July 25, 2000

EA-00-166

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BYRON INSPECTION REPORT 50-454/00-09(DRP); 50-455/00-09(DRP)

Dear Mr. Kingsley:

On June 30, 2000, the NRC completed an inspection at the Byron 1 and 2 reactor facilities. The enclosed report presents the results of that inspection. The results of this inspection were discussed on July 6, 2000, with Mr. B. Adams and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, one issue of very low safety significance (GREEN) was identified. The issue was determined to involve a violation of NRC requirements. However, the violation was not cited due to its very low safety significance and because it has been entered into your corrective action program. If you contest the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Byron Generating Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/RA/

Michael J. Jordan, Chief
Reactor Projects Branch 3

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

Enclosure: Inspection Report 50-454/00-09(DRP);
50-455/00-09(DRP)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-454; 50-455
License Nos:	NPF-37; NPF-66
Report No:	50-454/00-09(DRP); 50-455/00-09(DRP)
Licensee:	Commonwealth Edison Company
Facility:	Byron Generating Station, Units 1 and 2
Location:	4450 N. German Church Road Byron, IL 61010
Dates:	May 16 - June 30, 2000
Inspectors:	E. Cobey, Senior Resident Inspector B. Kemker, Resident Inspector J. Adams, Resident Inspector D. Schrum, Reactor Engineer P. Lougheed, Reactor Engineer C. Thompson, Illinois Department of Nuclear Safety
Approved by:	Michael J. Jordan, Chief Reactor Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

Byron Generating Station Units 1 and 2
NRC Inspection Report 50-454/00-09(DRP); 50-455/00-09(DRP)

The report covers a 7-week period of inspection activities by the resident staff and region based inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process.

Cornerstone: Mitigating Systems

- Green. The inspectors identified that the licensee failed to appropriately monitor the ventilation supply dampers for the motor driven auxiliary feedwater (AF) pumps within the scope of the licensee's maintenance rule program. A noncited violation was issued.

  The damper failures reviewed did not result in a loss of safety function for the motor driven AF pumps and with the diesel driven AF pumps still available, this issue was determined to be of very low safety significance. (Section 1R12)

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Report Details

Summary of Plant Status

The licensee operated Units 1 and 2 at or near full power for the duration of this inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather

a. Inspection Scope

The inspectors evaluated the licensee's preparation for adverse weather conditions during the spring and summer months (i.e., high winds and high temperatures), which could potentially lead to loss of offsite power or loss of mitigating systems. The inspectors interviewed maintenance, engineering and operations department personnel and walked down the electrical switchyard, ultimate heat sink and other areas of the station potentially affected by high winds and high temperatures. The inspectors also reviewed system engineering summer readiness reviews, applicable portions of the Updated Final Safety Analysis Report and the procedures listed below.

- Unit 0 Byron Abnormal Operating Procedure (0BOA) ENV-1, "Adverse Weather Conditions Unit 0," Revision 1A
- 1BOA ENV-1, "Adverse Weather Conditions Unit 1," Revision 3
- 2BOA ENV-1, "Adverse Weather Conditions Unit 2," Revision 3
- Unit 0 Byron Operating Surveillance Requirement Procedure XHT-A1, "High Temperature Equipment Protection," Revision 3
- Nuclear Station Procedure OP-AA-101-505, "Station Response to Interconnected Grid Status," Revision 0

b. Findings

There were no findings identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors verified the system alignment of the Unit 0 (U-0) and 2A component cooling water (CC) trains while the 2B CC train was out-of-service for maintenance. The Unit 0 CC pump and heat exchanger are common to both Units 1 and 2 and can be aligned to support unavailability of a single CC train on either unit. The U-0 CC pump and heat exchanger were aligned to Unit 2 to support the unavailability of the 2B CC pump. The inspectors reviewed the system drawings and the following procedures to determine the correct system alignment.

