U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-456/91017(DRP); 50-457/91015(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: June 2 through July 13, 1991

Inspectors: S. G. Du Pont
R. A. Kopriva
J. A. Gavula

Approved By: G.D. Smith for M. J. Farber, Chief
Reactor Projects Section 1A

7/29/91
Date

Inspection Summary

Inspection from June 2 through July 13, 1991 (Reports No. 50-456/91017(DRP); 50-457/91015(DRP))

Areas Inspected: Routine, unannounced safety inspection by the resident inspectors of licensee action on previously identified items; licensee event report review; review of Generic Letter 88-17 (TI 2515/103), loss of decay heat removal; operational safety verification; monthly maintenance observation; monthly surveillance observation; and report review and meetings.

Results: One non-cited violation was identified in Paragraph 6.

- A non-cited violation of Appendix B, Section 3.2, of Facility Operating Licenses NPF-72 and NPF-77 was identified for failure to provide notification of report changes within the required time period.

- Site and corporate engineering demonstrated good engineering practices by performing a detailed evaluation of a NSSS instrumentation error and independently discovered additional factors affecting the error. Engineering also provided timely information of the errors, recommended temporary actions and permanent corrective actions to the station.

- Station quality assurance demonstrated good performance base auditing by discovering that a notification requirement of the License was not met.

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- Operations continued to demonstrate good safe operations. Both units were actively pursuing annunciator black board conditions during the inspection period.

- The station's efforts to meet Generic Letter 88-17 are near completion with final completion expected during the Unit 2 1991 Fall refueling outage. The efforts completed on Unit 1 were effective and beyond the requirements of Generic Letter 88-17. The residual heat removal parameter display is considered to be a program strength.

DETAILS

1. Persons Contacted

   Commonwealth Edison Company (CECo)

   *K. L. Kofron, Station Manager
   *G. E. Groth, Production Superintendent
   D. E. O'Brien, Technical Superintendent
   G. R. Masters, Assistant Superintendent - Operations
   *R. J. Legner, Services Director
   A. D. Antonio, Nuclear Quality Program Superintendent
   D. E. Cooper, Technical Staff Supervisor
   S. Roth, Security Administrator
   K. G. Bartes, Nuclear Safety Supervisor
   *A. Haeger, Regulatory Assurance Supervisor
   *E. W. Carroll, Regulatory Assurance
   *P. L. Maher, Assistant Technical Staff Supervisor
   *R. Yungk, Operating Engineer
   *S. W. Mitchell, Nuclear Safety
   *R. H. Richard, Operating Staff
   *F. A. Lesage, Nuclear Quality Program

   *Denotes those attending the exit interview conducted on July 16, 1991, and at other times throughout the inspection period.

   The inspectors also talked with and interviewed several other licensee employees.

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

   a. Violation

      (Closed) 457/91011-01: The violation pertained to Technical Specification 6.8, where procedures shall be established, implemented, and maintained. On April 17, 1991, by not following Procedure BwOP CV-8, "CV System Mixed Bed/Cation Demineralizer Operation," the licensee experienced a ruptured valve diaphragm (ZCV 8524A) allowing a spill to occur, and contaminating an individual and the immediate area. The resident inspectors have reviewed the licensee's corrective actions to ensure adherence to procedures and consider this issue closed.

   b. Unresolved Item

      (Closed) 456/90019-01: Waterhammer in Unit 1 steamline on October 11, 1990. Revisions to Operating procedure BwOP MS-9 had not been implemented based upon a similar event at Byron to prevent waterhammers in the main steamline. BwOP MS-9 was revised and caution cards were installed to require use of BwOP MS-9 for opening main steam isolation and bypass valves. Additionally, BwOP MS-9 requires the technical staff to visually inspect the steamline after restoring main steam due to isolation of the steamline. The inspector reviewed the licensee's actions and subsequent training on the event and procedure revisions. The licensee's actions were found to be adequate and this item is closed.

No violations or deviations were identified.

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

Through review of records, the following LERs were reviewed to determine that reportability requirements were fulfilled, that immediate corrective action was accomplished, and that corrective action to prevent recurrence had been or would be accomplished in accordance with technical specifications:

(Closed) 457/91002-LL
(Closed) 456/91005-LL
(Closed) 456/91006-LL

No violations or deviations were identified.

