection. During Unit 1's refueling outage, a steam generator eddy current inspection was performed on 100% of the tubes in all four steam generators from the hot leg tube end through the U-bend. In addition, 50% of the tubes were examined full length from tube end to tube end. A total of 174 tubes were plugged due to indications at the hot leg support plates and antivibration bar (AVB) wear.

In addition to the tubes plugged, a total of 21 Inconel 600 mechanical tube plugs were replaced with Inconel 690 mechanical tube plugs. This was accomplished in accordance with NRC Bulletin 89-01, "Failure of Westinghouse SG Tube Mechanical Plugs."

8. Report Review

During the inspection period, the inspector reviewed the licensee's Monthly Performance Report for October 1992. The inspector confirmed that the information provided met the requirements of Technical Specification 6.9.1.8 and Regulatory Guide 1.16.

The inspector also reviewed the licensee's Monthly Plant Status Report for September 1992.

No violations or deviations were identified.

9. Exit Interview (30703)

The inspectors met with the licensee representatives denoted in Paragraph 1 during the inspection period and at the conclusion of the inspection on November 30, 1992. The inspectors summarized the scope and results of the inspection and discussed the likely content of this inspection report. The licensee acknowledged the information and did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.