- Byron Operating Procedure (BOP) CC-10, "Alignment of the U-0 CC Pump and U-0 CC HX [Heat Exchanger] to a Unit," Revision 9

- BOP CC-14, "Post LOCA [Loss of Coolant Accident] Alignment of the CC System," Revision 4

The inspectors performed walkdowns of the accessible portions of the system and verified the system lineup and each of the system operating parameters (i.e., temperature, pressure, flow, etc.). In addition, the inspectors reviewed applicable portions of the Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TS).

b. Findings

The inspectors identified that the U-0 CC pump, when aligned to a unit substituting for a "B" train pump, was unable to perform its design safety function as described in the UFSAR. The licensee did not recognize this and did not enter the applicable TS limiting condition for operation when the U-0 CC pump was aligned to Unit 2 to support maintenance on the 2B CC pump.

The limiting mode of operation described in the licensee's design basis for the CC system is to provide cooling following a LOCA on one unit and simultaneous safe shutdown of the other unit. The UFSAR states that two complete trains (pump, heat exchanger, and surge tank) are required to be operable for the affected unit prior to any single failure. The malfunction analysis in the UFSAR is based upon the assumption that two pumps can be expected for each unit prior to any single failure.

One complete train of CC equipment (pump, heat exchanger and surge tank) is sufficient to carry the required heat load during post LOCA recovery; however, since the CC system must be capable of accommodating an active or a passive failure under these conditions, two complete trains of CC equipment must be available before the event. The UFSAR also credits the capability to use the U-0 CC pump and heat exchanger on either unit.

When operating the system in the post LOCA recovery mode, the UFSAR credits the capability to separate the CC trains on the affected unit from one another by valves such that a leak (passive failure) in one train does not affect operation of the other. With the U-0 CC pump aligned to a unit, substituting for a "B" train pump, it is unable to perform this design function because procedure BOP CC-14, which is used to separate the trains from one another, isolates the U-0 CC pump from a surge tank. A change to the procedure based upon the results of an operability evaluation performed at Braidwood Station, directs operators to declare the U-0 CC pump inoperable and enter the applicable TS limiting condition for operation. The pump is inoperable when isolated from a surge tank because the pump would have insufficient net positive suction head to run.

The licensee has maintained that performing BOP CC-14 to separate the CC trains from one another would be a "beyond design basis evolution" because it would only be performed in response to a passive failure in excess of the makeup capability of the system. According to the licensee, the design basis assumes a maximum 50 gallons per minute (gpm) leakrate and the makeup capability for the system is 150 gpm. However, the inspectors were unable to reconcile this assertion with what the UFSAR states and concluded that, on the contrary, performance of BOP CC-14 to separate the CC trains from one another is not beyond the current design basis. This issue was entered into the licensee's corrective action program as Condition Report B2000-01837.

The inspectors reviewed TS 3.7.7 and noted that the allowed outage time for one CC pump is 7 days. The inspectors determined that the conditions of the TS were satisfied because the duration of the U-0 and 2B CC pump inoperability (approximately 1 day) was less than the allowed outage time.

Due to the differences between the NRC and the licensee on system meeting design bases, inspectors will review additional information to be provided by to licensee and this matter is unresolved. (50-454/455-00009-01)

1R05 Fire Protection

a. Inspection Scope

The inspectors toured the plant areas listed below to observe conditions related to fire protection.

- Main Control Room (Zone 2.1-0)
- Division 11 Miscellaneous Equipment Room (Zone 5.6-1)
- Division 21 Miscellaneous Equipment Room (Zone 5.6-2)

These areas were selected for inspection because they were identified as risk significant in the Byron Station Individual Plant Examination of External Events. The inspectors assessed the licensee's control of transient combustibles and ignition sources, material condition, and operational status of fire barriers and fire protection equipment. During this inspection, the inspectors interviewed engineering department personnel and the station's fire marshal.

In addition, the inspectors evaluated the licensee's corrective actions for fire protection issues documented in the following problem identification forms.

- PIF B1999-00474 Fire Protection Jockey Pump Material Condition
- PIF B1999-02293 Broke Fire Protection Valve Action Request Canceled With No Explanation
- PIF B1999-02577 As Left Condition of the OB Fire Protection Pump Following Maintenance

b. Findings

There were no findings identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed applicable portions of the documents listed below to identify areas that could be affected by internal or external flooding events.

