U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-454/90021(DRP); -455/90019(DRP)

Docket Nos. 50-454; 50-455 License Nos. NPF-37; NPF-66

Licensee: Commonwealth Edison Company
Opus West III
1400 Opus Place - Suite 300
Downers Grove, IL 60515

Facility Name: Byron Station, Units 1 and 2

Inspection At: Byron Site, Byron, Illinois

Inspection Conducted: August 12 through October 2, 1990

Inspectors: W. J. Kropp
R. N. Sutphin
T. Kobetz
D. Calhoun

Approved By: WDShafer for
Martin J. Farber, Chief
Reactor Projects Section 1A

10/17/90
Date

Inspection Summary

Inspection from August 12 through September 30, 1990 (Reports No. 50-454/90021(DRP); No. 50-455/90019(DRP))
Areas Inspected: Routine, unannounced safety inspection by the resident inspectors of action on previous inspection findings; operational safety, reactor startup, onsite event followup, current material condition, radiological controls, security, licensee event reports, deviation reports, maintenance activities, surveillance activities, estimated critical conditions, auto-start of 2A AFW pump and ability of AFW isolation valves to close.
Results: Of the fourteen areas inspected, no violations were identified. Two unresolved items pertaining to the inoperability of AFW due to a strut removal (Paragraph 4.a(1)) and various fuel movement problems (Paragraph 7) were identified. One Open Item was identified that pertained to a radiological release outside containment (Paragraph 3.c.(4)). The following is a summary of the licensee's performance during this inspection period:

9010300352 901018
PDR ADOCK 05000454
Q PDC

Plant Operations

The licensee's performance in this area was mixed during this inspection period. During RCS draining activities the Unit 2 operator noted disparity between the installed level indicator and the tygon level of about three feet. Draining activities were suspended until the cause was determined. The cause was suspected to be air entrapment in the tygon tubing. The air was removed from the tubing and the RCS level was then lowered 1/2 feet to ensure both level indications were tracking properly. RCS draining was then resumed. The Unit operator's action was considered effective and indicated attentiveness during a plant evolution that has caused problems in the past. However, during this inspection period equipment operators inadvertently deenergized the Unit 2 instrument inverters during a Refueling Outage Surveillance. The deenergization of instrument inverters caused a containment isolation valve for instrument air to automatically close.

Safety Assessment/Quality Verification

The licensee's performance in this area was considered mixed during this inspection period. Overall, the quality of the LERs in the area of root cause and corrective action was considered good. However, the root cause for one LER that pertained to the inoperability of train "A" of AFW due to a strut removed during a modification was identified as an Unresolved Item. The inspectors have a concern that all the root causes had not been sufficiently identified in the LER. Also, the inspectors have a concern with the increase in the number of personnel errors that have occurred over the last several weeks. The personnel errors were documented in LERs and other lower tier documents. The inspectors were concerned that the increased number of personnel errors prior to the Unit 2 refueling outage could adversely affect the licensee's performance during the outage. Licensee's management took prompt action when the inspectors expressed their concern.

Maintenance/Surveillance

The licensee's performance in this area was considered mixed during this inspection period. Overall, the maintenance activities associated with the 5 year inspection of the 2A DG, the troubleshooting of the ATWS mitigation system and the work associated with the RCFC breaker were considered good. However, there has been an increase in the number of problems in equipment and maintenance activities that appear to be attributed to lack of attention to detail. The licensee also had problems in the performance of surveillance activities due to lack of attention to detail as evidenced by LERs 454/90009 and 455/90004.

Engineering/Technical Support

The licensee's performance in this area was mixed during this inspection period. The engineering reviews that pertained to Generic Letter 89-10 appeared thorough as demonstrated by the issue identified with the ability of the AFW isolation valves (AF013-(A-H)) to close with the associated steam generator faulted. The licensee's response to the issue was timely and required good interface between operations and the technical staff. However, the aborted reactor startup on August 19, 1990 and the unexpected auto start of the 2A AFW pump were examples of the station's technical staff's inattentiveness to detail.

In conclusion, the inspection considered the licensee's overall performance during this inspection period at a level not commensurate with past performance. Several of the events and problems identified appeared to be caused by lack of attention to detail in the operating, maintenance, health physics and technical support areas. Even though the event or problems by themselves were not considered safety significant, the inspectors were concerned that the lack of attention to detail was evident in several areas of plant activities.

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

*R. Pleniewicz, Station Manager
K. Schwartz, Production Superintendent
R. Ward, Technical Superintendent
*J. Kudalis, Service Director
D. Brindle, Operating Engineer, Administration
T. Didier, Operating Engineer, Unit 0
T. Gierich, Operating Engineer, Unit 2
*T. Higgins, Assistant Superintendent, Operating
J. Schrock, Operating Engineer, Unit 1
*P. Johnson, Tech Staff Superintendent
*M. Snow, Regulatory Assurance Supervisor
D. St. Clair, Assistant Superintendent, Work Planning
*T. Tulon, Assistant Superintendent, Maintenance
D. Winchester, Quality Assurance Superintendent
*M. Rauckhorst, ENC Project Engineer
*E. Zittle, Regulatory Assurance Staff

*Denotes those attending the exit interview conducted on October 2, 1990, and at other times throughout the inspection period.