4. Review of Generic Letter 88-17, Loss of Decay Heat Removal (TI 2515/103)

Generic Letter 88-17, "Loss of Decay Heat Removal," recommended six long term program enhancements; (1) provide reliable indication of parameters that describe the state of the Reactor Coolant System (RCS) and systems used to cool the RCS for both normal and accident conditions, (2) develop and implement procedures that cover reduced inventory operation, (3) assure that adequate operating and available equipment of high reliability is provided for cooling the RCS and for avoiding a loss of RCS cooling, (4) conduct analyses to supplement existing information and develop a basis for procedures, instrumentation, and equipment response, (5) review technical specifications (TS) to ensure that TS do not restrict or limit the safety benefit of the actions identified by Generic Letter 88-17, and (6) procedures should be examined as necessary to reasonably minimize the likelihood of loss of decay heat removal (DHR).

The licensee made the following commitments in a letter, dated January 31, 1989, to Dr. T. E. Murley, Director of the Office of Nuclear Reactor Regulation:

a. Program Enhancement 1, Instrumentation

RCS level indication is provided by two independent (duplicated) indicating systems consisting of three level indicators; a narrow range indication for mid-loop operation, a refueling cavity indication, and a wide range indication that spans the ranges of both the narrow range and refueling range.

The inspector verified that the licensee completed the installation of the level indication system on Unit 1 during the recently completed refueling outage via modification M20-1-89-014. The inspector verified that modification M20-2-89-013 is scheduled to be installed on Unit 2 during the fall 1991 refueling outage. The inspector also verified that Unit 2 modification M20-2-89-013 meets the requirements of Generic Letter 88-17 and the licensee's commitments. The requirements of Program Enhancement 1 is considered to be met since modification M20-2-89-013 is scheduled and will duplicate the Unit 1 modified level indicating system on Unit 2.

b. Program Enhancement 2, Procedures

The licensee committed to revi ; the existing normal and abnormal operating procedures to cover normal and off-normal operation of the RCS, containment, and support systems for plant conditions requiring DHR system operation. The following procedures were identified:

General Procedure BwGP 100-5, Plant Shutdown and Cooldown.

General Procedure BwGP 100-6, Refueling Outage.

Operating Procedure BwOP RC-4a and b, Unit 1 (Unit 2) Reactor Coolant System Drain.

Operating Procedure BwOP RC-7, Isolating a Reactor Coolant System Loop.

Operating Procedures BwOP RC-8a(b), Unit 1 (Unit 2) Restoring a Reactor Coolant System Loop to Service.

Abnormal Operating Procedure BwOA PRI-10, Loss of RH Cooling.

The licensee also committed to review the Westinghouse Owners Group recommendations on Generic Letter 88-17 and provide further procedure revisions if warranted.

The inspector reviewed the following procedures to verify that the licensee's commitments were met:

1(2)BwGP 100-5, Revision 3 (3)
1(2)BwGP 100-6, Revision 2 (2)
BwOP Rc-4a, Revision 1
BwOP RC-4b, Revision 0
BwOP RC-7, Revision 6
BwOP RC-8a, Revision 5
BwOP RC-8b, Revision 6
1(2)BwOA PRI-10, Revision 55 (53)

In addition, the inspector reviewed and verified that operating surveillance procedures 1(2)BwOS 0.1-5, Unit 1(2) Mode 5 Shiftly and Daily Surveillance and 1(2)BwOS 0.1-6, Unit 1(2) Mode 6 Shiftly and Daily Surveillance met the requirements of Generic Letter 88-17 and the licensee's commitments.

c. Program Enhancement 3, Equipment

The licensee committed to modify the DHR and RCS by removing the autoclosure feature of the residual heat removal system suction valves from the RCS to assure operability and availability of highly reliable equipment for cooling the RCS.

