U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-456/86050(DRP); 50-457/86037(DRP)

Docket Nos. 50-456; 50-457 Licenses No. NPF-59; CPPR-132; CPPR-133

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: Braidwood Station, Units 1 and 2

Inspection At: Braidwood Site, Braidwood, IL

Inspection Conducted: August 31 through November 1, 1986

Inspectors: NRC

T. M. Tongue

W. J. Kropp

T. E. Taylor

EG&G Idaho, Inc.

B. Barnes

R. Larson

J. Townsend

R. Evans

Approved By: P. R. Pelke for
Ronald N. Gardner, Chief
Braidwood Project Section

11/20/86
Date

Inspection Summary

Inspection on August 31 through November 1, 1986 (Reports No. 50-456/86050(DRP); 50-457/86037(DRP))
Areas Inspected: Routine, unannounced safety inspection of licensee action on previously identified items; regional request; construction worker concern; allegations; temporary license request; NUREG-0737; quality first; comparison of as-built plant to FSAR description; Title 10 requirements; events occurring onsite during inspection period; Commissioner Bernthal visit; operational staffing; plant tours and independent assessments; pressurizer code safeties; and meetings, training, and other activities.
Results: No violations or deviations were identified.

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

Corporate Personnel

*B. Thomas, Executive Vice President
*C. Reed, Vice President, Nuclear Operations
*T. J. Maiman, Vice President, Projects
*B. M. Saunders, Nuclear Security Administrator
*D. J. Scott, Operations Manager, NSD
*K. Graesser, Division Vice President
*D. Galle, Assistant Vice President and General Manager
*W. Shewski, Quality Assurance Manager
*D. Farrar, Director, Nuclear Licensing

Braidwood Personnel

*M. J. Wallace, Project Manager
*R. M. Preston, Quality First Director
*T. F. Hallaren, Administrative Services Director
*E. E. Fitzpatrick, Station Manager
*C. J. Tomashek, Project Startup Superintendent
*D. L. Shamblin, Project Construction Superintendent
*T. E. Quaka, Site Quality Assurance Superintendent
*K. L. Kofron, Production Superintendent
*C. W. Schroeder, Station Services Superintendent
*G. F. Marcus, Assistant to Manager Quality Assurance
*G. E. Groth, Assistant Superintendent - Construction
*D. E. Paquette, Assistant Superintendent - Maintenance
*L. E. Davis, Assistant Superintendent - Technical Services
*P. Cretens, Assistant Superintendent - Station Startup
*D. O'Brien, Assistant Superintendent - Operations
*F. D. Willaford, Security Administrator
*T. C. Meyer, Station Fire Marshall
*P. L. Barnes, Regulatory Assurance Supervisor
*R. D. Kyrouac, Station Quality Assurance Supervisor
*M. Takaki, Quality Control Supervisor
*A. J. D'Antonio, Regulatory Assurance Supervisor
*E. R. Wendorf, PFE Mechanical supervisor
*L. M. Kline, Regulatory Assurance Group Leader
*D. M. Kapinus, Startup Test Program Supervisor
*L. W. Raney, Supervisor Nuclear Safety Group
*N. Tomis, OAD Group Supervising Engineer
*A. Iturrieta, OAD Supervising Engineer
*L. Bush, Station Readiness Coordinator
*R. J. Ungeran, Operating Engineer
*T. W. Simpkin, Regulatory Assurance
*D. L. Cecchett, Regulatory Assurance

*J. K. Jasnosz, Regulatory Assurance
*P. A. Boyle, Regulatory Assurance
*R. J. Sievert, KSA Consultant
*J. P. Knight, Quality First Consultant
W. J. Gagnon, Quality First Consultant
*T. Bobic, Master Electrician - Maintenance
*J. P. Leider, PED
*J. Huffman, Master Electrician - Maintenance
*J. A. Zych, Quality Assurance

Sargent & Lundy

*D. A. Gallagher, Project Field Manager

Nova Power

*T. Lewis, Startup Staff

Westinghouse

*W. Poirier, Project Manager

MHB Technical Associates

S. Sholly
J. Kieberman

Illinois Department of Nuclear Safety

*B. Metrow, ASME Code Engineer/Division of Engineering

NRR/NRC

*D. Hickman, Training Assessment Specialist
*W. H. Swenson, Operating Experience Engineer
*C. E. Rossi, Assistant Director, PWR Licensing
*B. Clayton, Technical Assistant to Director, PWR Licensing
*S. Varga, Project Director
*H. J. Miller, Deputy Director
*T. M. Novak, Deputy Director

The inspectors also talked with and interviewed other licensee employees, including members of the technical and engineering staffs, startup engineers, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, contract security personnel, and construction personnel.

*Denotes those attending one or more exit interviews conducted on September 11, 18, 30, October 2, 8, 21, 23, 27 and 30, 1986, and informally at various times throughout the inspection period.

2. Licensee Action on Previously Identified Items

a. Open Items

(Closed) 456/84013-03: The only mechanical items that require a gas purge are the SI accumulators and control room refrigeration units. Vapor inhibitors or the coating of internal parts were not part of the contractor's preventive maintenance program. This item is considered closed based on the closure of Unresolved Item No. 456/84013-02 in this report which pertained to the licensee's mechanical preventive maintenance program. The licensee's activities to resolve Item No. 456/84013-02 included an evaluation of every safety-related component to determine if the maintenance required by the vendor agreed with the actual maintenance performed by the site contractors. Any differences were then evaluated for potential impact on the hardware and any necessary corrective actions required were identified to restore the components. This item is considered closed.

(Closed) 456/84042-05; 457/84038-05: Engineering Change Notice (ECN) 22822, issued on January 24, 1985, defined clearance guidelines for component supports. This ECN did not require clearance guidelines for component supports to be applicable for supports installed prior to January 29, 1985. During the closure of violations No. 456/8309-02; 457/83009-01, documented in Inspection Reports No. 456/86028; 457/86022, the inspector determined that the clearance guidelines established by ECN 22822 were adequately addressed in the final clearance walkdown program. Therefore, any clearance problems resulting from installation prior to January 24, 1985 should be identified. This item is considered closed.

(Closed) 456/85023-05; 457/85024-05: Thirteen structural steel bolted connections had torque valves below the installation torque, and turn-of-the-nut installations were subject to fluctuations in foot-pound torque. A followup inspection documented in Inspection Reports No. 456/85058; 457/85054 was performed. The followup inspection identified that the only issue remaining was a reinspection of "slip-critical connections" which were tightened by the turn-of-the-nut method. The inspector reviewed the results of this reinspection which were documented in a Sargent & Lundy letter, dated September 24, 1984. This letter stated that all "slip-critical connections" that were previously installed using the turn-of-the-nut method were reinspected utilizing a calibrated torque wrench. This item is considered closed.

(Closed) 456/85033-01: A previous inspection determined that several locations on the polar crane had the following deficiencies: loose bolts; missing washers; shims used during installation that were still in place; extra bolting material lying loose; and up and down movement of the rail. A review of G. K. Newberg's (GKN) reinspection was conducted and it was determined that the following actions were taken to correct the above findings. Reinspections were performed by GKN which identified all locations that had missing washers or remaining shims. Rework was then performed to add any missing washers and

remove all remaining shims. All bolts were re-torqued with Pittsburgh Testing Lab performing inspection on 10% of the bolts. Inspection procedure PCD-14 will be used during operational surveillances to inspect for loose bolts and wear on the polar crane. This item is considered closed.

(Open) 456/86016-03(DRP): Quality of Release to Operations (RTO) Evaluations. The inspectors reviewed the following completed RTO packages:

AP-13, "Auxiliary Power"
IP-10, "Instrument Power"
RH-10, "Residual Heat Removal"

Any deficiencies identified were minor and would not affect system ability to meet its intended safety function and none were required to be completed prior to fuel load. This item will remain open pending the inspector's review of a larger RTO sample prior to initial criticality.

b. Unresolved Items

(Closed) 456/83017-04: Inspection of as-built fuel oil piping against appropriate pipe drawings revealed that nameplate information was not consistent between the as-built condition and the drawings. The inspector verified that correct nameplates were installed on the transfer pump suction strainer filter housing diesel oil system (Unit 1) by inspection of the filter housing and review of the installation Field Change Order and Equipment Installation Repair-Retrofit Record. The Field Change Order and Equipment Installation Repair-Retrofit Record also changed out the old filter cartridges and replaced them with the filters specified on the appropriate drawings. The paperwork is in place for Unit 2, but the work has not been performed. A review was also made of the Phillips Getschow Co. drawings to verify that they showed the new condition. Based on the above inspection, this item is closed.

(Closed) 456/84013-02; (Open) 457/84013-02: An NRC sample inspection of Phillips Getschow and Company (PGCo) safety-related mechanical equipment maintenance procedures showed the preventive maintenance portion of these procedures to be inadequate. The NRC report indicated that an inspection program was needed to document that preventive maintenance had been properly performed. This inspection program should verify that mechanical equipment quality control inspectors have witnessed preventive maintenance such as lubrication, periodic rotation of the shafts on rotating equipment, and that protective covers are installed and functional. In addition, the licensee was to review the preventive maintenance requirements of all safety-related equipment to assure that this equipment had been properly maintained in the past and to assess any equipment degradation and initiate corrective action where necessary.

PGCo Procedure No. PGCP-37, Revision 7, along with CECo Procedures No. PCD-14 and PCD-29 were developed and/or revised by the licensee to address both past and future preventive maintenance requirements of mechanical equipment. These procedures were used to evaluate possible degradation related to the lack of preventive maintenance on safety-related equipment and to provide corresponding corrective actions where required.