- Byron/Braidwood Stations Updated Final Safety Analysis Report (UFSAR), Sections 2.4, 3.4, 9.2, 9.3, and 10.4

- Calculation 3C8-1281-001, “Auxiliary Building Flood Calculation,” Revision 11

- Calculation 3C8-0787-001, “Confirmation of Safe Shutdown Capability After Auxiliary Building Flooding,” Revision 1

- Safety Evaluation BRW-SE-1997-676, “Evaluation of Braidwood’s Positive Displacement Pump Out-of-Service for An Extended Period of Time”

- Byron and Braidwood Stations Individual Plant Examinations (IPE), Volume 2, March 1997

Based on the review of the above documents, the inspectors selected the river screen house and the essential service water pump rooms to walkdown and evaluate. The river screen house was selected because it houses the safety-related essential service water make up pumps and because the structure is located on the Rock River, increasing its susceptibility to an external flooding event. The essential service water pump rooms were selected due to their location (lowest level in the auxiliary building) and their high risk ranking in Byron Station’s plant specific risk analysis.

During the walkdown, the inspectors looked at equipment below the postulated floodline, penetrations in floors and walls, room drains and sumps, flooding detection sumps, and watertight doors. The inspectors conducted a walkdown of the exterior of main plant site buildings to verify that an elevation gradient existed sloping away from the buildings as specified in the UFSAR.

The inspectors reviewed the following documents related to the maintenance and testing of structures, systems and components in the selected areas.

- Byron Electrical Maintenance Procedure (BHP) 4200-81, “Calibration of Magnetrol Flood Level Switch,” Revision 2

- Unit 0 Byron Operating Surveillance Requirement Procedure (0BOSR) WF-SA1, “Auxiliary Building Floor Drain Semi-Annual Surveillance,” Revision 1

- Unit 0 Byron Mechanical Maintenance Surveillance Requirement Procedure (0BMSR) DD-1, “Water-Tight Barrier Inspection,” Revision 2

- Work Request (WR) 980091552-01, “Flood Seal Opening Inspection”

- WR 990145586-01, “Flood Seal Opening Inspection”

- WR 950099191, “Motor Operated Valve Thermal Overload Surveillance for the 1A Essential Service Water Pump Suction Valve Breaker”

- WR 980027178, “Motor Operated Valve Thermal Overload Surveillance for the 2A Essential Service Water Pump Suction Valve Breaker”

The inspectors also reviewed inservice testing program (IST) valve performance data for essential service water valves included in the IST program in order to verify that essential service water valves were being tested on the appropriate intervals and were operating within the action range.

During the review of the Auxiliary Building Flood and Confirmation of Safe Shutdown Capability After Auxiliary Building Flooding calculations, the inspectors noted that operator actions were credited for the isolation of the source of flooding in 30 minutes. The inspectors reviewed the following procedures to verify that procedures were available for coping with the event and could be used to achieve isolation of the flooding source.

- Unit 0 Byron Abnormal Operating Procedure (0BOA) ENV-2, “Rock River Abnormal Water Level Unit 0,” Revision 4

- 0BOA ENV-4, “Earthquake Unit 0," Revision 53A

- Unit 1 Byron Abnormal Operating Procedure (1BOA) PRI -1, “Excessive Primary Plant Leakage Unit 1,” Revision 54B

- 1BOA PRI-6, “Component Cooling Water Malfunction Unit 1,” Revision 56A

- 1BOA PRI-7, “Essential Service Water Malfunction Unit 1,” Revision 6A

- 1BOA Refuel-2, “Refueling Cavity or Spent Fuel Pool Level Loss,” Revision 55A

- 1BOA SD-2, “Shutdown LOCA [Loss of Coolant Accident] Unit 1,” Revision 51B

- Unit 1 Byron Emergency Procedure (1BEP)-1, “Loss of Reactor or Secondary Coolant,” Revision 1

- Unit 0 Byron Annunciator Response Procedure (0BAR) 0PL01J-9-A1, “Essential Service Water Pump 1A Leak Detection Sump Level High,” Revision 2

- 0BAR 0PL01J-9-A2, “Essential Service Water Pump 1B Leak Detection Sump Level High,” Revision 2

- 0BAR 0PL01J-9-B1, “Essential Service Water Pump 2A Leak Detection Sump Level High,” Revision 1

- 0BAR 0PL01J-9-B2, “Essential Service Water Pump 2B Leak Detection Sump Level High,” Revision 1

The inspectors reviewed the training documents listed below and verified that operators had been provided training for the internal flooding event, including the credited operator action to isolate the source of flooding within 30 minutes.