The modification was verified to be installed on Unit 1 during the recent refueling outage ending May 1991. The inspector also verified that modification M20-2-89-030 will remove the autoclosure features on Unit 2 during the scheduled Fall 1991 refueling outage. These enhancement actions are considered to meet the requirements of Generic Letter 88-17 and the licensee's commitments. Installation of modification M20-2-89-030 is scheduled to be observed by the NRC during the Unit 2 refueling outage and this program enhancement is considered to be closed.

d. Program Enhancement 4, Analyses

The inspector verified that Westinghouse analyses were performed for the modifications to the residual heat removal system suction valve autoclosure feature and the installation of a second independent RCS level indication system. Additionally, the Westinghouse Owners Group performed analyses to predict the magnitude of level variations which exist throughout the RCS due to operation of the residual heat removal system during mid-loop operating conditions. Other analyses were verified to have been performed in support of providing procedure revisions. These actions are considered to meet the requirements of Generic Letter 88-17 and the licensee's commitment and this program enhancement is considered to be closed.

e. Program Enhancement 5, TS Changes

The inspector verified that the following TS change requests were made by the licensee based upon analyses discussed per Program Enhancement 4:

TS 4.4.9.3.2 Deletion of autoclosure interlock on residual heat removal (RH) suction valves.

TS 3/4.5.4.1 Requiring all safety injection pumps to be operable while in Mode 5 with pressurizer level greater than 5% and in Mode 6 with the vessel head on, and

TS 3/4.5.4.2 Requiring either one safety injection pump available or an adequate hot leg vent path to allow gravity feed from the refueling water storage tank to the RCS while in Modes 5 and 6 with pressurizer level equal to or less than 5%.

TS 4.9.8.1 and 4.9.8.2 Revised to allow RH flow rate to be reduced to greater than or equal to 1000 gpm with RCS temperature greater than or equal to 140 degrees F.

These actions are considered to meet the requirements of Generic Letter 88-17 and the licensee's commitments. This program enhancement is considered to be closed.

f. Program Enhancement 6, RCS Perturbations

The licensee re-examined their response to Generic Letter 88-17, Expeditious Action No. 5, as documented in a letter, dated December 30, 1988, to Dr. T. E. Murley, Director of the Office of Nuclear Reactor Regulation (NRR). In addition, the licensee met with NRR on January 25, 1989. As a result of the meeting, the licensee evaluated providing training on loss of decay heat removal to other than licensed reactor and senior reactor operators. The licensee committed to providing this training to licensed operators, non-licensed operators, chemistry technicians, and technical staff personnel in a letter, dated September 15, 1989, to Dr. T. E. Murley. These actions are considered to meet the requirements of Generic Letter 88-17 and the licensee's commitment. This program enhancement is considered closed.

g. Summary

The licensee met the recommended enhancements and their commitments with the exceptions of installing the second independent RCS level indicating system and removing the RH suction valve autoclosure interlocks on Unit 2. Both of these commitments are scheduled to be completed during the Fall 1991 refueling outage. The inspector verified that the related modifications accomplishing these commitments are scheduled, prepared, and approved. The inspector observed and reviewed the modifications completed on Unit 1 during the Spring 1991 outage and will also observe the modifications during the Fall 1991 outage. This issue is considered closed.

Additionally, the licensee designed a computer graphic display for the control room. The graphic displays data in real time of refueling water level and RH pump parameters. The display is a mimic with reference to the plant elevation using bar graphs and literal values for refueling and pressurizer levels. This enhancement is considered to be beyond the requirements of the generic letter and a program strength providing operators with essential information without requiring interpretation.

No violations or deviations were identified.

5. Corporate and Onsite Engineering Response to Steam Generator Indicated Level Errors

The licensee's corporate engineering organization discovered, during an engineering evaluation, that errors existed in the Braidwood steam generator level indicating system for scaling. The original Westinghouse supplied information for the steam generator narrow range level transmitters was for 223 inches at 0% indication and 60 inches at 100% indication. The correct information was determined to be 227.85 inches at 0% indication and 64.85 inches at 100% indication. This error, about four inches, resulted in a less than conservative effect on the reactor protection low-low steam generator water level trip (40.8% water level).