A review of the procedures cited above indicates that the licensee now has in place an adequate preventive maintenance program covering 1,647 pieces of safety-related mechanical equipment for Braidwood Units 1 and 2. Approximately 402 pieces of equipment (about 25%) required some form of corrective action because of prior inadequate preventive maintenance. To check on the adequacy of these corrective actions, six documents, each relating to a different piece of safety-related mechanical equipment, were selected from the populace of 1,647 documents generated as a part of CECo NCR 689. All six pieces of equipment were intentionally selected because each required some corrective action because of prior inadequate preventive maintenance. The six pieces of equipment selected were:

Equipment Name - Equipment Number	Corrective Action Summary
(1) Aux. Feedwater Pump - 1AF01PB	Inspect shaft bearings, oil flush bearings, megger motor windings, rotate shaft 180° every 3 months, etc.
(2) Excess Letdown Heat Exchanger - 1CV01AA	Check and change desiccant, inspect internal cleanliness, check covers and contamination barriers, circulate air to dry insides.
(3) Control Room HVAC System Return Fan - 0VC02CB Fan	Rotate shaft every 3 months, megger motor windings, check space heaters for function, grease bearings, perform initial "vibration check."
(4) Primary Containment Vent System - 1VP01CC	Periodic fan rotation and greasing of bearings, motor windings meggered, check function of motor winding space heaters, perform initial vibration check.
(5) Essential Service Water Pump Motor - 2SX01PA-M	Check and fill bearing reservoir with oil, rotate shaft periodically, check oil ring function, check space heaters for function, check winding electrical resistance, inspect bearings.

(6)	Regenerative Heat Exchanger - 2CV03AB	Periodically check and as required change desiccant, air ventilate to remove wetness, inspect internals for corrosion.

A detailed review of each of the above six pieces of equipment and associated corrective actions, showed the corrective actions to be both appropriate and adequate.

Even though the six piece equipment sample was determined to have received adequate corrective maintenance, CECo personnel were questioned further regarding any known degradation related failures that may have occurred because of improper preventive maintenance. In response to this questioning, CECo subcontractor/consultant, Mr. Ron Cagne of Daniel Construction Company described eight nickel-cadmium batteries used for starting diesel engines that drive the auxiliary feedwater pumps which, as a result of improper preventive maintenance, were required to be replaced. These eight failures were the only instances of degradation related failures cited of the 1,647 safety-related items. The previously cited preventive maintenance procedures are considered adequate to assure these batteries receive proper preventive maintenance in the future.

To date, about 1,600 of the 1,647 equipment reviews which are part of CECo NCR 689 are complete. All Braidwood Unit 1 and unit common safety-related equipment is complete; about 47 Unit 2 items remain to be completed. When the 47 Unit 2 items are complete, this issue will be reviewed by the inspector. Based upon a review of revised preventive maintenance procedures, review of CECo NCR 689, a detailed review of six pieces of equipment requiring corrective action resulting from prior improper maintenance, and a detailed review of the only known degradation related failure, the inspector concluded that both the preventive maintenance programs now in place and the corrective actions implemented under NCR 689 are adequate. For Unit 1 and unit common equipment, this item is considered closed.

(Closed) 456/84013-04; 457/84013-04: The electrical preventative maintenance program was not clearly defined, implemented and documented. The licensee issued, in February 1985, Procedure No. PCD-29, "Instruction for Completing The Braidwood Equipment Preventive Maintenance Evaluation Form." The purpose of this procedure was to define the methods to be used by project personnel for completing the "Braidwood Equipment Preventative Evaluation" sheets. These evaluations were performed to document the comparison of past preventive maintenance activities for each piece of safety-related equipment with applicable American National Standard Institute (ANSI) standards and the manufacturer's recommendations. This comparison was utilized to assess any equipment degradation and to initiate the necessary corrective action. The inspector reviewed

30 evaluation forms and noted no problems. The inspector selected three of these (PM-0121, PM-0083, and PM-0022) evaluations to determine the effectiveness of the licensee's evaluations. This was accomplished by reviewing the manufacturer's recommendations to determine if they were correctly described on the evaluation forms and that any corrective actions described on the forms were implemented. The inspector also verified that the licensee considered the effects of past preventive maintenance on the equipment's environmental qualifications. No problems were noted with these three evaluation forms and therefore it appears the licensee effectively implemented Procedure No. PCD-29. This item is considered closed.

(Closed) 456/85023-04; 457/85024-04: Category 1, seismic nonsafety-related, electrical bolted connections are not being inspected by personnel independent of cost and schedule.

The inspector verified that this item has been properly resolved by the licensee. The electrical bolted connections have been walked down, inspected, and deficiencies have been corrected. L. K. Comstock (LKC) Procedures No. 4.3.12 and 4.8.12 (including previous revisions) were reviewed. These procedures required that the bolted connections be inspected by LKC QC inspectors for proper material and torque. The results of these inspections were documented on appropriate inspection reports. These procedures have been in effect since the beginning of the project. Based on the licensee actions and the review of LKC procedures this item is considered closed.

(Closed) 456/85038-01; 457/85037-01: Two pressure switches, 1PSL-AF051 and 1PSL-AF055 were procured from United Electric as nonsafety-related. There were no technical or quality requirements identified on the purchase order. The Sargent & Lundy (S&L) Instrument Data Index and Instrument Data Sheet, PS-180, identified these instruments as Class 1E and Seismic Category I. The inspector verified that documentation existed which qualified the United Electric pressure switch, Model 552, for nuclear applications. Since these switches were procured as nonsafety-related, the inspector was concerned that these switches might not have been properly receipt inspected, stored, installed, calibrated, or otherwise controlled. The licensee committed to: (1) evaluate the procurement and subsequent activities associated with these switches to determine their acceptability, and (2) evaluate nonsafety purchase orders to determine if other safety-related balance of plant instruments were procured as nonsafety-related. This review was conducted by the site QA organization.

The inspector reviewed PGCo NCR 6189 which was issued to track this issue. Documentation (Instrument Installation and Removal Form, Equipment Installation Record, Supplemental Data Record sheet) shows that the pressure switches were handled as safety-related with the exception of a receipt inspection. A receipt inspection was completed

and given to QA for evaluation on Material and Equipment Receiving and Inspection Report (MRR) 20276. The inspector reviewed MRR 20276 which was evaluated by QA and found to be acceptable. The inspector verified closeout of NCR 6189.

The licensee performed an evaluation of nonsafety-related purchase orders to determine if other safety-related balance of plant instruments were procured as nonsafety-related. This evaluation was performed by the licensee's site QA organization and the results were documented in Surveillance Report No. 5073, dated October 3-17, 1985. During this surveillance, site QA reviewed 55 nonsafety-related purchase orders which procured 376 instruments. This review verified that at the time of purchase all these instruments were classified as nonsafety-related. However, the surveillance did identify that six of these instruments were subsequently upgraded to safety-related. The inspector reviewed MRR 20862 which documented the receipt inspection required for the six safety-related instruments. This MRR documented that the six instruments were qualified for safety-related service by Sargent & Lundy's Component Qualification Division letter EQBB-013237. This item is considered closed.

(Closed) 456/85057-03: An NRC open item was issued due to inconsistencies in the revisions of the piping and instrumentation diagrams (P&IDs) and the control and instrumentation diagrams (C&IDs) listed in the procedures and included in the drawing package for the Unit 1 reactor coolant system hydrostatic test. In a number of cases revisions listed in the hydro procedure appeared to have been superseded by revisions listed in the fill and vent procedure. The inspector requested that the licensee review the changes made to each drawing used in the test and evaluate them for impact on test results.

The current NRC inspection reveals that the licensee has satisfactorily reviewed the changes made to the drawings and concludes the changes have no impact on the test results. The licensee's response is sufficient and the item is considered closed.

(Open) 456/86025-03: Four issues concerning: (1) Administrative procedures describing scope and responsibility of Preventive Maintenance Program; (2) Procedures for inservice program administration, corrective and preventive maintenance and surveillance procedures; (3) Maintenance history trending; and (4) Work request procedure lacking detail for some work request activities. The inspector reviewed information the licensee submitted for these items. The corrective action taken is considered adequate to close parts (1), (3), and (4) of this unresolved item. Item (2) is complete except for inservice inspection program administrative procedures. The remaining item will be reviewed during a subsequent inspection.

(Closed) 456/86025-04: Administrative procedure for control of licensee's "GSIN" program which is used for controlling scheduling of calibration and surveillances of safety-related components. The inspector reviewed the licensee's corrective action, using Procedure No. BwIP 2100-001, "Frequency of Calibration of Plant Instrumentation," to control the activities of concern and considers this item closed.

(Closed) 456/86025-05: The licensee failed to write departmental procedures describing duties of department surveillance coordinators. The inspector's review of this item showed that this requirement has been deleted from BwAP 1400-1. The coordination and tracking of surveillances required by technical specifications and performed on safety-related equipment is not affected by this change. This item is considered closed.

(Closed) 456/86031-01; 457/86024-02: Receiving and issuing material manufactured by a vendor removed from the Approved Bidders List (ABL). ASME Section III material (studs, nuts, rods, etc.) manufactured by a vendor removed from the ABL was received onsite and issued to site contractors. This material was procured from an ASME "material supplier" on the ABL. A significant amount of this material procured from this material supplier was shipped directly from the unapproved manufacturer's facility to the Braidwood site. The inspector reviewed the implementation of the licensee's corrective action which was delineated in Inspection Report 456/86031; 457/86024. The results of this review are as follows:

(1) The licensee's Quality Assurance Manual was revised to preclude issuance of material from a vendor removed from the ABL without a technical evaluation. Quality Procedure No. 4-2, "Evaluation of Contractor's Quality Assurance Program," and Quality Procedure No. 4-51, Procurement Document Control for Operations Processing Purchase Orders," were revised on August 12, 1986. These revisions require a technical evaluation of any material manufactured by a vendor removed from the ABL when that material was procured from a vendor on the ABL. Per discussion with the licensee's Assistant QA Manager, the technical evaluation would be required if the material manufactured by the vendor removed from the ABL was received directly from that vendor or received from the approved vendor to which the purchase order was awarded.

(2) The inspector reviewed Nonconformance Report (NCR) 679, Revision 1, which was issued to resolve technical questions with material received from Cardinal Industrial Products Corporation (CIPC). This NCR also identified the receipt of material from CIPC procured through an approved vendor when CIPC was not on the ABL. Since this NCR resulted in the reporting of a potential 50.55(e) the adequacy of the licensee's disposition of this NCR will be addressed in the NRC followup to 50.55(e) 456/86006-EE; 457/86006-EE.

(3) To determine if there are other similar cases where material was manufactured by an unapproved vendor and supplied by a vendor on the ABL, the licensee's QA Department performed Surveillance No. 6229. This surveillance did not identify any similar instances as described in this unresolved item.