- Lesson Plan for SOER [Significant Operating Event Report] 85-05, “Case Study - Flooding,” operator training for Cycle 2, 1991

- Byron Station Operator Briefing Package, “IPE Vulnerability for Essential Service Water Pipe Breaks,” operator required reading April 17, 1997

- Initial License Training/Licensed Operator Requalification Training Module I-AM-16, “Internal Flooding of Power Plant Buildings - Case Study”, operator training for Cycle 00-2, February, 2000

The inspectors reviewed the Byron Modified IPE and noted that the core damage frequency was dominated by contributions from loss of essential service water sequences, one of which was the internal flooding event. The inspectors verified that the licensee understood how these flooding scenarios contributed to the overall core damage frequency. The inspectors discussed equipment modification plans and completion dates with engineering personnel. The inspectors reviewed procedure enhancements made to abnormal operating procedures and operator required reading, which were provided to enhance operator actions in response to an internal flooding event.

b. Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors evaluated the licensee’s implementation of the maintenance rule, 10 CFR Part 50.65, as it pertained to identified performance problems with the 1A and 2B circulating water (CW) pumps, main steam line radiation monitor failures, and motor driven auxiliary feedwater (AF) pump ventilation damper failures that had been documented in the following problem identification forms.

- PIF B1999-00827 Potentially Unanalyzed Condition With No VA [Auxiliary Building Ventilation] Supply Fans Available
- PIF B1999-00841 Failed VA Dampers to Unit 2 Motor Driven AF Pump Found Failed Closed
- PIF B1997-01749 1B CW Pump Tripped

- PIF B1999-02334 Trip of 2A CW Pump During Start of Pump
- PIF B1999-03305 Failure of Motor Driven AF Pump Ventilation Damper
- PIF B1999-04373 2A AF Pump VA Damper Failed to Operate
- PIF B1999-04453 1A CW Pump Seal Cooling Line Failure
- PIF B1999-04496 CW Pump 2B Failure to Develop Discharge Pressure
- PIF B1999-04597 Failure of Main Steam Line Area Rad Monitor
- PIF B1998-04680 High Alarm on 1AR023B/C/D

During this inspection, the inspectors evaluated the licensee's monitoring and trending of performance data, verified that performance criteria were established commensurate with safety, and verified that the equipment failures were appropriately evaluated in accordance with the maintenance rule. The inspectors also interviewed the station's maintenance rule coordinator and reviewed Nuclear Station Procedure ER-3010, "Maintenance Rule," Revision 0 and Calculation VA-102, "Auxiliary Building Energy Load Calcs for Elevation 330', 346', 364', 383', 401', and 426' in Abnormal Condition".

b. Findings

The inspectors identified that the VA supply dampers for the motor driven AF pumps were not appropriately monitored within the scope of the licensee's maintenance rule program. The important design function of the unit specific ductwork and supply dampers is to provide cooling flow to the motor driven ("A" train) AF pump area following a loss of offsite power/safety injection (LOOP/SI) event. Each unit specific ductwork has two normally closed dampers which are interlocked with the main VA supply fans to open when both supply fans are off.

The inspectors determined that inoperable dampers can render the motor driven AF pumps inoperable. The licensee performed Calculation VA-102 in 1987 to demonstrate that the plant could meet temperature requirements described in the UFSAR with outside air temperature at 95 degrees Fahrenheit with all four dampers operable.

The inspectors noted that several times in 1999 some of these dampers failed to function. Operability evaluations were performed (99-005 and 99-025) which determined that the VA system function was still operable at lower outside air temperatures with one of two unit specific VA supply flowpaths available.

Previously, the VA system ductwork and damper flows to the motor driven AF pump area were considered to be low risk significant support functions of the pumps and were monitored at the plant level for overall affect on VA system operability. As a result, the licensee did not document the inoperable dampers as maintenance rule functional failures associated with the operability of the motor driven AF pumps. Only an actual failure of an AF pump would be identified as a functional failure for the AF system.