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

2. Action on Previous Inspection Findings (92701 & 92702)

(Closed) 454/90017-05; 455/90016-05: The as-built weld configuration assessment to determine acceptability of the welds based on specific seismic criteria for Byron. The inspectors reviewed the licensee's documentation of the assessment of the as-built configuration and identified no problems. The conclusion of the engineering evaluation stated that all as-built weld sizes for the centrifugal charging and safety injection pumps were adequate for the plant specific loads.

3. Plant Operations

Unit 1 operated at power levels up to 100% in the load following mode until August 19, 1990 when the unit tripped from 78% reactor power caused by a "power range flux negative rate high". At the time, Byron station was experiencing heavy thunderstorms and lightning in the area. Lightning appeared to have struck the Unit 1 containment which caused all but one rod control power supply to trip. Loss of the power supplies resulted in the insertion of several control rod banks which resulted in the high negative flux trip signal. Unit 1 was returned to service on August 20, 1990 and has operated up to 100% reactor power in the load following mode.

Unit 2 operated at power levels commensurate with coastdown limits. The unit was taken off line at 2:00 a.m. (CDT) on September 1, 1990, to begin a scheduled 59 day refueling outage. Planned activities include: ILRT-pressurization of the reactor containment building, main generator winding module work, eddy current and sludge lance work on the steam generators, refueling of the reactor core, 5 year inspection of the 2A diesel generator, 18 month inspection of the 2B diesel generator, replacement of the 2A low pressure turbine rotor, miscellaneous primary and secondary work, and 43 modifications. The licensee expects the unit to be returned to service October 30, 1990.

a. Operational Safety (71707)

During the inspection period, the inspectors verified that the facility was being operated in conformance with the licenses and regulatory requirements and the licensee's management responsibilities were effectively carried out for safe operation. Verification was based on routine direct observation of activities and equipment performance, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety system status and limiting conditions for operation action requirements (LCOARs), corrective action, and review of facility records.

On a sampling basis the inspectors daily verified proper control room staffing and access, operator behavior, and coordination of plant activities with ongoing control room operations; verified operator adherence with the latest revisions of procedures for ongoing activities; verified operation as required by Technical Specifications (TS); including compliance with LCOARs, with emphasis on engineered safety features (ESF) and ESF electrical alignment and valve positions; monitored instrumentation recorder traces and duplicate channels for abnormalities; verified status of various lit annunciators for operator understanding, off-normal condition, and compensatory actions; examined nuclear instrumentation (NI) and other protection channels for proper operability; reviewed radiation monitors and stack monitors for abnormal conditions; verified that onsite and offsite power was available as required; observed the frequency of plant/control room visits by the station manager, superintendents, assistant operations superintendent, and other managers; and observed the Safety Parameter Display System (SPDS) for operability. No problems were noted.

During the draining of the Unit 2 reactor coolant system (RCS) in preparation for head removal, the operators noted disparity between the installed vessel level indicator and the tygon tube level indicator of about three feet. The operators immediately secured draining of the RCS until the discrepancy was resolved. The problem appeared to be an air bubble entrapped in the tygon tubing. After removal of the air bubble, the operators drained the RCS about 1/2 foot to ensure both indications were in agreement and functional. After performing this evolution, the operators resumed draining the RCS with no further problems noted.

b. Reactor Startup (71707)

On August 19, 1990 at 8:45 p.m. the licensee commenced a Unit 1 reactor startup. The startup was aborted when the operators determined that the rod position for criticality would be outside the 500 pcm administrative limit. The licensee has had two other aborted reactor startups (November 20, 1989 and February 11, 1990) on Unit 2 when the rod position for criticality was predicted to be outside the 500 pcm administrative limit. The licensee investigated the reason for the error in the estimated critical condition for the August 19, 1990 reactor startup and determined the cause to be different than for the other two aborted startups. For further details see Section 6.a of this report. The licensee commenced another reactor startup at 10:57 a.m. on August 20, 1990. The inspectors monitored the approach to criticality which occurred at 106 steps on control bank "D". The approach to criticality was good with good interface with, and support from, the station's nuclear engineers.

c. Onsite Event Follow-up (93702)

(1) On August 19, 1990, at 4:25 a.m., Unit 1 tripped from 78% reactor power. The cause of the trip was an automatic reactor protection system signal of "power range flux negative rate high." At the time, Byron Station was experiencing heavy thunderstorms and lightning in the area. Several station personnel reported seeing lightning strike the Unit 1 containment and cooling tower. Inspection of the rod control system identified that all but one power supply had tripped, which caused Control Banks "B" and "D" and Shutdown Banks "B", "C", "D", and "E" to insert into the core initiating a high negative flux rate and a reactor trip.