This item is considered closed.

c. 10 CFR 21 Report

(Closed) 456/85003-PP; 457/85003-PP: Guyan Alloys, Inc. notified the NRC Vendor Program Branch on February 14, 1985, that Commonwealth Edison received 3/4" nominal pipe with two imperfections. The imperfections were classified as mandrel extrusion gouges. These imperfections were discovered in a 3" long pipe nipple and in one length of pipe from heat No. 783243. The licensee issued Nonconformance Report (NCR) 571 and the site mechanical contractor (Phillips Getschow) issued NCR No. 1094. NCR No. 1094 was closed based on the closure of the licensee's NCR No. 571. The disposition of NCR No. 571 required that all suspect pipe from heat No. 783243 be either replaced with new pipe or be nondestructively examined using the eddy current method, radiography, ultrasonic or visually examined using a boroscope. The final disposition of NCR No. 571 included:

- Eddy current testing of approximately 400 feet of installed pipe.
- Approximately 150 feet of the suspected pipe which was noted as being installed after the eddy current testing contractor had left the site, was cut out and replaced with new pipe.
- Six sections of the suspect pipe which could not be examined by eddy current testing because of their configurations were examined by radiography or ultrasonics.
- Two sections of pipe were visually examined using a boroscope.

The final disposition of NCR 571 was found acceptable by the inspector. This item is considered closed.

d. Safety Evaluation Report Items

(Closed) 456/8600001: Verify that an outer screen had been added to containment recirculation sump. The inspector verified that an outer screen has been installed around the containment recirculation sump. This item is closed.

(Closed) 456/8600008: "Engineered Safety Feature (ESF) Reset Controls." This SER item is the same subject as IE Bulletin No. 86006-BB and will be reviewed by an inspector from the Division of Reactor Safety; therefore, this SER item is considered closed.

(Closed) 456/8600020: Verify that the 125 VDC tie breakers between ESF buses are padlocked open and administrative control has been established to govern the operation of these breakers. When the breakers were inspected, they were found to be in use for plant testing and were not locked open. The inspector did verify that the breakers could be locked open and Braidwood Operation Procedure No. BwOP DC-7, Revision 51, "125V DC ESF Bus Cross-Tie/Restoration," was reviewed and verified to contain administrative control to assure these breakers would be locked open. During a subsequent plant tour, the breakers were verified locked open. This item is considered closed.

(Open) 456/86000024: Ensure procedures that are in place for depressurizing using PORVs include cautions to ensure the integrity of the pressurizer relief tank (Byron SER Page 5-24).

The inspector reviewed the draft changes to BwEPs No. ES-0.2, ES-0.3, and ES-0.4 which contain the required caution statement. These procedures have been reviewed and approved by an Operating Engineer and the Technical Staff. Since these are combined Byron/Braidwood procedures, the Byron on site review and station manager approvals remain. This item will be closed upon final approval and implementation of these procedures.

(Closed) 456/8600022: The SER documented a commitment by the applicant to install an automatic system to ensure adequate minimum charging pump (CV) flow to prevent deadheading that could otherwise damage the pumps. NUREG-1002, Supplement No. 1, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2, September 1986," Paragraphs 6.3.2 and 7.3.2 documented the NRR acceptance of the licensee's design. The inspector reviewed Preoperational Tests BwPT-RC-12 Retest No. 102 and BwPT-SI-12 Retest No. 90 which verified that the CV pumps mini flow isolation valves would open and close, depending on the reactor system pressure to prevent CV pump deadheading. The opening and closing of these valves occurred at the required system pressures as defined in the test procedures. This item is considered closed.

e. 10 CFR 50.55(e) Reportable Items

(Closed) 456/82009-EE; (Open) 457/82009-EE: On January 10, 1983, CECo notified the NRC of a deficiency reportable pursuant to 10 CFR 50.55(e) concerning Byron and Braidwood Figures No. 306 and 307, size 3 and size 35 pipe snubbers purchased from ITT Grinnell prior to April 1980. Size 3 snubber assemblies were reported to possibly have a pipe clamp which could interfere with the snubber and size 35 assemblies were reported to have the potential for end bracket interference with the snubber.

The licensee's final report to the NRC, dated June 2, 1983, stated the Byron snubbers were in the process of being inspected for potential interference problems and that approximately 3% had been inspected at that point in time. After writing the above final report

of June 2, 1983, the licensee received a letter from ITT Grinnell, dated June 14, 1983, which said the questionable snubbers would function properly under the reported worst case tolerances and conditions. With this information the license notified the NRC that they wished to withdraw the 10 CFR 50.55(e) report of January 10, 1983. An NRC review of the ITT Grinnell report of June 14, 1983, prompted the licensee to revert to their earlier position described in the final report letter to the NRC of June 2, 1983. The licensee resumed inspecting the snubbers in question and added other ITT Grinnell Figure No. 306 and Figure No. 307 snubber assemblies (sizes PSA 1/4, 1/2, 10 and 100) and certain ITT Grinnell Sway braces (struts) to the inspection list which were thought to also have the potential for similar interference problems.

The licensee completed the Byron snubber and sway braces inspections as documented in Inspection Reports No. 454/84051; 455/84035. The NRC closed this issue out on April 3, 1985, for Byron.

To date, the licensee has performed identical reviews of the cited snubbers and sway braces for Braidwood Unit 1 and unit common hardware using Sargent & Lundy (S&L) ECN No. 29770 criteria and Phillips Getschow (PGCo) QCP-B23B, Revision 0, dated May 19, 1986, and QCP-B23A, Revision 2 and Supplement Revision 0, dated May 19, 1986. The Braidwood Unit 2 inspections, using the above procedures, are partially complete. All Braidwood snubber and sway brace inspections have been and continue to be tracked, monitored, and documented using PGCo Corrective Action Request (CAR) No. 003. Corrective actions were verified to have been completed for Braidwood Unit 1 and unit common hardware by review of PGCo Interoffice Correspondence No. B-B-946, dated September 18, 1986, and CAR 003, Report No. 13, dated September 15, 1986. CAR 003, Report No. 13 documented that 3,363 safety-related snubbers and struts were inspected.

This resulted in a total of 406 of the 3,363 assemblies that did not meet the requirements of ECN 29770. The 406 assemblies in question were given an engineering analysis by S&L to determine if rework was required. The S&L analysis of 406 assemblies found only two that required actual rework. CAR 003 Resolution Forms No. N5-350 (dated August 16, 1986) and N5277 (dated 8-2-86) indicated rework was accomplished by grinding to reduce clamp dimension "ED" as defined in QCP-B23B, Revision 1, Figure No. 4, Page 9, to eliminate interference. Braidwood Unit 2 snubbers and sway braces fall into two classes, those that have not been installed yet and those that have been installed. Of the two classes, only the latter populace of 566 can be inspected. At the time of this inspection, about 10% of the 566 assemblies have been inspected and of these none were found to require rework or redesign. The above statements are documented in CAR 003, Report No. 14, dated September 22, 1986.

Since Braidwood Unit 1 and unit common snubbers and sway braces have been completely inspected, this issue is considered closed for Unit 1.

Similarly, this matter for Braidwood Unit 2 will be reinspected for closure when the licensee notifies the NRC that all Unit 2 inspections have been satisfactorily completed and when any required rework or design has been completed.

(Closed) 456/85004-EE; 457/85004-EE: On May 16, 1985, CECo notified the NRC of a deficiency reportable pursuant to 10 CFR 50.55(e) concerning Byron and Braidwood containment sump pump valves No. 1RF026 and 1RF027 which were determined to lack required seismic qualification. A seismically induced failure to close for either of these valves could constitute a potential radioactive material release path via the containment floor drains and the radwaste system in the auxiliary building. CECo (A. D. Miosi) letter to the NRC (J. G. Keppler), dated June 14, 1985, was used to satisfy the 10 CFR 50.55(e) 30 day and final reporting requirements.

Valve manufacturer Xomox Corporation in a May 7, 1985 letter to Sargent & Lundy (S&L) (D. W. Robinson) indicated that, based on preliminary non-site specific seismic testing, both Byron and Braidwood valves could be upgraded to meet seismic qualification requirements by simply adding a Xomox supplied tank support bracket to increase the rigidity and natural vibration frequency of the valve assemblies. Site specific seismic testing was first completed for the Byron Unit 1 valves with brackets attached and later similarly completed for Byron Unit 2 and Braidwood Units 1 and 2. S&L Component Qualification Division Report No. CQD-4391-DQSR, Page 211, documents that the Braidwood Units 1 and 2 valves received the required seismic testing. S&L (D. W. Robinson) letter to CECo (D. J. Hobson) indicates all Byron and Braidwood solenoids and limit switches used on valves No. 1RF026 and 1RF027 are exempt from additional seismic testing. The October 3, 1986, S&L (D. W. Robinson) letter to CECo (E. R. Wendorf) indicated these valves were seismically qualified with the tank support brackets installed.

Phillips Getschow Company Field Change Orders No. IRF 32781 and IRF 32783 show that the Braidwood Units 1 and 2 valve brackets were installed and approved by QC on September 8, 1986. CECo Startup Deficiency Report No. 1RF20-72 documents that the modified No. 1RF026 and 1RF027 Braidwood Units 1 and 2 valves passed functional testing in accordance with Test Procedure No. BwPT-EF-10, Revision 3, Subsections No. 9.22 and 9.60. QA approved these functional test results on September 22, 1986. This issue is closed for Braidwood Units 1 and 2.

(Closed) 456/86005-EE: During ECCS full flow testing for Byron Station, Unit 2, in February 1986, a 1 1/2 inch socket welded-elbow on a (A loop) high head cold leg injection cracked and began leaking. The elbow was replaced and the test continued. The examination concluded there were micro-cracks which had propagated through the wall.