Additionally, engineering determined that the T hot (hot leg temperature) reduction program initiated by the industry had a compounding impact on the scaling of these level transmitters in that the scaling of the narrow range level transmitters assumes a determined reference steam generator pressure. The reference pressure is selected to have a 2% tolerance due to density deference between the 100% and 0% power conditions. Since reducing T hot affects steam generator pressure, the accuracy of these transmitters was also affected.

The licensee took several immediate actions, including requesting corrected scaling from Westinghouse and performing an operability assessment.

The effects of the Westinghouse miscalibration and T hot reduction were evaluated by the licensee to determine the total error and the effect upon accident analyses relying on the low-low steam generator water level trip. These analyses assumes an initial nominal level error of plus 5% to minus 5% of the narrow range scale. The combined effect of the miscalibration and reduced T hot resulted in a lower actual or less conservative water level. Calculations performed by engineering demonstrated that the magnitude of error at the low-low level trip was bounded by the existing margin determined in the current licensee's statistical setpoint study and did not impact the TS value. It was also determined that the combined error did not impact the high-high steam generator water level isolation and turbine trip function, since the lower actual level is conservative for this function and would result in an earlier actuation of the high-high level isolation and turbine trip.

A new system error was determined to be plus 6.6% to minus 9.5%. The increase in the negative error (minus 9.5%) is directly related to the effects from the combination of the miscalibration error and T hot reduction. The positive error (plus 6.6%) was not related to either the miscalibration error or T hot reduction, but a Westinghouse evaluation of the Braidwood specific level indicating system which increased the typical plus 5% error that is generally applied to Westinghouse four-loop pressurized light water reactors. The licensee is currently evaluating strategies to apply the correct scaling. These strategies include recalibrating the level transmitters at power. Engineering supplied the revised calibration numbers to both the Braidwood and Byron stations on July 12, 1991.

Engineering also made recommendations that a special operating order should be issued providing an operator aid at the control room control panels indicating the revised steam generator levels.

Initially, station management decided not to incorporate engineering's recommendations to initiate a temporary change to the emergency operating procedures or to develop a temporary operator aid. This decision was sound because the existing margin contained in the emergency operating procedures provided adequate bounding of the accident analysis and would not require a permanent change once the instruments were recalibrated. However, the inspector noted to the licensee that not all of the operations personnel were aware of the level error. Station management agreed and issued a notice to all operations personnel of the error and the requirement to adhere to the existing procedures. The inspector found these actions to be appropriate.

The inspector determined that the actions of corporate and site engineering were detailed, accurate, and timely. This issue is considered to be closed.

No violations or deviations were identified.

6. Evaluation of Licensee Self-Assessment Capability (40500)

On January 29, 1991, the Illinois Environmental Protection Agency issued their final National Pollutant Discharge Elimination System (NPDES) permit to the Braidwood Nuclear Power Station. The licensee received the NPDES permit, reviewed it, and on February 28, 1991, the permit became effective for the station.

On July 3, 1991, the licensee informed the NRC that their final NPDES Permit had been accepted and was effective as of February 28, 1991. The station quality assurance organization discovered during an audit that the revision of the NPDES permit had been completed without meeting the 30 day notification of the NRC as required by Appendix B of the License. Quality assurance immediately notified station management of the failure to make the required notification and corrective actions were immediately initiated. Appendix B, Section 3.2, of Facility Operating Licenses NPF-72 and NPF-77 requires changes to, or renewals of, the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved.

The actions of the station quality assurance organization demonstrated good performance based audit practices.

The licensee's failure to notify the NRC within the required time requirement is a violation. Due to the fact that the licensee met the requirements of 10 CFR 2, Appendix C, Section V.G., by identifying and resolving the deficiency, this will be a non-cited violation as stated in 10 CFR 2, Appendix C, Section V.A., and a Notice of Violation will not be issued.

One non-cited violation was identified.

7. 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.

On a sampling basis the inspectors verified proper control room staffing and coordination of plant activities; verified operator adherence with procedures and TS; monitored control room indications for abnormalities; verified that electrical power was available; and observed the frequency of plant and control room visits by station managers. During the inspection period, both units maintained a near annunciator black board.