(2) On August 18, 1990, at 2:47 a.m., the 1A AFW pump auto-started unexpectedly and ran for approximately one minute. The auto-start signal was determined to originate from the ATWS mitigation system. The licensee bypassed the ATWS system and the 1A AFW pump was returned to service. For further details see Section 6.b of this report. Both Unit 1 AFW pumps auto-started on SG low levels as expected during the Unit 1 trip on August 19, 1990. However, Sequence Events Recorder (SER) did not record the auto start of the 1A AFW pump. Investigation by the licensee determined there was a loose connection between the interface with the annunciator circuit and the SER which was corrected by the licensee.

(3) On September 3, 1990, an unexpected auto safety injection (SI) signal caused instrument air containment isolation valve, 2IA065, to close. At the time, the licensee was performing the Train "A" manual SI and phase "A" containment isolation surveillance, 2BOS3.2.1.1.a-1. Prior to the SI signal, the Train "A" 480 Vac bus 231X was de-energized to remove The 2C Reactor Containment Fan Cooler (RCFC) low speed breaker, which had failed to auto close during the surveillance and was manually closed by the operators. Due to high current conditions (335 amps), attempts were made to open the 2C RCFC low speed breaker from the control room and locally without success. Subsequently, bus 231X was de-energized to remove the 2C RCFC low speed breaker. Instrument inverters 211 and 213 were then inadvertently de-energized by plant personnel which resulted in the loss of the low steamline pressure SI blocks. The manual SI signal had been reset per the surveillance with all Train "A" equipment with the exception of 2IA065 in the SI activated state. When the "A" reactor trip breaker was closed, the P-4 signal was reset and the auto SI occurred.

(4) On September 7, 1990, at approximately 4:00 a.m., it was discovered that an unplanned radiological release had occurred on Unit 2. The licensee was in day 6 of a 59 day refueling outage and in the process of pressurizing containment to 45 psig to perform integrated leak rate testing (ILRT). Pressurization of containment began at 8:00 p.m. on September 6, 1990, after ILRT final valve lineups had been completed. The release was identified by a Tech Staff member while performing rounds. The source of the release was a leak in the 2A steam generator (SG) secondary manway cover. The release path was through the SG secondary manway cover, through the Main Steam Line, then through valve 2MS5014A, which was vented open to atmosphere per ILRT valve lineups. The licensee's immediate corrective actions were to close main steam isolation valve, 2MS001, place a pressure gauge on the line, and obtain samples of containment. In addition, communications were established between the Tech Staff and Radiation Protection departments. The licensee's calculated total release was 8.42 microcuries total, which was broken down into 8.38 microcuries of noble gas, .070 microcuries of iodine and 38.2 microcuries of tritium. The release was well below NRC regulations. The licensee informed the resident inspector's office of the release and issued a DVR for the event. Prior to the Unit 2 shutdown, the licensee was aware that the 2A SG secondary manway had a leak. This matter is considered an Open Item pending further review by the NRC (455/90019-04(DRP)).

(5) On September 9, 1990, the licensee informed the NRC that during engineering reviews associated with Generic Letter 89-10, a discrepancy was identified with the ability of Auxiliary feedwater (AFW) isolation valves, AF013s, to isolate AFW to a faulted steam generator under certain postulated conditions. For further details see Section 6.c of this report.

(6) On September 28, 1990, at 9:30 p.m., with Unit 2 in Mode 6 with no fuel in the reactor vessel, the licensee discovered that the intermediate head safety injection (SI) throttle valve for loop "A" was closed. The licensee discovered the mispositioned valve, 2SI8822A, while performing a 18 month technical specification surveillance to verify proper stroke time for SI valves. The intermediate head SI flow to the other three loops was at the correct value, greater than or equal to 146.4 gpm per loop with loop "A" flow indicating "0" gpm. Preliminary investigation by the licensee determined that maintenance work had been performed since the surveillance was performed the previous outage in January - February, 1989. The licensee is also reviewing records to determine if the valve was used for isolation purposes subsequent to the successful 1989 surveillance. Since valve, 2SI8822A, is required to be open very little (approximately .2 inches) to achieve the correct intermediate head SI flow to loop "A", a micrometer was used to position the valve. The licensee is reviewing any previous work records on valve 2SI8822A, to determine if the valve was repositioned to the correct throttle setting after the completion of the work. The licensee inspected all throttle valves for ECCS injection for Unit 1 which is presently operating in the load following mode and found all twelve valves to have the correct throttle setting except valve 1SI8810A, the high head cold leg injection for loop "A". The valve throttle setting was .1 inches more than required by procedure. The licensee considers the ECCS systems operable for Unit 1. The inspectors will review the LER for proper root cause and corrective actions.