Damper failures were identified and corrected in 1999. In addition, the dampers were put on 5 and 8 year preventive maintenance cycles. These preventive maintenance activities included oil level check, leak check, oil change, recycle timing, and stroke adjustment. However, the inspectors identified that the licensee did not perform periodic surveillance testing to prove that the dampers were operable. Previous damper failures were identified by observant operators during fan testing. The lack of damper testing included not testing the interlock circuitry between the fans and dampers. As a result, the licensee would not know if one or all of the dampers were inoperable for extended periods of time. The licensee did not perform periodic surveillance testing because it believed the dampers had low risk significance. One reason the licensee believed the dampers were low risk significant was that this function was not specifically modeled in the licensee's Plant Specific Analysis (PSA) as being required to support motor driven AF pump operability. As a result, the PSA model did not take into account the importance of these dampers with respect to motor driven AF pump operability.

Subsequently, the licensee's maintenance rule expert panel has reviewed this issue and determined the dampers to be high risk significance. Monitoring was established at the train level as required by the maintenance rule program (in this case by monitoring the unit specific dampers and flowpaths individually). This issue was entered into the licensee's corrective action program as problem identification form B2000-01781.

10 CFR 50.65(b) established the scope of the monitoring program for selection of safety-related and nonsafety-related structures, systems, and components (SSCs) to be included within the maintenance rule program. 10 CFR 50.65(b) states, in part, that the scope of the monitoring program shall include nonsafety-related SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function. Contrary to this requirement, the VA supply dampers for the motor driven AF pumps performed a function that required their inclusion in the scope of the monitoring program and were not included. This issue is a violation of 10 CFR 50.65(b).

The damper failures reviewed did not result in a loss of safety function for the motor driven AF pumps and with the diesel driven AF pump still available, this issue was determined to be of very low safety significance (green). Therefore, in accordance with Section VI.A.1 of the NRC Enforcement Policy, this violation is being treated as a noncited violation (NCV 50-454/455-00009-02).

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk for planned maintenance activities on the 1B essential service water pump, the 1B diesel driven auxiliary feedwater pump battery, and the 2B component cooling water pump. The inspectors selected these maintenance activities because they involved systems which were risk significant in the licensee's risk analysis. The maintenance activity associated with the 1B essential service water pump was considered emergent work to repair excessive leakage from a seal supply line.

During this inspection, the inspectors assessed the operability of redundant train equipment and evaluated the licensee's implementation of planned contingency actions to minimize plant risk, where appropriate. The inspectors also interviewed operations and work control department personnel and reviewed Nuclear Station Procedure WC-AA-103, "On-Line Maintenance," Revision 0.

In addition, the inspectors evaluated the licensee's corrective actions for maintenance issues documented in the following problem identification forms.

- PIF B1999-04591 Premature Failure of Damper Actuator
- PIF B1999-04106 ACB [Air Circuit Breaker] 7-10 Failure to Close
- PIF B1999-04713 Failed Stroke Time of ASME [American Society of Mechanical Engineers] Valve

b. <u>Findings</u>

There were no findings identified.

1R14 <u>Personnel Performance During Non-routine Plant Evolutions</u>

a. <u>Inspection Scope</u>

On May 18, 2000, the Unit 2 345-kilovolt station auxiliary transformer (SAT) disconnect was damaged during high winds which resulted in arcing. The inspectors evaluated the licensee's response to the failure of the disconnect, including contingency actions for a potential loss of SAT power, compliance with the requirements of the Technical Specifications, and operator actions to transfer power and de-energize the SAT. This evolution was selected for observation and evaluation because of the potential for a loss of off-site power to the unit.

b. <u>Findings</u>

There were no findings identified.

1R15 <u>Operability Evaluations</u>

a. <u>Inspection Scope</u>

The inspectors evaluated the licensee's basis that the issues identified in the following operability evaluations did not render the involved equipment inoperable or result in an unrecognized increase in plant risk.

- 1999-005 Failure of Dampers 0VA472Y and 0VA473Y
- 1999-025 Failure of Damper 0VA471Y
- 1999-027 Thermal Relief Valve 0CC9432 Installed Backwards

The inspectors interviewed engineering department personnel and reviewed applicable portions of the Updated Final Safety Analysis Report, Technical Specifications, and the following documents.