The subject item was discussed and inspected (Braidwood Units 1 and 2) with Mr. J. Gavula, NRC regional specialist. The analysis and test results were discussed with Mr. Donovan (Sargent & Lundy (S&L)) who provided the S&L analysis report. The S&L analysis results were reviewed and found to be acceptable. The subject item has been reviewed and is satisfactory as reported in Inspection Report No. 456/86047; 457/86035 and therefore this item is considered closed.

f. Licensee Action on IE Bulletins

(Closed) 456/75006-BB; 457/75006-BB: "Defective Westinghouse Type No. OT-3 Control Switches." During 1975, Westinghouse reported to the NRC that a number of defective Westinghouse type No. OT-2 main control board electrical switches were found during initial installation at the Sequoyah Station. The defective switches are of the "spring return to neutral" type. Manual operator switch action involves rotating the switch knob clockwise or counter-clockwise, against the force generated by an internal spring which normally drives the knob back to a neutral position. Westinghouse reported that excessive internal friction caused some switches to bind and fail to return to the neutral position. Adverse tolerance stack-up on internal switch parts was blamed as the reason for the defect. Westinghouse issued Technical Bulletin No. NSD-TB-75-4, dated March 21, 1975, to document a simple test procedure to be used by various nuclear plant facilities to determine if type No. OT-2 switches used at these plants were defective. Switches not found to be defective were to be used as is; defective switches were to be replaced.

CECo NLA letter No. 75-55, dated March 11, 1985, from John F. Gudac (Superintendent Braidwood Nuclear Station) to Dennis L. Farrar (Director of Nuclear Licensing) defines where Westinghouse Type No. OT-2 switches are used at Braidwood Station. All switches, including Westinghouse No. OT-2 switches, are routinely tested equivalent to and in accordance with Westinghouse Technical Bulletin No. NSD-TB-75-4 as part of the Project Operational Analysis Departments preturnover-for-test construction tests and during Project Startup's Preoperational and system Demonstration tests. These facts are documented on CECo Startup letter (STUP) No. 86-734, dated October 3, 1986, from C. J. Tomashek (CECo Startup Superintendent, Braidwood Station) to P. L. Barnes (CECo Nuclear Licensing Administrator, Braidwood Station).

All test deficiencies discovered in the above routine tests are recorded and tracked by a computerized Deficiency Log. In response to Bulletin No. 75006, a computer search of the Deficiency Log identified a total of 84 deficiencies associated with all types of control switches. A review of these 84 deficiencies identified 13 deficiencies within the switches themselves. The switch manufacturer and type of each switch identified for each of the 13 deficiencies were verified using a Sargent & Lundy data sheet. None of the 13 deficiencies were Westinghouse type No. OT-2 switches. It can be concluded that Braidwood Units 1 and 2 do not have any defective Westinghouse type

No. OT-2 switches. Other kinds of switches found to be defective have been replaced using the Braidwood deficiency reporting system. This item is closed.

(Closed) 456/86001-BB; 457/86001-BB: "Minimum Flow Logic Problems That Could Disable RHR Pumps." This bulletin was directed to all General Electric (GE) BWR facilities and was provided to Braidwood for information. The licensee's review was that no action was required because the design of the Braidwood PWR differs from that of a BWR. The inspector inquired about the possibility of the potential for similar logic problems on ECCS pumps where a failed flow detection system could disable similar components of ECCS Trains. The licensee responded with their review of Information Notice No. 85-94 where the potential was reviewed based on events at other operating stations. Based on the foregoing, IE Bulletin No. 86001 is considered closed.

(Closed) 456/86002-BB; 457/86002-BB: "Static "O" Ring Differential Pressure Switches." This bulletin was issued following the failure of the Static "O" Ring (SOR) Differential Pressure (D/P) Switches during an event at LaSalle County Station on June 1, 1986. The licensees were requested to provide a response within seven days by submitting a report on the extent to which SOR Model No. 102 or 103 D/P switches are installed (or planned) as electrical equipment important to safety as defined in 10 CFR 50.49(b). The bulletin also provided additional instruction for those installations. Braidwood conducted a search and found there are three SOR switches installed on each Braidwood Unit, of which none are Models No. 102 or 103. These SOR D/P switches provide alarm functions only for feedwater pump gland water pressure. The inspector reviewed the licensee's method of searching and found that their search included instrument calibration sheets, Sargent & Lundy Architect Engineers equipment records, Westinghouse records, and contacting the Static "O" Ring company for verification that no Model No. 102 or 103 D/P switches had ever been ordered or delivered to Braidwood and none were planned. This bulletin is considered closed.

No violations or deviations were identified.

3. Regional Request

The resident inspector performed a review of proposed Temporary Instruction (TI) for inspection of Licensee's implementation of 10 CFR 50.62 ATWS Rule, as requested by Region III DRS. The review consisted of comparing the contents of the TI to requirements of ATWS Rule 10 CFR 50.62 and Generic Letter No. 85-06. A memo containing review comments was sent to Region III DRS for forwarding to B. K. Grimes, Director, Division of Quality Assurance, Vendor, and Technical Training Center Programs, Office of Inspection and Enforcement.

No violations or deviations were identified.

4. Construction Worker Concern

During the course of the inspection, the Senior Resident Inspector (SRI) (Operations) was contacted by a minority construction worker who expressed concern about being set up to be fired. The concerns were related to work load and associated stress that could result in a loss of quality or safety in the construction of the plant. The individual stated that he thought his particular situation was unique. After being questioned by the SRI, the individual stated he knew of no specific instance where quality or safety had been compromised.

After a telecon between the Region III Allegation Coordinator and the SRI, it was agreed that this was not an allegation within the scope of the responsibilities of the NRC. The SRI obtained the address and phone number of the Equal Employment Opportunity Commission and passed it on to the individual. The matter was also discussed with licensee construction management personnel. This issue is considered closed.

No violations or deviations were identified.

5. Allegations

(Closed) RIII-86-A-0091: "Training of Mechanical Maintenance Personnel." This issue was provided to the Senior Resident Inspector (Operations) as a concern regarding training for mechanical maintenance (MM) personnel conducted at the Production Training Center (PTC). The concern was that training was deficient in terms of: course materials (books, handouts, etc.) were not available; presentations were vague or incorrect lectures by course instructors; insufficient time to allow students to understand course materials; and inadequate feedback from instructors regarding pre-test results. This was turned over to CECo for followup by NRC letter, dated June 4, 1986.

The licensee conducted an investigation through Quality First and submitted the results to the NRC Region III by a letter, dated July 23, 1986. The results of the investigation showed that there were no safety-related issues. In addition, the letter stated that some of the instructional areas for the pilot training program required attention. More consideration was given to theory and shop work, available reference material, size and makeup of classes, and the adaptation of classes to student needs.

The inspector reviewed the licensee's evaluation Report No. QF-86-1351; which included the interview results of a number of students and management personnel, the summary of the corrective actions, and student responses to those actions. The inspector reviewed the licensee's corrective actions to the responses and found them to be acceptable. The inspector verified that this was a pilot series of courses being taught to the MM students in preparation for INPO accreditation and the need for improvements or refinement was anticipated. This allegation was substantiated in that some of the instructional areas of the pilot training program were deficient; however, the deficient areas have been corrected. Therefore, this allegation is considered closed.

(Closed) RIII-86-A-0097: On June 5, 1986, an anonymous individual called Region III and stated that on June 4, 1986, the big vessel exploded under high water pressure and four people were hurt. The caller demanded the NRC to investigate and contended that no report was made by the licensee.

This event occurred on June 4, 1986 and was reported by NRC on the same day even though there was no reporting requirement. The event involved the overpressurization of the pressurizer relief tank (PRT) with pure water, bursting the rupture diaphragm and releasing water to the containment drain system. Region III issued Preliminary Notification No. PNO-III-86-53 on June 5, 1986. The event was also described in Inspection Reports No. 456/86031/(DRP); 457/86024(DRP). The licensee assigned a special PRT Event Task Group to investigate the event, find the causes and make recommendations to the Project Manager. The inspector reviewed the task force report and found it to be accurate and with good corrective actions. These varied from completing the plant communications systems, to reemphasizing the importance of following up on alarms in the plant even though the cause may appear to be understood.

The report also described the injuries incurred, of which the worst was an individual who received a cut on the back of his head. He was treated at a hospital and released. Other complaints were bruises and/or abrasions and ringing in the ears. None of these required extensive medical treatment. This allegation was not substantiated in that even though the event was not reportable, the licensee notified Region III on the same day. Additionally, the inspectors have reviewed the event previously and in this report. This allegation is considered closed.

(Closed) RIII-86-A-0115: On July 7, 1986, an individual called the Braidwood resident's office with a concern pertaining to the independence of QA/QC from production. The individual stated that he and another QC inspector were terminated by the Phillip Getschow (PGCo) Job Superintendent on July 3, 1986. The individual stated the Job Superintendent terminated them for leaving early, even though he and the other QC inspector were still in the building. The individual believed that production should not be the organization terminating a QC inspector. The individual stated that the Job Superintendent would not sign the dismissal slip and stated the reason for termination as "violation of rules". The inspector reviewed the termination papers for the two QC inspectors. The termination paper identified the reasons for termination as being "violation of company work rule," and "violation of site work rule." A comment on the termination papers stated that the men were in Turbine 1 Track Alley ready to go home at 5:10 p.m. This comment was written by Mr. Nicholson, the Job Superintendent. The termination papers were signed by the Assistant Job Superintendent and a QC Superintendent. The inspector interviewed Mr. Nicholson concerning the circumstances pertaining to the termination of the two QC inspectors. Mr. Nicholson stated that these two individuals along with five craft individuals were noted, on July 2, 1986, as not being in their assigned work area. These individuals were in the Unit 1 Turbine Track Alley waiting for the siren indicating the end of the work day. Since this was a violation of company and site work rules, all seven men were terminated on July 3, 1986. Mr. Nicholson stated that he had discussions

with the PGCo Quality Assurance Manager the night of July 2 concerning the two QC inspectors not in their assigned work area. The QA Manager agreed that these two individuals should be terminated. The inspector interviewed the QA Manager and confirmed that he discussed the termination of the two QC inspectors with the Job Superintendent, Mr. Nicholson. The inspector also interviewed the QC Superintendent that was responsible for supervising the two QC inspectors. The QC Superintendent stated that the two QC inspectors were average inspectors and did not have any problems interfacing with craft personnel. The following is a summary of the inspectors investigation into this allegation.

- The two QC inspectors were terminated for violating company and site rules in that they left their work area before the specified time. Five craft personnel were also terminated for violating the same rule at the same time the two QC inspectors were terminated.