(7) On September 29, 1990, at 9:34 p.m., with Unit 1 in the load following mode and Unit 2 in a refueling outage with no fuel in the reactor vessel, a Unit 2 spent fuel assembly slipped out of the basket used for fuel reconstitution in the spent fuel pool. This event occurred following reconstitution of the fuel assembly when the basket holding the assembly was being rotated to the upright position. See Section 7 of this report for further details. The bottom door of the basket opened, allowing the fuel assembly to slip out. The fuel assembly came to rest on top of a empty spent fuel rack at a 45 degree angle with approximately 4 grids outside the basket and the remaining assembly in the basket. The licensee inspected the assembly with a video camera and observed no bubbles. It appears the fuel assembly struck the spent fuel rack at the #2 grid. The #2 grid appears to have severed in one location. The licensee placed a probe in the water set to alarm at 5 mrem above background. A sample of water around the assembly was obtained with no indication of increased radiation levels. All fuel moves were suspended and the fuel building HVAC system was placed in the charcoal absorption mode as a precautionary measure. A Bus 242 outage planned over the weekend was

postponed until resolution of this event. The licensee worked with Westinghouse to obtain instructions on how to recover the fuel assembly and on September 30, 1990, the licensee was successful in recovering the assembly. The fuel assembly is presently in the reconstitution basket in the vertical position with the bottom in the up position. The licensee plans to test the securing dogs on the basket door prior to attempt[illegible] to rotate the basket to remove the assembly for storage in the spent fuel rack. The resident inspectors will conduct an independent investigation and review the results of the licensee's investigation.

d. Current Material Condition (71707)

The inspectors performed general plant as well as selected system and component walkdowns to assess the general and specific material condition of the plant, to verify that Nuclear Work Requests (NWRs) had been initiated for identified equipment problems, and to evaluate housekeeping. Walkdowns included an assessment of the buildings, components, and systems for proper identification and tagging, accessibility, fire and security door integrity, scaffolding, radiological controls, and any unusual conditions. Unusual conditions included but were not limited to water, oil, or other liquids on the floor or equipment; indications of leakage through ceiling, walls or floors; loose insulation; corrosion; excessive noise; unusual temperatures; and abnormal ventilation and lighting. The walkdowns of the plant identified concerns with the 2A Auxiliary Feedwater (AFW) systems when on August 17, 1990, the inspectors noted a pipe strut on scaffolding by the 2A AFW pump. For further details see Section 4.a of this report.

The material condition of Unit 1 during this inspection period continues to be considered good overall with some large steam leaks noted in the turbine building. The steam leaks were identified for repair with a schedule consistent with plant operations. The material condition of Unit 2 prior to shutdown for a refueling outage on September 1, 1990 was also considered good. Housekeeping in the plant was considered satisfactory and consistent with a plant in a refueling outage.

e. Radiological Controls (71707)

The inspectors verified that personnel were following health physics procedures for dosimetry, protective clothing, frisking, posting, etc. and randomly examined radiation protection instrumentation for use, operability, and calibration.

f. Security (81064)

Each week during routine activities or tours, the inspectors monitored the licensee's security program to ensure that observed actions were being implemented according to the approved security plan. The inspectors noted that persons within the protected area displayed proper photo-identification badges and those individuals requiring escorts were properly escorted. The inspectors also verified that checked vital areas were locked and alarmed. Additionally, the inspectors also verified that observed personnel and packages entering the protected area were searched by appropriate equipment or by hand.

No violations or deviations were identified.

4. Safety Assessment/Quality Verification (40500, 90712, 92700)

a. Licensee Event Report (LER) Follow-up (90712, 92700)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, that immediate corrective action was accomplished, and that corrective action to prevent recurrence had been or would be accomplished in accordance with Technical Specifications (TS):

(Closed) 454/90008-LL: Plant shutdown required by Technical Specifications due to high containment air temperature. For further details see Paragraph 2.c in Inspection Report 454/90014; 455/90013.

(Closed) 454/90009-LL: Grab samples of effluent releases of station blowdown exceeded Technical Specification requirement of every 12 hours with radiation monitor, ORE-PR010J, inoperable. Due to a communication problem between a chemistry technician and a health physicist, a grab sample was missed. A contributory factor also was the untimely managements review of the LCOAR package due to cognitive personnel error by the Health Physics Laboratory Supervisor. A similar event occurred in 1988 (LER 455/88-010) that involved personnel error. The corrective action for LER 454/90009 should preclude recurrence of a similar event.

(Closed) 454/90010-LL: An unexpected autostart of the 1A Auxiliary Feedwater pump caused by the energization of a relay in the Anticipated Transient Without Scram (ATWS) Mitigation Systems without the necessary logic signals present. For further details see Section 6.b of this report.