- Nuclear Station Procedure CC-3001, "Operability Determination Process," Revision 0
- Byron/Braidwood Calculation VA-102, "Auxiliary Building Energy Load Calculations for Elevations 330 Feet, 346 Feet, 364 Feet, 383 Feet, 401 Feet and 426 Feet in Abnormal Condition," Revision 3

- Byron Operating Procedure CC-14, “Post LOCA [Loss of Coolant Accident] Alignment of the CC [Component Cooling Water] System,” Revision 4
- BOP VA-1, “Auxiliary Building HVAC [Heating, Ventilation and Air Conditioning] System Startup and Operation,” Revision 7

b. Findings

There were no findings identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed operator work-around (OWA) 211, “Primary Water Flow Deviation Alarm During Routine Dilutions.” The inspectors selected this OWA because the recurring alarm and subsequent isolation of makeup water to the chemical and volume control system creates a challenge to operators and has contributed to two previous reactivity control events. The inspectors evaluated the OWA to identify any potential affect on the functionality of mitigating systems or on the operators’ response to initiating events. The inspectors interviewed operating and engineering department personnel and reviewed Nuclear Station Procedure OP-AA-101-303, “Operator Work-Around Program,” Revision 0.

In addition, the inspectors evaluated the licensee’s corrective actions for the OWA issue documented in the following problem identification forms.

- PIF B1998-04451 Operator Work-Around “OWA 172" Deleted Without Notice Nor Repair of Deficiency
- PIF B1998-04518 PIF No. B1998-04451 Issued Closed Without Correction

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors evaluated the licensee’s post maintenance testing activities for the maintenance conducted on the 1B essential service water (SX) pump, the 1B diesel driven auxiliary feedwater (AF) pump battery, and the 2A steam generator (SG) power operated relief valve (PORV), which included the following work requests.

- WR 970049681-01 2A SG PORV - Replace Damaged Seal Tight, Cable, Fittings on Limit Switch
- WR 980036798-01 2A SG PORV - Add Small Window to SG PORV Control Box and a Top Vent Hole
- WR 980077790-01 2A SG PORV - Replace Actuator with Overhauled Actuator

- WR 980108423-01 2A SG PORV - Replace Nitrogen Pressure Switch
- WR 980135259-01 1B Diesel Driven AF Pump #1 Battery - Replace Entire Battery Bank With 2-Cell Blocks
- WR 990176621-01 Excessive Leakage From 1B SX Pump Seal Package Supply Line
- WR 990176621-02 Remove/Reinstall Flood Seal to Support Electrical Maintenance
- WR 990176621-03 Remove/Reinstall Blank Flange to Facilitate SX Line Draining

The inspectors selected these post maintenance testing activities because they involved systems which were risk significant in the licensee's risk analysis. The inspectors interviewed operations, maintenance and engineering department personnel. The inspectors also reviewed the completed post-maintenance testing documentation and data for compliance with the associated procedures, applicable portions of the Updated Final Safety Analysis Report and the Technical Specifications to verify that the systems and components were capable of performing their intended safety functions.

In addition, the inspectors evaluated the licensee's corrective actions for a post maintenance testing issue documented in the following problem identification form.

- PIF B1999-04600 Failed Post Maintenance Test on 1CS010A

b. Findings

There were no findings identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors evaluated the surveillance testing activities listed below to verify that the testing demonstrated that the equipment was capable of performing its intended function.

- 1BOSR 3.2.7-616B Unit One ESFAS [Engineered Safety Feature Actuation System] Instrumentation Slave Relay Surveillance (Train B Steam Line Isolation - K616)
- 1BOSR 3.2.7-638B Unit One ESFAS Instrumentation Slave Relay Surveillance (Train B Feedwater Isolation, Hi-Hi S/G [Steam Generator] Level - K638)
- 2BVSR 6.6.4-1 Unit 2 ASME [American Society of Mechanical Engineers] Surveillance Requirements for 2A Containment Spray Pump

The inspectors selected 1BOSR 3.2.7-616B and 1BOSR 3.2.7-638B because the licensee considered these surveillance tests high risk activities. The inspectors selected 2BVSR 6.6.4-1 because the containment spray system, as described in the Updated Final Safety Analysis Report (UFSAR), is credited in the event of a design basis accident to ensure that the offsite and site boundary dose limits of 10 CFR Part 100 are not exceeded. The inspectors interviewed operations and engineering department personnel; reviewed the completed test documentation and applicable portions of the UFSAR and the Technical Specifications; and observed the performance of these surveillance testing activities.