- Even though termination was based on the Job Superintendent's observation of the two QC inspectors being away from their assigned work area, there was discussion between the Job Superintendent and the QA Manager concerning the need for terminating the two QC inspectors for violating a site and company rule. The inspector confirmed these discussions occurred by interviewing the Job Superintendent and the QA Manager.

- The termination papers were signed by the Assistant Job Superintendent and a QC Superintendent. Discussion with the Job Superintendent revealed that there was no specific procedure/instruction on completing the termination papers. The normal practice for terminating QC personnel requires both signatures to be QA/QC personnel. However, since these two QC inspectors were terminated one day before a holiday (July 3, 1986), the QA and QC Managers were not on site to sign the termination papers. It does not appear that the signing of the termination papers by the Assistant Job Superintendent is significant since termination required two signatures, one of which was a QC Superintendent. The inspector reviewed approximately 15 other termination papers for QA/QC personnel. The termination papers in all cases were signed by two individuals from the QA/QC Department. Therefore, the co-signing of the termination papers for the two QC inspectors by the Assistant Job Superintendent is considered an anomaly caused by the terminations occurring during a holiday period.

Based on the above information, the inspector substantiated that the PGCo Job Superintendent did participate in the termination of the two QC inspectors. However, this participation was justified and included active participation by the appropriate QA/QC personnel. This allegation is considered closed.

No violations or deviations were identified.

6. Temporary License Request

On August 18, 1986, Commonwealth Edison submitted a request to the Atomic Safety and Licensing Board (ASLB) for a temporary license in accordance

with 10 CFR 50.57(c) to allow fuel loading at Braidwood Unit 1. On September 19, 1986, the ASLB gave approval to the request. In acknowledgement of the request, the NRC, NRR and Region III have submitted affidavits to affirm NRC actions on this request. In order to meet the commitments of the affidavits, the inspectors have reviewed the licensee's original request, the associated affidavits, and other related correspondence for familiarization.

Prior to the issuance of the license, the inspectors re-reviewed the entire list of previous inspection findings to assure none affecting fuel load were unresolved. The associated inspection finding results can be found in Paragraph 2. The inspectors reviewed the findings of the licensee Station Quality Assurance (QA) group for any pending issues relative to licensing. For those items of concern, the licensee was in the process of submitting an appropriate change to the FSAR and it was noted that the issues were resolved or would not affect the activities to be performed under the special 10 CFR 50.57(c) license. In addition, a Region III operator licensing inspector conducted a review of the 10 CFR 50.57(c) license request and associated documentation. A number of questions and concerns were raised; however, they were all resolved through the resident inspector's evaluation and discussion with licensee representatives.

In addition to the foregoing, and prior to the license issuance, the resident inspectors and the Braidwood Section Chief made several plant tours to assess the plant's readiness for licensing. During the tours, several observations were made, of which none specifically would have affected the license issuance. However, the licensee took prompt corrective action for assurance that these findings would not have an adverse affect on the plant during subsequent evolutions.

Most of the corrective actions were evident to the resident inspectors when they were implemented; however, on October 30, 1986, the Station Manager and both Superintendents held a meeting with the operation resident inspector and Braidwood Section Chief to reaffirm the actions.

One concern was control of personnel in the plant; the licensee implemented a program for access control to Unit 1, posted signs on appropriate doors and control panels, and issued instructions to construction personnel. This was similar to the corrective actions taken in response to an observation regarding the use of motor control centers and control panels for hanging equipment, coats, etc. Additionally, tools, tool boxes, and equipment were found in the Auxiliary Electric Room. The licensee evaluated all material and removed the unnecessary equipment from that space.

License Issuance

On October 17, 1986, the NRC issued Facility Operating License No. NPF-59, its associated Technical Specifications, and Environmental Protection Plan to Commonwealth Edison Company for Braidwood Station Unit 1. This authorized the loading of fuel and subcritical testing of Unit 1. The license condition C(10), "Inadvertent Boron Dilution," is specific to fuel

loading and prevention of approaching the shutdown margin (SDM). The resident inspectors conducted a number of inspections related to this subject.

Preparation For Fuel Loading

In preparation for fuel loading, the inspectors conducted a number of activities in this area. The license, NPF-59, Item C(10) lists a number of special measures to prevent inadvertent boron dilution of the reactor coolant system (RCS). The inspectors verified that the RCS and the makeup system sampling was performed and the boron concentration results were within proper specifications, the valves listed in Attachment 2 of the license were in their proper position and locked sufficiently to prevent inadvertent operation and that the RCS was sampled following makeup. The inspectors also reviewed the special procedures developed by the licensee to conduct these operations. The procedures reviewed were:

BwAP 300-101, Revision 2 - "Maintaining RCS Boron Concentration Greater Than 2000 PPM Prior to Issuance of a Low Power License"

BwOP CV-23, Revision 1 - "Makeup to the RWST/OB Hut"

BwOP CV-24, Revision 1 - "Makeup to the RCS"

BwOP CV-25, Revision 2 - "Transferring Boric Acid From the Batching Tank to the Bat"

BwOP CV-26, Revision 0 - "Venting RHR, Seal Water and Letdown Heat Exchangers"

BwOP CV-27, Revision 0 - "Chemical Addition to RCS"

BwCP 323-15, Revision 1 - "Special Chemistry Sampling Requirements During Fuel Load and Precritical Operation"

1BwOS XLE-D1, Revision 0 - "Boron Dilution Prevention Locked Valve Daily Surveillance"

1BwOA PRI-13, Revision 0 - "Uncontrolled Dilution With Reactor Vessel Head Removed"

1BwOA PRI-14, Revision 1 - "Uncontrolled Dilution With Reactor Vessel Head On"

During the review of the procedures, a number of items were identified by the inspectors which needed corrections, such as: inclusion of administrative controls for the SI accumulators as an alternate source of borated water, for Modes No. 6, 5, and 4; the Volume Control Tank (VCT) level; and valve position changes without restoration or being omitted from the procedure.

After discussion with licensee representatives, the licensee incorporated the changes identified by the inspectors and by themselves into the procedures. This was done prior to when the procedures would have been used. Through discussion with licensee personnel, the resident inspectors determined that the evaluations described in the special procedures were previously verified for acceptability during plant activities except for makeup to the RCS utilizing the OB Hold Up Tank (HUT). The licensee agreed to verify by testing the acceptability of the flowpath for RCS makeup utilizing the OB HUT as described in special Procedure No. BwOP CV-24, "Makeup to the RCS," prior to fuel load. The resident inspectors reviewed the results of this test and found them acceptable.

With respect to the deficiencies identified in the procedures, and considering the sensitivity of this issue, the number of deficiencies found in the procedures is considered excessive. This is considered to be a weakness in the onsite review (OSR) process in that a careful, detailed review should have identified these deficiencies prior to the implementation of the procedures. Since OSR is the subject of routine inspections, the OSR reviews will be monitored during future inspections.

The inspectors also monitored activities and status of fuel load readiness on a daily basis, including status and numbers of completed release to operations (RTOs), system test reviews, surveillances, and numbers and status of pertinent work requests.

Just prior to fuel load, the inspectors witnessed the position and locking of the valves identified in Attachment 2 of the license and the licensee's administrative control. One inspector attended a licensee training session on the 10 CFR 50.57(c) license and associated activities, and found the training to be appropriate and informative.

Other than the weaknesses identified, the process was well organized, thought out, and carried out smoothly.

Initial Fuel Load

On October 25, 1986, at 8:22 p.m., the licensee placed the first fuel assembly in the reactor vessel. This activity and the activities leading up to this occasion were witnessed by the resident inspectors.

During that time, the inspectors verified that the applicable license, Technical Specification, and procedural requirements were adhered to, and verified that required nuclear instruments were properly calibrated and operating with a measurable count rate. The inspectors verified that prerequisites were met, crew requirements were met, proper procedures were in place and being followed, verified that inverse multiplication plots were maintained, confirmed that boron concentrations were as required, verified that security access requirements were met as applicable, that refueling status boards were properly updated, that personnel understood their responsibilities, and shift schedules were within administrative guidelines.

As appropriate, the inspectors also reviewed procedure changes for technical adequacy and proper reviews, and reviewed data sheets and various logs routinely.

This activity was ongoing at the close of the inspection period and evidence showed that activities were carried out in a cautious, well planned manner. Observations will continue during the following inspection period.

No violations or deviations were identified.

7. TMI Action Plan Items

- I.A.1.1 Operating Personnel and Staffing - Shift Technical Advisor

  The staff has reviewed the licensee's program for training and use of the "Station Control Room Engineer" (SCRE) as the Shift Technical Advisor (STA). The staff concluded that this practice is satisfactory and will continue to be evaluated. The SCRE is a permanent position and part of the normal shift manning. The requirements for a STA for fuel load activities is satisfied and this matter is considered closed.

- I.A.I.2 and I.C.3 Shift Supervisor Responsibilities

  This item was open pending implementation of Procedure No. BwAP 300-1, "Conduct of Operations." The licensee has implemented BwAP 300-1; therefore, this item is considered closed.

- I.A.1.3 Shift Manning

  The licensee's overtime and minimum shift manning procedures meet the requirements of NUREG-0737 as stated in previous reports. Administrative procedure controlling these activities were implemented prior to commencing fuel load operations. This item is considered closed.

- I.B.1.2 Evaluation of Organization and Management Improvements of OL Applications

  Previous Inspection Report No. 456/86031 identified that the licensee's Nuclear Safety Department Manual did not address the Braidwood Onsite Nuclear Safety Engineering Group (ONSEG). The inspector identified that the latest revision to the manual describes the ONSEG responsibilities and identifies the Braidwood organization. This item is considered closed.

- I.C.2 Shift and Relief Turnover Procedures

  In previous Inspection Report No. 456/86031, the inspector identified that the licensee's procedures did not appear to meet the requirements of NUREG-0578. The area of interest was relative to identifying

acceptable plant parameter limits required by NUREG-0578 to be on turnover checklists. Further review of additional information supplied by the licensee showed that the turnover checklist does supply the operators with the required system parameter acceptance criteria. This item is considered closed.