(Closed) 454/90011-LL: Unit 1 reactor trip occurred from 78% reactor power. A lightning strike induced a voltage surge that activated nine out of ten over-voltage protection devices installed on power supplies in the rod drive power cabinets. The activation released twelve out of fifteen rod control cluster assembly groups into the core and resulted in a high negative flux rate reactor trip. Due to several Commonwealth Edison and industry wide lightning induced reactor trips, several modifications have previously been installed to both the containment lightning protection system and the rod drive over-voltage protectors. Additional enhancement will be pursued and will be documented in a supplemental report.

(Closed) 455/90002-LL: Unexpected P-4 feedwater isolation signal from Train "B" reactor trip breaker. See paragraph 3.c.2 of inspection report 454/90017; 455/90016 for further details. The specific cause for this event could not be identified. The probable cause was identified as the cell switch that houses the 33a contacts. The excessive play in the actuation arm and two split terminal lugs probably contributed to the intermittent feedwater isolation signal. The licensee initiated a Nuclear Work Request to replace the cell switch and repaired the split terminal lugs. The inadvertent feedwater isolation signal could not be repeated and the Train "B" reactor trip breaker was returned to service.

(Closed) 455/90003-LL: Feedwater isolation occurred when the 2A steam generator narrow range level reached the high-high water level setpoint. See paragraph 3.d of inspection report 454/90017; 455/90016 for further details.

(Closed) 455/90004-LL: Valve stroke surveillance was not performed at the required frequency. Stroke test data from the May 24, 1990 surveillance for valve 2PS9356B (process sampling containment isolation valve) indicated an increase of greater than 50% over the last stroke time which required an increased test frequency per ASME Section XI.

(Closed) 455/90005-LL: Pipe supports on the 2A Auxiliary Feedwater (AFW) pump essential service suction pipe had been replaced without Operating Department concurrence. This condition was identified by the licensee after the resident inspector's inquiry about a strut laying on scaffolding by the 2A AFW pump.

Based on the review of the above LERs and other less significant events, the inspectors identified the following two concerns:

(1) The event described in LER 455/90005 was the result of an inquiry by the resident inspector about a removed strut from the Essential Service (SX) suction pipe to the 2A AFW pump. The system engineer determined during a walkdown that a support appeared to have been replaced and notified the Shift Control Room Engineer (SCRE). The SCRE determined that control room personnel were not aware of the scope of the work in progress on the AFW system and had therefore, not entered a Technical Specification (TS) Limiting Condition for Operation Action Requirement (LCOAR). The 2A AFW pump was immediately declared inoperable and the appropriate TS LCOAR was entered. The strut replacement was performed during the installation of modification M6-2-88-060, that included installation of a flushing line, with associated isolation valves and supports. In addition, existing supports were to be modified to support the additional weight of the new pipe. Subsequent to the identification of the removed strut, the Architect/Engineer, Sargent & Lundy (S&L) performed calculations to determine the operability of the AFW system during installation of modification M6-2-88-060. The results of S&L's evaluation of the pipe support installation sequence and scaffolding loads indicated that the normal operating loads were

within code allowable values. However, code allowable values were exceeded for design basis load combinations (seismic) during the time supports M-2AF03021R and M-2AF03019R were individually removed. Therefore, the 2A AFW pump should have been declared inoperable on August 8, 1990 and returned to an operable status within 72 hours since the plant was operated in a condition outside the design basis of the plant. In addition to the strut removal issue, LER 455/90005 also addressed two other issues that related to the installation of modification M-6-2-88-060. The two issues were scaffolding attached to a safety related pipe and loose load bearing nuts on component support M-2AF03021R.

The inspectors' review of the licensee's root cause analysis and corrective action documented in LER 455/90005 identified a potential concern. The root cause for the supports being removed was identified by the licensee as personnel error by the installation contractor's foreman for not requesting an Out-of-Service (OOS) prior to performing work on existing component supports of an operational system. Per procedure BAP 330-1, "Station Equipment Out-of-Service Procedure", an operability review would have recognized the proper time restraints for rendering the AFW system inoperable. The contractor foreman incorrectly assumed that the existing supports had no effect on the operating system pressure boundary and therefore did not require an OOS. The inspectors reviewed the Modification Review Checklist for modification M6-2-88-060. Part E of the checklist identified that an outage (Mode 5) was required for installation requirements. The inspectors were concerned with the work planning controls to ensure that modifications identified for outages and rescheduled for pre-outage work have been adequately reviewed by plant personnel for affect on operability. The inspectors had no concerns with the root cause analysis and corrective actions for the scaffolding attached to safety related piping and the loose bearing nuts on support M-2AF03021R. Pending further review of the work planning and modification process by the NRC, the root cause analysis and corrective action for the strut removals that caused an inoperable 2A AFW pump is considered an Unresolved Item (455/90019-02(DRP)).