During tours of accessible areas of the plant, the inspectors made note of general plant and equipment conditions, including control of activities in progress. The specific areas observed were:

- Engineered Safety Features (ESF) Systems

  Accessible portions of ESF systems and their support systems components were inspected to verify operability through observation of instrumentation and proper valve and electrical power alignment. The inspectors also visually inspected components for material conditions. The material conditions of the ESF systems were good and the systems were available through most of the inspection period.

- Radiation Protection Controls

  The inspectors verified that workers were following health physics procedures and randomly examined radiation protection instrumentation for operability and calibration. During this inspection, only a few minor contaminated spills occurred, indicating an improvement compared to the previous inspection periods.

- Security

  During the inspection period, the inspectors monitored the licensee's security program to ensure that observed actions were being implemented according to their approved security plan. No problems were observed or encountered during the inspection period.

- Housekeeping and Plant Cleanliness

  The inspectors monitored the status of housekeeping and plant cleanliness for fire protection and protection of safety-related equipment from intrusion of foreign matter. The efforts to remove the various equipment, rods, and debris remaining from the completed Unit 1 refueling outage continued with some progress. However, additional attention is still required.

The inspectors also monitored various records, such as tagouts, jumpers, shiftly logs and surveillances, daily orders, maintenance items, various chemistry and radiological sampling and analysis, third party review results, overtime records, QA and/or QC audit results and postings required per 10 CFR 19.11. No problems were encountered and the unit logs continued to improve in both the quality and quantity of entries.

No violations or deviations were identified.

8. Monthly Maintenance Observation (62703)

Routinely, station maintenance activities were observed and/or reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards, and in conformance with TS.

The following items were also considered during this review: approvals were obtained prior to initiating the work; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; and activities were accomplished by qualified personnel.

The following maintenance activities were observed and reviewed:

OPR01J - Liquid Radwaste Radiation Monitor

Unit 1B Heater Drain Pump

No violations or deviations were identified.

9. Monthly Surveillance Observation (61726)

The inspectors observed several of the surveillance tests required by TS during the inspection period and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that results conformed with TS and procedure requirements and were reviewed, and that any deficiencies identified during the testing were properly resolved.

The following surveillance activities were observed and reviewed:

Unit 1

Flux Mapping.

Unit 1A and 1B Diesel Generator - BwOS 8.1.1.2a-2, Diesel Generator Operability Monthly (Staggered) and Semi-Annual (Staggered) Surveillance.

Unit 2

Unit 2A and 2B Diesel Generator - BwOS 8.1.1.2a-2, Diesel Generator Operability Monthly (Staggered) and Semi-Annual (Staggered) Surveillance.

Unit 2B Diesel Generator - BwOS 3.2.1-816, ESFAS Instrumentation Slave Relay Surveillance.

No violations or deviations were identified.

10. Report Review

During the inspection period, the inspector reviewed the licensee's Monthly Performance Report for May and June 1991. 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 April and May 1991.

No violations or deviations were identified.

11. Meetings and Other Activities (30702)

Site Visits by NRC Staff

On July 2-3, 1991, the Braidwood Section Chief of the Division of Reactor Projects was onsite for a site tour and to interface with the licensee and resident inspectors. Concerns pertaining to housekeeping and locked carts during the course of the plant tour were discussed with the licensee.

12. Violations For Which A "Notice of Violation" Will Not Be Issued

The NRC uses the Notice of Violation as a standard method for formalizing the existence of a violation of a legally binding requirement. However, because the NRC wants to encourage and support licensee's initiatives for self-identification and correction of problems, the NRC will not generally issue a Notice of Violation for a violation that meets the tests of 10 CFR 2, Appendix C, Section V.A. These tests are: 1) the violation was identified by the licensee; 2) the violation would be categorized as Severity Level IV or V; 3) the violation will be corrected, including measures to prevent recurrence, within a reasonable time period; and 4) it was not a violation that could reasonably be expected to have been prevented by the licensee's corrective action for a previous violation. A violation of regulatory requirements identified during this inspection for which a Notice of Violation will not be issued is discussed in Paragraph 6.

13. Exit Interview

The inspectors met with the licensee representatives denoted in Paragraph 1 during the inspection period and at the conclusion of the inspection on July 16, 1991. 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.