- I.C.5 Feedback of Operating Experience

  As stated in Inspection Report No. 456/86041, the licensee's adequate controls for the Operating Experience Feedback System are addressed in BwAP 1260-1, "Operating Experience Review Program," and Section V of the Nuclear Safety Department's Administration Manual. This item was open pending formation of the Onsite Nuclear Safety Group (ONSEG) which is now in operation. Therefore, this item is considered closed.

- I.C.7 NSSS Vendor Review of Low Power Test Program Procedures

  The inspector reviewed correspondence between Westinghouse representatives and the licensee relative to NSSS vendor review of low power test program procedures. Documents identifying vendor review of four low power test procedures were reviewed. The documents identified vendor comments for the procedures reviewed and subsequent revisions by licensee as applicable. This item is considered closed for low power test procedures.

This completes the inspection of NUREG-0737 items required for fuel load. The items completed in this inspection pertain only to those items assigned to the resident inspectors.

No violations or deviations were identified.

8. Quality First

The licensee has established a Quality First Program to address concerns identified by various individuals associated with the Braidwood project. The programmatic controls are described in Procedure No. PM-09, "Quality First Program," Revision 3, dated June 16, 1986. This procedure was reviewed by the inspector with the following comments:

- Concerns expressed by individuals are assigned a unique number. This number is recorded in the Quality First (QF) Concerns Log along with the individual's name. This log is strictly confidential with access limited to the QF group. In this manner, the individual's identity is safeguarded from unauthorized disclosure.

- A Steering Committee consisting of the Project Manager, QF Program Director and the Assistant to the Manager of Quality Assurance provides oversight of the QF Program. Oversight of the QF Program consisted of: (1) approval of the QF Program procedures; (2) a review of QF monthly reports; (3) assessment of resolution of selected employee concerns; and (4) reports to upper management.

- The Site QA Superintendent is required to perform monthly surveillances of Record of Concerns (ROC) to determine if Quality Concerns reported to QF require further checks by QA. An ROC is the document utilized by QF to document the concern and the results of the investigation. The Site QA Superintendent also is required to perform periodic audits or surveillances of QF for conformance to procedures, acceptability of files and verification of corrective action when required.

- The procedure did not require the evaluation of concerns for their potential effect on fuel load, criticality, and power ascension. This was identified to the licensee. The licensee took immediate corrective action and established a system. This system was reviewed by the inspector and was determined acceptable. The inspector reviewed the licensee's classification of quality concerns prior to the issuance of the 50.57(c) license and agreed with the licensee's classifications. There were no quality concerns identified that were required to be resolved prior to issuance of the 50.57(c) license.

To determine the effectiveness of the QF program the inspectors reviewed ten ROCs and their supporting documentation. Of these ten, the inspectors performed a detailed followup on six ROCs to determine if the QF program's conclusions could be substantiated. The ten QF concerns reviewed were as follows:

Identification	Description of Concern	Detailed Review by NRC
QF8-1719A-Q	Approved for the Design for Installation (ADIs) are not being used on items and defects already installed in the field as an alternative to writing a nonconformance report.	No
QF-86-1509-Q	Thread engagement insufficient on Diesel Oil Storage Tank	No
QF-86-1385-Q	Design Changes at Braidwood which have not been looked at for Environmental Qualification Impact. Scope of design changes pertain to Westinghouse.	No
QF-86-1207-Q	1/4" anchors are being used for junction boxes 24" x 24" x 8"	No
QF-85-97-Q	Several Phillips Getschow Company employees had concerns about the materials, welding, inspection, and documentation dealing with the Essential Service Water System.	Yes

Identification	Description of Concern	Detailed Review by NRC
QF-85-1139-Q	The documentation of inspection work has much to be desired. The information on forms does not fully document inspections carried out. Some inspector qualifications may not be adequate for the work being done.	Yes
QF-86-803-Q	Priority is being given to hiring people with prior experience. No background checks are being made to verify experience.	Yes
QF-85-1375B-Q	If the Boric Acid Transfer Pumps are safety-related, why is the power supply to pumps nonsafety-related?	Yes
QF-86-1395-Q	Phillips Getschow welding crews told to weld through paint.	Yes
QF-85-693-Q	The control and metering connectors in the Main Control Board have braided shielding that tends to fray at the junctions where the conductors are peeled off.	Yes

The review of documentation pertaining to the four QF concerns that were not subjected to NRC followup determined that the QF program adequately addressed the identified concerns except for QF-1719A-Q. This QF concern pertained to ADIs not being used properly. The licensee's investigation of this QF concern was completed September 2, 1986. The investigation results documented on the ROC reiterated the use of the ADI as described in L.K. Comstock Procedure No. 4.2.3. The investigation, as documented on the ROC, did not identify that a determination was made to see if the ADIs were being misused. This was identified to the QF Program Director. The licensee took immediate corrective action which was reviewed by the inspector and found to be satisfactory.

In regards to the six QF concerns which were independently investigated by the inspectors to verify that the QF program was performing effective investigations, the results are as follows:

Identification No.	NRC Investigation Conclusions
QF-85-97-Q	Of the six concerns identified, three were substantiated, of which two had previously been identified and corrective action was being implemented. Another concern, which was not substantiated, required a review which uncovered a documentation problem that was addressed and

Identification No.	NRC Investigation Conclusions
	corrected. The inspector found that the findings of the QF Program investigation were correct and that the action taken corrected both the specific concerns and generic issues.
QF-85-1139-Q	After reviewing the concerns, the inspector found that they were tracked by NCRs and a 50.55(e) item. Additional documentation in the form of letters also substantiated the QF Program conclusions. A review of the dates of the documents and of the concern indicate that the problems had been identified and corrective action started prior to the QF concern. It is the conclusion of the inspector that the results of the QF Program were correct and that the action taken corrected both the specific concern and generic issues.
QF-86-803-Q	It was the conclusion of the inspector that the findings of the QF Program were misleading in that background checks were performed. No corrective action was required.
QF-85-1375B-Q	The inspector substantiated that the QF investigation was proper and satisfactorily resolved the concern.
QF-86-1395-Q	The inspector substantiated the QF conclusions and verified that there was no evidence of improper weld preparation and/or welds.
QF-85-693-Q	The inspector substantiated the QF conclusions and verified the implementation of corrective action.

Based on the inspectors independent investigation of six of the QF concerns, the QF Program appears to be effective in investigating concerns. The inspectors also verified that the site QA organization was performing audits and surveillance for conformance to procedures, acceptability of files and verification of corrective action when required.

No violations or deviations were identified.

9. Comparison of As-Built Plant to FSAR Description

This inspection was conducted in order to ascertain that selected mechanical and fluid system installations are in agreement with P&IDs contained in the current FSAR, that Draft Technical Specification surveillances could be accomplished with the as-built plant, and control and logic instrumentation of selected systems conform to the descriptions contained in the FSAR.

The systems selected were done so as to avoid duplication of effort with system walkdowns performed during construction and construction appraisal team (CAT) inspections.

For the systems reviewed, the inspectors verified that if design changes were in effect, they were reviewed, processed and implemented in accordance with appropriate procedural controls; that the cognizant test and/or system engineer was aware of changes or discrepancies; that proposed changes to the FSAR and proposed technical specifications were or would be submitted to NRR as appropriate; and by field observation that component installation, including control and instrumentation is as described in drawings, Draft Technical Specifications, and the FSAR.

With Unit 1 construction at 98% complete and the sample of Technical Specification related systems selected, the inspectors were able to determine that the facility is constructed essentially in conformance with the FSAR. Since Braidwood is a replicate of Byron, the FSAR versus Technical Specification comparison was not required at Braidwood for systems that received this review at Byron.

The following is a list of the systems reviewed and the relevant findings:

a. Reactor Trip System Instrumentation - Technical Specification No. 2.2.1

The reactor trip system was reviewed by comparing Sections 7.1 and 7.2, with accompanying tables and figures, of the FSAR, to Section 3/4.3.1, with accompanying tables, of the Technical Specification (TS). There is an apparent conflict in the reactor trip system response times given in Table No. 7.2-3 of the FSAR and Table No. 3.3-2 of the TS. The times listed in Table No. 7.2-3 are given as "typical time response." The response times used in the accident analysis are contained in Table No. 15.0-5 of the FSAR. These times agree with the times listed in the TS. No other discrepancies were noted.

Three of the fifteen types of instrumentation were selected for an in-depth inspection. They were the Overtemperature Differential Temperature, the Pressurizer High Pressure, and the Low Reactor Coolant Flow instrument channels. A review of the 21 surveillance procedures was made in comparison to the TS. No discrepancies were noted, and the inspector verified that the surveillances could be conducted, and that they would serve to meet the TS requirements.

The FSAR logic drawings were reviewed with no problems noted. Walkdowns of the systems were performed using the schematic and P&ID/C&IDs from the sensor to the reactor trip breaker. No conflicts between the drawings and the as-built plant were noted. During the walkdown the inspector verified, to the extent possible, that the necessary equipment was in place to perform the surveillance tests.

b. Boration Systems - Technical Specification No. 3/4.1.1

The Braidwood Unit 1 Boration system was reviewed by walkdown of the accessible system components and comparison with the latest revision of the applicable P&ID's (M-65, sheet No. 5A, Revision AS; M-65, sheet No. 5B, Revision AT; and M-64, sheet No. 4, Revision AU). The walkdown inspection considered all safety class I system components.

Discrepancies identified were: lines 1AB80A, 1ABJ7A, and 2ABJ7B were not heat traced and a flex cable shield was broken on control valve No. 1CV11CB in line 1CV84A2.

P&ID errors identified were:

(1) M-65, Sheet No. 5B, Revision AT:

(a) Shows line No. 1WE M4A connected below the boric acid storage tank No. 1AB03J above the diaphragm rather than below as installed. FCR L-23661 was prepared and approved for correction of the P&ID. The same condition exists on Unit 2.

(b) Line No. AB 94A does not contain a flanged fitting as shown. Safety doors must be retained when the P&ID is corrected.

(2) M-65, Sheet No. 5A, Revision AS:

Note 3 states ". . . lower loop to extend 20" below overflow connection . . . ." The 20" dimension should be changed to 9" to comply with the system design drawings and as-built system configuration.