(2) Four of the eight LERs identified above involved personnel error and/or lack of attention to detail. (LERs 454/90009; 454/90010; 455/90004 and 455/90005). In addition to the four LERs, there have been the following other incidents that have occurred due to lack of attention to detail:

(a) missed criticality prediction on Unit 1 startup on August 19, 1990.

(b) unauthorized removal of a temporary alteration documented in DR 90-0163.

(c) work instructions exceeded without authorization documented in DR-90-0169.

(d) installation of incorrect diaphragms on Unit 1 pressurizer power operated relief valve air operators in June 1990.

(e) damage to the Unit 2 Turbine building 10 ton crane caused by an incorrect temp lift of an OOS.

(f) unexpected closure of a containment isolation valve for instrument air caused by the instrument inverters mistakenly being deenergized.

(g) failure to have a radiation protection technician enter Unit 2 containment for an initial maintenance activity as documented in Radiation Occurrence Report 90-012.

(h) Fault on a 1B condensate/condensate booster pump caused by wrong size lugs installed during maintenance activities.

Early in the inspection period, the resident inspectors expressed a concern to plant management with the increased number of events caused by lack of attention to detail. The station's performance in the past three years in this area has been good with few events caused by lack of attention to detail. At this time, the inspectors have not considered the increased number of events/occurrences due to a lack of attention to detail as a negative trend. However, since Unit 2 was scheduled for a refueling outage commencing September 1, 1990, the inspectors were concerned that the number of events/incidents could affect the station's overall performance during the outage and increased management attention was required in this area. Plant management conducted tailgate sessions with plant personnel, assigned plant personnel to review various events (19) in more detail and issued a memorandum to all badged personnel on the importance of attentiveness and self-checking before implementing a work step or activity.

b. Deviation Reports

In addition to the foregoing, the inspector reviewed the licensee's Deviation Reports (DVRs) and Discrepancy Records (DR) generated during the inspection period. This was done in an effort to monitor the conditions related to plant or personnel performance, potential trends, etc. DVRs were also reviewed to ensure that they were generated appropriately and dispositioned in a manner consistent with the applicable procedures and the QA manual.

The following DVRs and DRs were reviewed:

DR90-0148	6/22/90	"Wrong diaphragms installed in PORVS".
DR90-0163		"Unauthorized removal of temporary alteration".

DR90-0169 "Work instructions exceeded".
DR90-018 7/13/90 "RCS leakage from Process Sample System".

The corrective actions and root cause analysis were reviewed by the inspectors and found adequate.

5. Maintenance/Surveillance (62703 & 61726)

a. Maintenance Activities (62" 3)

Station maintenance activities that affected the safety-related and associated systems and components were observed or reviewed to ascertain compliance with approved procedures, regulatory guides and industry codes or standards, and in conformance with Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from and restored to service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and fire prevention controls were implemented. Work requests were reviewed to determine the status of outstanding jobs and to assure that priority is assigned to safety-related equipment maintenance which may affect system performance.

Portions of the following maintenance activities were observed and/or reviewed:

B 73830, "Preventive Maintenance on 2AF017B".
B 75542, "5 year Inspection of 2A DG".
B 76121, "Preventive Maintenance on 2AF017B, 2B Aux Feed Pump Suction Isolation Valve Operator (EM)".
B 76618, "Remove and Reinstall Hanger in Support of 2AF017B-L05, Electrical Maintenance Inspection".
B 77248, "Perform Preventive Maintenance on 1B SI Pump".
B 78503, "Troubleshoot Thermocouple Causing Negative Delta-T Readings".
B 78832, "Troubleshoot ATWS".
B 79257, "RCFC Breaker".

The inspectors periodically monitored the licensee's work in progress and verified performance was in accordance with proper procedures and approved work packages, that 10 CFR 50.59 safety reviews were conducted, as appropriate, applicable drawing updates were made and/or planned, and that operator training was conducted in a reasonable period of time.

b. Surveillance Activities (61726)

The inspectors observed or reviewed surveillance tests required by Technical Specifications during the inspection period and verified that tests were performed in accordance with adequate procedures, test instrumentation was calibrated, limiting conditions for operation were met, removal and restoration of the affected components were accomplished, results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and any deficiencies identified during the tests were properly reviewed and resolved by appropriate management personnel.