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 (Closed) Unresolved Item (URI) 50-454/455-99019-02: “Review of the Licensee’s Safety Evaluation to Accept the As-Built Condition of the Emergency Core Cooling System Recirculation Sump Screens." As documented in NRC Inspection Report 50-454/455-99019(DRP), the licensee performed an evaluation in accordance with 10 CFR 50.59 and determined that NRC approval was not required. Because of the complexities of the evaluation, the unresolved item was written to allow for specialist review of the evaluation. The specialists completed their review in May 2000 and determined that sump screen hole sizing was a generic issue. Based upon this finding, the specialists recommended that the concern be considered as part of Generic Safety Issue 191, "Effects of Pressurized Water Reactor Sump Screen Blockage," or other ongoing research into generic sump issues. Upon completion of the generic research, licensees will be informed if any specific actions are required through issuance of an appropriate generic communication. Therefore, no action will be taken on this specific issue. This item is closed.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. B. Adams and other members of licensee management at the conclusion of the inspection on July 6, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Adams, Regulatory Assurance Manager
S. Gackstetter, Shift Operations Superintendent
D. Hoots, Operations Manager
J. Kramer, Work Control Manager
S. Kuczynski, Maintenance Manager
W. Levis, Site Vice President
R. Lopriore, Station Manager
D. McDermott, On-Line Work Control Superintendent
T. Roberts, Design Engineering Manager
D. Wozniak , Engineering Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-454/455-00009-01	URI	NRC to review additional information on split trains of component cooling.
50-454/455-00009-02	NCV	Failure to properly monitor supply dampers for the motor driven auxiliary feedwater pumps in the scope maintenance rule program

Closed

50-454/455-00009-02	NCV	Failure to properly monitor supply dampers for the motor driven auxiliary feedwater pumps in the scope maintenance rule program
50-454/455-99019-02	URI	Review of the licensee's safety evaluation to accept the as-built condition of the emergency core cooling system recirculation sump screens

Discussed

None

LIST OF BASELINE INSPECTIONS PERFORMED

The following inspectable-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

Inspection Procedure		
Number	Title	Report Section
71111-01	Adverse Weather Protection	1R01
71111-04	Equipment Alignment	1R04
71111-05	Fire Protection	1R05
71111-06	Flood Protection Measures	1R06
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Risk Assessments and Emergent Work Control	1R13
71111-14	Personnel Performance During Non-routine Plant Evolutions and Events	1R14
71111-15	Operability Evaluations	1R15
71111-16	Operator Work-Arounds	1R16
71111-19	Post Maintenance Testing	1R19
71111-22	Surveillance Testing	1R22

LIST OF ACRONYMS USED

ACB	Air Circuit Breaker
AF	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
BAR	Byron Annunciator Response Procedure
BEP	Byron Emergency Procedure
BHP	Byron Electrical Maintenance Procedure
BMSR	Byron Mechanical Maintenance Surveillance Requirements Procedure
BOA	Byron Abnormal Operating Procedure
BOP	Byron Operating Procedure
BOSR	Byron Operating Surveillance Requirement Procedure
BVSR	Byron Technical Surveillance Requirement Procedure
CC	Component Cooling Water
CW	Circulating Water
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
ESFAS	Engineered Safety Feature Actuation System
gpm	gallons per minute
HVAC	Heating, Ventilation and Air Conditioning
IPE	Individual Plant Examination
IST	Inservice Testing
LOCA	Loss of Coolant Accident
LOOP/SI	Loss of Offsite Power/Safety Injection
NCV	Noncited Violation
NRC	Nuclear Regulatory Commission
OWA	Operator Work-Around
PIF	Problem Identification Form
PORV	Power Operated Relief Valve
PSA	Plant Specific Analysis
SAT	Station Auxiliary Transformer
SG or S/G	Steam Generator
SOER	Significant Operating Event Report
SX	Essential Service Water
TS	Technical Specification
U-0	Unit 0
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VA	Auxiliary Building Ventilation
WR	Work Request