A member of the startup group was with the inspector during the inspection and was informed of the findings.

c. Accumulators - Technical Specification No. 3/4.5.1

The safety injection accumulators were walked down and compared to the latest version of the FSAR P&ID's:

- M61, Sheet No. 5, Revision T
- M61, Sheet No. 6, Revision AK

Piping classified as Safety Class 1 was walked down at safety Class I - Class II interface and accessible portions of the accumulators.

The following discrepancies were identified:

(1) No pipe caps were on the stubs from 1SI8980C, 1SI032A and 1SI032C valves.

(2) The pressure gauge glass was broken and the needle was bent near valve No. 1SI8078D.

(3) The stainless steel conduit was not installed on the level transmitter cables for 1B accumulator.

(4) Air leaks were identified on the air regulators No. 1SI8877C, 1SI8879A, and 1SI8877D.

(5) The power cable and ground were disconnected from valve No. MOV 8808A.

(6) Ten to twenty bolts were missing from the cover of No. MOV 8808D.

d. ECCS Subsystems - Technical Specifications No. 3/4.5.2 ($T \geq 350°F$) 3/4.5.3 ($T < 350°F$)

Technical Specification (TS) Items 3/4.5.2 and 3/4.5.3 address:

- Centrifugal charging pump/system, Reactor Heat Removal Heat exchanger (RHR HX) and the RHR pump. TS No. 3/4.5.2 also addresses the SI pump/system.

The SI and RHR systems were previously walked down as part of this or earlier inspections; therefore, the centrifugal charging system is of interest during this portion of this inspection.

The Safety Class 1 piping was walked down, including all centrifugal charging system piping from the pump suction to the primary coolant system injection point.

The latest version of the FSAR P&ID's were used for reference as follows:

- M61, sheet No. 2, Revision AD
- M64, sheet No. 3A, Revision AV
- M64, sheet No. 4, Revision AU

The following discrepancies were identified and pointed out to the licensee:

(1) No valve tag on 1SI045.

(2) Motor operators removed from 1SI 8801B-2 and 1SI 8801A-1 for servicing (re-greasing) prior to fuel loading.

(3) Valve No. 1SI 8900A had a hold tag attached, No. 35349, and insulation cover not replaced so to not conceal the hold tag.

e. Residual Heat Removal System - Technical Specification No. 3/4.5.2, 3/4.5.3, and 3/4.9

The Unit 1 Residual Heat Removal (RHR) system was reviewed by a walk down of all accessible pipes, valves, vessels, pumps, and instruments and was compared with the latest version of P&ID drawing No. M-62, Revision AU of the FSAR.

Ten discrepancies were found in all; these included two broken gauge face glasses, two broken electrical conduits, one detached electrical ground cable, two sump pumps that were not electrically connected, and a number of valve labeling and tagging errors both on the P&ID drawing and on the as-built hardware. Of these, the single most common discrepancy was valve labeling on the P&ID drawing and errors in physically tagging valves in each of the Unit 1 systems.

The licensee system expert personally filed deficiency reports on each discovered discrepancy not already covered by an existing discrepancy report.

f. Containment Depressurization and Cooling System - Technical Specification No. 3/4.6.2

The containment spray system, spray additive system, and containment cooling systems were walked down to verify that the installed piping and equipment are as described in the FSAR. The installed equipment and piping were compared to the revision of the applicable P&ID drawing in the FSAR.

The following is a list of the P&IDs used:

- M61, Sheet No. 1B, Revision AN, and
  Sheet No. 4, Revision AN,

- M46, Sheet No. 1A, Revision AN,
  Sheet No. 1C, Revision AM, and
  Sheet No. 1B, Revision AM

The walkdown was an inspection of the Class 1 piping and the following discrepancies were identified:

(1) No locking devices were on valves (Normal valve locked position): 1CS016A (L.C.), 1CS004A (L.O.), 1CS004B (L.O.), 1CS012B (L.C.), 1CS021B (L.O.) was locked closed with chain and lock, 1CS046B (L.O.), 1CS018A (L.O.), 1CS017A (L.O.), 1CS001B (L.C.), 1CS017B, (L.O.), 1CS045 (L.C.), 1CS018B (L.O.), and 1CS050 (L.C.).

(2) Valve ID tag discrepancies noted were no tag on valve No. 1CS010A, no tag on valve No. 1CS043A, valve No. 1CS081B tagged as --286G, valve No. 1CS084B tagged as --9881N, and valve No. 1CS084A tagged as --9881M.

(3) As-built/P&ID discrepancies were 1CS11AB line connected to 1SI06A between strainer screen and 1CS002B valve. This is acceptable as per note 1 on drawing No. M46, sheet No. 1A, "locate at low point for drain after pump test," and pipe stub cap not shown on P&ID but is installed on valves No. 1CS081A, 1CS084A (tagged --9881M as noted above), 1CS084B (tagged --9881N as noted above), 1CS074B, and 1CS074A.

(4) Instrument air lines for valve No. 1CS010A (F.O.) were disconnected, the air line in the adjacent instrument tray was cut off and partially crimped closed; Valve No. 1CS010B (F.O.) instrument air lines were disconnected and reconnected differently than on 1CS010A.

Most, if not all, of these discrepancies were reported to the licensee system experts who verified that discrepancy reporting had been or would be submitted prior to the system or room turnover to CECo. It should be noted that these reviews were conducted prior to the licensee walkdown, and area and system turnover.

In general, it should be noted that none of the discrepancies discovered would have kept the systems from performing in accordance with the systems intended function as described within the FSAR.

g. Containment Isolation Valves - Technical Specification No. 3/4.6.3

An as-built walkdown was performed on the Containment Isolation System. Verification for agreement was made between the FSAR, Technical Specifications, and P&ID drawings. The inspector compared Table No. 6.2-58 of the FSAR with the applicable P&ID drawings and Table No. 3.6-1 of Section 3/4.6.3 of the Technical Specifications. The following is a list of valves with different closing times as specified in the FSAR and Technical Specifications.

Valve No.	Maximum Isolation Time Specified in the FSAR (Sec.)	Closure Time As Specified in the Tech Spec (Sec.)
PS 228 A	15	N.A.**
PS 229 A	15	N.A.**
PS 230 A	15	N.A.**
PS 228 B	15	N.A.**
PS 229 B	15	N.A.**
PS 230 B	15	N.A.**

**Proper valve operation will be demonstrated by verifying that the valve strokes to its required position.

The preoperational tests were reviewed and it was established that the actual isolation times were within those specified by the FSAR.

Verification of 222 valves of the total 240 containment isolation valves were made during the as-built walkdown. During the walkdown only one valve (1PS231B) was not tagged. No other deficiencies were found.

h. Component Cooling System - Technical Specification No. 3/4.7.3

The Braidwood Unit 1 Component Cooling System was reviewed by walkdown of the accessible system components and comparison with the latest revision of the applicable P&ID's (M-66, sheet No. 1A, Revision AG; M-66, sheet No. 1B, Revision AG; M-66, sheet No. 2, Revision AG; M-66, sheet No. 3A, Revision AK; M-66, sheet No. 3B, Revision AJ; M-66, sheet No. 4C, Revision AP; M-66, sheet No. 4A, Revision AP; M-66, sheet No. 4D, Revision AP; and M-65, sheet No. 3, Revision AP).

The walkdown inspection was performed on the safety class I piping systems for Braidwood Unit 1. The inspection was terminated at the Unit 1 - Unit 2 cross connect valves.

Five discrepancies were identified: packing leaks on valves No. 1CC9496A and 1CC10; no identification tag on valve No. 1CC9460B; mounting clamp on flow element No. 1FE-CC007 was not installed; and flow element No. 1FE-CC068 was not installed. None of these were of major significance and none will prevent the system from performing its intended function.

i. Essential Service Water System - Technical Specification No. 3/4.7.4

A comparison of Section 9.2.1.2, including Tables No. 9.2-1 and 9.2-2, of the Final Safety Analysis Report (FSAR) was made to Section 3/4.7.4 of the Draft Braidwood Technical Specifications (TS) for the Essential Service Water System. All of the required parameters addressed in the FSAR (valve position, valve actuation on signal and pump actuation on safety injection signal) are addressed in the TS.

There are two surveillance procedures for the Essential Service Water (SX) System. They were compared to the requirements in the TS and were found to address all of the necessary testing.

A walkdown of Unit 1 SX System was performed from the inlet valves through the coolers and back to the return line. The lake screenhouse forebay and the underground piping were not walked down due to inaccessibility. However, they were walked down during the initial TS review in early 1986. The Train B portion of the SX System, which is located on the Unit 2 side of the auxiliary building, appears to all be in place; however, considerable work remained, such as insulation and painting. It was also noted that none of the clamp-on temperature measurement devices were in place. The inspector was informed that because they are very susceptible to damage, the clamp-on temperature measurements are not installed until the "last minute". It was also noted that several valve tags were missing. The cognizant system engineer made note of the missing tags. In one

location, a flow indicator was missing. All of the piping and valves were in place, but the flow indicator had been removed. The cognizant system engineer acknowledged that deficiency. The walkdown was performed with the latest P&IDs for the SX system available on the day of the walkdown and not the drawings contained in the FSAR. During the walkdown, the inspector also verified to the extent possible, that hardware was in place to perform the surveillance tests.

j. Ultimate Heat Sink - Technical Specification No. 3/4.7.5

The ultimate heat sink, Section 3/4.7.5 of the Draft Braidwood Technical Specifications (TS) was compared to the safety requirements of the Final Safety Analysis Report (FSAR), Section 9.2.5. All of the required parameters addressed in the FSAR (average water temperature and lake level) are addressed in the TS.

There are three surveillance test procedures dealing with the Ultimate Heat Sink. Two of these are for recording the water temperature and lake level on a daily basis and the third is for determining the elevation of the lake bottom. These procedures address the surveillance requirements of the TS.

Since the Ultimate Heat Sink is a passive system there are no P&ID drawings to walk down. The inspector did locate and view the water temperature measurement (two RTDs, one just downstream of each of the two Essential Service Water Pumps) and the lake level readout on Panel OPM-01J in the control room. The level transducer, an air bubbler located upstream of the moving filters in the Lake Screenhouse, was not observed due to inaccessibility.