The inspectors also witnessed portions of the following activities:

1 BOS 6.1.1.a-1, Revision 3, Primary Containment Integrity Verification of Outside Containment Isolation Devices.

1 BOS 7.4.1.b-1, Revision 0, Unit-2, Essential Service Water Pump Availability to Unit-1, Monthly Surveillance.

1 BVS 2.1.4-1, Revision 7, Monthly Target Axial Flux Difference Determination.

1 BVS 2.2.2-1, Revision 8, Heat Flux Hot Channel Factor Checkout Using Peaking Factors.

1 BVS 2.3.2-1, Revision 6, Monthly Nuclear Enthalpy Rise Hot Channel Factor and RCS Total Flow Rate Check.

1 BVS 3.1.1-4, Revision 6, Incore-Excore Axial Flux Single Point Comparison Monthly Surveillance.

1 BVS 3.3.2-1, Revision 6, Moveable Incore Detectors Operability Check.

2 BIS 3.2.1-022, Revision 6, Surveillance Functional Test of Auxiliary Feedwater Pump Suction Pressure Loop.

2 BOS 7.4.1.b-1, Revision 0, Unit-1 Essential Service Water Pump Availability to Unit-2, Monthly Surveillance.

2 BOS 8.1.1.2.a-2, Revision 3, 2B Diesel Generator Operability Monthly and Semi-Annual Surveillance.

2 BOS 8.2.1.3-1, Revision 1, 125V DSC BUS 211 Load Shed When Cross-Tied to DC BUS 111.

2 BOS 9.1.1-1, Revision 20, Reactor Coolant System Refueling Reactivity Limit Surveillance.

2 BOS 9.10-1, Revision 0, Refuel Cavity Level Verification Within 2 Hours Prior to Movement of Fuel Assembly or Control Rods Within Containment.

No violations or deviations were identified.

6. Engineering & Technical Support (37700)

a. Estimated Critical Conditions

As discussed in Section 3.b of this report, on August 19, 1990, the licensee aborted a reactor startup, when it was determined that the critical rod position on Control Bank "D" would exceed the Estimated Critical Condition (ECC) by more than the administrative limit of 500 pcm. Subsequent licensee investigation determined that the Westinghouse computer calculation used for the ECC had an error that resulted in an erroneous ECC. The error was a result of some of the data used in the calculation being lost in the communication link between Westinghouse's Monroeville facility and the main computer. The licensee, as part of a verification process, also performs a ECC using a hand calculation and a computer program called Beacon. The ECC performed with the Beacon program agreed with the ECC performed by Westinghouse. However, the ECC performed with the Beacon also had an error caused by not having input for the power history 2 hours before the reactor trip. The station's nuclear engineers were unaware at the time of performing the ECC with the Beacon program that the point history from the process computer for reactor power had been lost 2 hours before and two hours after the reactor trip. The errors in the Westinghouse and Beacon programs were of such a nature that the ECCs happened to agree. Even though the hand calculation didn't agree with the Westinghouse or Beacon ECCs (approximately 400 pcm difference), the licensee decided to use the Westinghouse ECC based on 1) the Westinghouse and Beacon program ECCs were in agreement and the Westinghouse calculations had been accurate in the past and 2) the hand calculation had been different from the Westinghouse ECC on previous successful reactor startups by as much as 350 pcm. The difference between the Westinghouse calculation and the hand calculation appears to occur when the Unit has been subjected to large load swings which was the case just prior to the reactor trip on August 19, 1990 when just two hours prior to the trip the Unit had ramped down from 100% reactor power to 78% power. Also the Unit had been subjected to large load swings for several days prior to the trip. To prevent future ECCS outside the 500 pcm administrative limit, the licensee plans to revise the Beacon software to identify to the nuclear engineers any data that was not available from point history. Since this revision will not be completed for several months, the licensee in the interim will require the nuclear engineers to verify availability of the point history data prior to utilization of the Beacon program on ECCs.

b. Auto Start of 2A AFW Pump

On August 18, 1990, an unexpected auto start of the 2A AFW pump occurred. The cause was determined to be the gradual degradation of a transistor in the ATWS mitigation system, due to inductive flyback. Prior to the event, there was indication present on the ATWS control panel that a degradation of the circuit was present. However, due to lack of attention to detail, the system engineer did not initiate action to investigate the possible degradation.

c. Ability of the AFW Isolation Valves to Close

During engineering reviews associated with Generic letter 89-10, a discrepancy was identified with the ability of the AFW isolation valves 9AF013a-h) to isolate AFW to a faulted steam generator. Failure of the AF013 valves to isolate the faulted steam generator could potentially affect the steam line break analysis in the UFSAR. Procedure revisions were made to the applicable emergency operating procedures (EOP) to ensure AF013 valves would be closed within 10 minutes as required in the accident analysis. The EOP revisions require the operators to either locally close the AF013, if possible, or close the associated AF005 valve (flow control valves for AFW), or trip the associated AFW pump to decrease the pressure drop across the AF013 associated with the faulted steam generator if the AF013 fails to completely close from the control room. The EOP revisions were interim corrective actions. The licensee has initiated work to change the gears and spring packs for the Unit 2 AF013 valves that will be completed during the Unit 2 refueling outage. The Unit 1 AF013 valves will be modified commensurate with plant operations. Until these modifications occur the revised EOPs will remain in effect.