During the inspection, the question was raised by the inspector about the amount of water, lake level, bottom elevation, and effects of dike break. He was informed that the normal lake level is about 595 feet with the bottom of the Ultimate Heat Sink at 584 feet. If the dike does break, the level of the Ultimate Heat Sink would drop to 590 feet, at which point it loses continuity with the rest of the lake. There is a surveillance procedure in place to check the dike in accordance with Regulatory Guide 1.127, Revision 1, Position C.

It should be noted that most of the systems were walked down by the NRC prior to the licensee's final walkdown, release to operations, and area turnovers. For most deficiencies identified, it was found that the licensee had already instituted corrective actions and for the remainder, initiated corrective actions upon notification by the inspectors. In all cases, the deficiencies identified were not major and the systems could perform their function as stated in the FSAR.

No violations or deviations were identified.

10. Title 10 Requirements

This inspection was conducted to ascertain the licensee's conformance with selected Title 10 requirements during the plant testing phase.

The following selected requirements were reviewed:

a. 10 CFR 19

(1) Posting Requirements - 10 CFR 19.11

(2) Instructions to Workers - 10 CFR 19.12

b. 10 CFR 20

Storage of Licensed Material - 10 CFR 20.207

c. 10 CFR 50

(1) Construction Deficiency Reporting - 10 CFR 50.55(e)

(2) Changes, Tests and Experiments - 10 CFR 50.59

(3) Codes and standards in use correspond to the revisions required by 10 CFR 50.55(a)

The review of Title No. 10 requirements has been an ongoing process during the past year and portions were reviewed as part of each inspection. This was done through procedure reviews, NGET training, specialist inspections, followup on deficiencies, and general inspections.

No violations or deviations were identified.

11. Events Occurring Onsite During the Inspection Period

On September 3, 1986, the licensee experienced a loss of pressure on the station fire main. The cause was investigated and found to be an error in the P&ID drawing that resulted in the isolation of an incorrect valve while placing Out of Service tags for maintenance. The licensee identified and corrected the problem within 2 1/2 hours and took fcllowup steps to prevent future recurrence. This was reported to the Resident Inspectors who followed the investigation and relayed the information to Region III. At the time, the greatest concern was protection of the new fuel stored onsite; however, 24 hour security/fire watches are established in the new fuel vault and storage area of the spent fuel pool. The licensee generated a deviation report and a copy will be forwarded to the resident inspectors for review.

Upon issuance of the license NPF-59, the reporting requirements were implemented pursuant to 10 CFR 50.72 and 10 CFR 50.73. This was also the instrument for implementation of the licensee's deviation reporting (DVR) system. At the writing of this report, any DVRs that would have followup licensee event reports (LERs) had not been written due to not reaching the 30 day reporting requirement. In future inspections, LERs will be reviewed and evaluated as they are received.

No violations or deviations were identified.

12. Commissioner Bernthal - Site Visit

On October 21, 1986, Commissioner Bernthal and two advisors were onsite for meetings with the resident inspectors, members of the regional staff, and the licensee. A plant tour was conducted and an exercise on the Byron/Braidwood simulator at the Production Training Center was observed.

During the meetings, discussions were on general NRC topics and how they relate to Braidwood and other issues specific to Braidwood. Matters of interest discussed were the plant construction and operational status, replication with Byron, lessons learned, station organization and shift of management control during startup and power ascension, training and experience of station and corporate personnel, licensee initiatives, corporate overview of a large commitment to nuclear power, performance indicators and fitness for duty.

Commissioner Bernthal commented that the plant was well along and acknowledged that some work remained. He expressed approval of the licensee's model spaces program and emphasized the importance of good housekeeping procedures in relation to plant performance.

13. Operational Staffing Inspection

Through discussions with licensee personnel, review of personnel work experience, ANSI 18.1, and inspection reports, the inspector has determined that the licensee's operating staff meets the qualification requirements of Inspection Module No. 36301, "Operational Staffing Inspection." The inspector reviewed past work experience for several operations and maintenance and inspection personnel. The inspector also verified that all staff positions are filled and that the licensee has training programs in place to assure that personnel receive the required training applicable to their staff position. Additionally, licensee quality assurance audits in this area were reviewed and they also determined that the operating staff personnel meet the requirements of ANSI 18.1, "Selection and Training of Nuclear Power Plant Personnel."

No violations or deviations were identified.

14. Plant Tours and Independent Assessments

The inspectors conducted routine plant tours during the inspection period to make an independent assessment of equipment conditions, plant conditions, construction activities, security, fire protection, general personnel safety, housekeeping, and adherence to applicable regulatory requirements. During the tours, the inspector reviewed various logs, daily orders, interviewed personnel, attended shift briefings and plan of the day meetings, witnessed various construction work activities, and independently determined equipment status. During the shift changes, the inspector observed operator and shift engineer turnovers and panel walkdowns.

During the week of September 15, 1986, the inspector noted that during the morning shift change meetings, there was no one present from the Operational Analysis Department or System Operational Analysis Department

(OAD or SOAD). This situation was discussed in previous Inspection Report 456/85045; 457/85044, Paragraph 7, and was corrected for a period of time. The shift change meetings are a time where all persons conducting some function in the plant that could affect operations or equipment status, either directly or indirectly, provide explanations such that all personnel on shift could be made aware of the activities planned. This does not preclude direct and timely communications with shift operations personnel at the time that an evolution is to occur, but is to serve as an informational session to all personnel. In light of previous activities at Braidwood as well as enforcement actions at other CECo stations, the inspector relayed his concerns to station operations management personnel, OAD, and station startup management personnel. They acknowledged the inspector's concerns by relaying the concern to project startup, onsite OAD/SOAD management, placing instructions in the daily orders, and reinstructing shift engineers to assure input from OAD/SOAD. They are now requiring a more detailed input from the OAD/SOAD representative during the morning meetings. This could help preclude any events during the early stages of licensed activities. The inspectors will continue to follow communications as part of the normal inspection activities.

No violations or deviations were identified.

15. Pressurizer Code Safeties

While performing a procedure review relative to testing requirements of the pressurizer code safety valves, Technical Specification (TS), Section 3.4.2.2 was reviewed. The inspector determined that the licensee's procedure did not test the safeties as required by the TS. The TS states that the lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. The method used by the licensee did not include the valve ambient operating pressure and temperature parameters. Subsequent to the inspectors initial discussions with licensee personnel on this matter, the testing of the code safeties was contracted out and the testing and results conform to the TS requirements. The inspector has no further concerns regarding this issue.

No violations or deviations were identified.

16. Meetings, Training, and Other Activities

NRR/IE Site Visit

On October 14, 1986, a site visit was made by a party from the NRC Headquarters offices representing NRR, I&E, Region III, and the resident inspectors. The purpose of the visit was to meet with members of the licensee's corporate and site staffs. The meeting included a presentation by the licensee on plant status/schedule; project objectives including preoperational test program status; N-5 code data reports; surveillance status; open items status; area completion; security; emergency preparedness; system walkdowns; and NRC open items. The schedule also included a plant tour.

Licensing Status Meetings

On September 12 and 30, 1986, licensing status meetings were held onsite between NRC Region III and Commonwealth Edison. The purpose of these meetings was for the licensee to provide an update with respect to readiness for fuel load. Several members of the Region III staff were present at the September 12 meeting for a comparison with Byron Unit 2 and Clinton in order to plan allocation of agency resources.

Additional meetings were held on September 18, and October 16, 1986, between the CECo Project Manager, the Region III Project Director, and members of each of their staffs. The meetings provided an opportunity to discuss the licensee's list of items that must be dispositioned prior to loading fuel in Unit 1. Another matter discussed was the number of open items that were considered acceptable for fuel load.

Contact by Union Steward

On September 26, 1986, the CECo Mechanical Maintenance (MM) Union Steward contacted the Senior Resident Inspector (Operations) (SRI) with a concern relative to qualifications of individuals issuing controlled tools for safety-related work in the plant. The concern was that the persons in the tool issue cage (B level mechanics) had insufficient credentials for that work and he understood that, by administrative procedure, they should be at the "A" level. The SRI contacted a number of people in the MM management chain and found that there was no requirement on the level (A or B) of the person assigned to issue controlled tools; however, a helper could not perform the function. In addition, the inspector found that everyone assigned to the tool issue position receives on-the-job training (OJT) for tool issues prior to being assigned that position unassisted.

It was also found that an "A" level mechanic is required to conduct tool calibrations, e.g. micrometer, dial indicator, torque wrench tests, etc.

An NRC Region III inspection was recently conducted in this area and tool issue and retrieval for safety-related work and the associated record keeping were found to be acceptable.

The SRI relayed this information to the union steward who acknowledged the findings and stated that he found them to be acceptable. Based on the foregoing no further inspection activity is anticipated on this matter.

Contact by Outside Agency

During the week of September 29, 1986, the Senior Resident Inspector (Operations) was contacted by two members of the staff of MHB Technical Associates (Minor, Hubbard, and Biedenbaugh), a California based consulting organization who stated they were under contract with the Attorney General of the State of Illinois for followup and technical assessment of events and activities at nuclear plants in Illinois. They stated that this was an initial contact to establish a continuing line of communication with the resident inspectors and planned to do the same at other nuclear stations in

Illinois and with NRR. Their initial interest in Braidwood was upcoming significant dates. The SRI relayed this information to Region III. A confirmatory call to the Attorney General's Office of Illinois from Region III confirmed the association of MHB Technical Associates and the Attorney General. During the conversation, the two individuals stated that they had worked for the Union of Concerned Scientists (UCS) in Washington, D.C. on Indian Point issues; had worked for the California Public Utilities Commission on Diablo Canyon; and had worked on Shoreham issues. They also stated that the three individuals (Minor, Hubbard, and Biedenbaugh) were involved in controversial resignations from General Electric, San Jose, California in 1976 that received considerable media interest.

A memorandum dated October 8, 1986, from the Region III Director of Division of Reactor Projects, provided guidance to all SRI's at nuclear plants in Illinois. The memo stated that future communications with MHB Technical Associates should be conducted through the Region III State Liaison Office in order to provide continuity.

17. Exit Interview

The inspector met with licensee and contractor representatives denoted in Paragraph 1 during and at the conclusion of the inspection on October 30, 1986. The inspector 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.