No violations or deviations were identified.

7. Refueling and Spent Fuel Pool Activities (60710, 86700)

The inspectors observed or reviewed the Unit 2 refueling and associated Spent Fuel Pool activities to verify the licensee had implemented controls for the conduct of refueling operations and for maintaining control of plant conditions, in accordance with the requirements of Technical Specifications (TS) and 10 CFR 50, Appendix A.

The following procedures were reviewed by the inspector:

BFP FH-A2 (Rev.2), September 18, 1990, "General Limitations and Actions for Fuel Movements".

BFP FH-2 Rev.6), November 13, 1989, "New Fuel Inspection".

2 BGP 100-6T4, Revision 0, Core Alteration/Fuel Movement Checklist.

The inspectors interviewed key licensee and contractor personnel regarding responsibilities, understanding of administrative and surveillance requirements and responses, prerequisities for refueling, equipment checkout, fuel receipt and inspection, and overall management direction and involvement. Observations of the activity were completed in the control room, fuel building and the containment.

During the Unit 2 outage, all of the fuel was unloaded from the reactor, moved to the spent fuel pool, ultrasonically tested for indications of fuel leaks, stored in the spent fuel pool, and will be reloaded into the reactor as required for the next fuel cycle.

The refueling activity was initiated on schedule, and proceeded in accordance with the plan and requirements except for the following anomalies which the licensee identified:

a. On September 25, 1990, during the movement of fuel from the reactor vessel (RV) to the spent fuel pool (SFP), at 1:30 p.m., the licensee found that SFP location D-D03 was already occupied by another fuel assembly. Fuel moves were stopped to perform an investigation. The nuclear component transfer list (NCTL) showed that at step 1859 no assembly should reside at the SFP location D-D03. The tag boards were reviewed and the licensee determined that SFP rack location D-E03 should have had an assembly, and did not. The assembly located in SFP location D-D03 was checked by underwater video equipment and found to be assembly number 561J, which should have been at location D-E03. Assembly 561J was moved from SFP location D-D03 to D-E03 using a procedure variation (per BAP 370-3TI), and a Deviation Report was initiated.

b. On September 29, 1990, at 1:45 p.m., during Unit 2 fuel reconstitution, of fuel assembly number T-77K, the licensee discovered that fuel rod B-04 was incorrectly removed from the fuel assembly. The intended rodlet for removal was number D-02. The Westinghouse procedure had been revised to require an independent verification of the correct rod prior to rod removal. Westinghouse failed to communicate to the licensee personnel, in the spent fuel pool area, that the independent verification was required. A Deviation Report was initiated.

c. On September 29, 1990, at 9:34 p.m., in the SFP, following reconstitution of fuel assembly T77K, the basket containing the fuel assembly was being inverted to return the fuel assembly to the proper vertical position. During the process the T77K fuel assembly slipped out of the basket when the basket lid opened, At the time, the basket was about 40 degrees above the horizontal position. The fuel assembly came to rest on the top of an empty fuel rack, halfway out of the basket. Some damage had occurred at the number 2 grid strap. Initial investigation indicated that the assembly fuel pins did not suffer any loss of integrity, as no bubbling or increase in radiation levels was observed. At 10:22 p.m., the licensee made an ENS call to the NRC, notifying them of this event and work was suspended in the SFP pending the evaluation of recovery steps. The recovery plan was developed, approved by management, and was successfully completed on September 30, 1990. Initial indications were that the lid on the basket did not latch properly and the triple verification steps had failed to detect this condition. The licensee did not plan to reuse the fuel assembly at this time. Further fuel reconstitution has been postponed pending a thorough investigati-- of the event.

These three events, occurring at the end of the inspection p-riod, and not having been completely resolved are considered an Unresolved Item by the inspectors (455/90019-03)(DRP)).

8. Open Items

Open Items are matters which have been discussed with the licensee, which will be reviewed by the inspector and which involve some action on the part of the NRC or licensee or both. An open item disclosed during the inspection is discussed in Paragraph 3.c(4).

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. Unresolved items disc .. during the inspection are discussed in Paragraphs 4.a(1) anc 7.

10. Meetings and Other Activities

a. Management Meetings (30702)

On September 17, 1990, M. J. Farber, Chief, Division of Reactor Projects, Section 1A, toured the Byron plant and met with licensee management to discuss plant performance and plant material condition.

On September 24, 1990, T. H. Boyce, Licensee Project Manager, NRR, met with licensee management a d discussed the status of outstanding licensing actions.

b. Exit Interview (30703)

The inspectors met with the licensee representatives denoted in paragraph 1 during the inspection period and at the conclusion of the inspection on October 2, 1990. The inspectors summarized the scope and results of the inspection and discussed the likely content of this inspection report. The licensee acknowledged the information and did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.