UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

November 5, 2009

Mr. Charles G. Pardee
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO), Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNITS 1 AND 2 INTEGRATED INSPECTION REPORT 05000454/2009004; 05000455/2009004

Dear Mr. Pardee:

On September 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Byron Station, Units 1 and 2. The enclosed inspection report documents the inspection findings which were discussed on October 9, 2009, with D. Enright and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, two NRC-identified findings of low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Byron Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Byron Station. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Richard A. Skokowski, Chief
Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000454/2009-004 and 05000455/2009-004
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-454; 50-455
License Nos:	NPF-37; NPF-66
Report Nos:	05000454/2009004 and 05000455/2009004
Licensee:	Exelon Generation Company, LLC
Facility:	Byron Station, Units 1 and 2
Location:	Byron, IL
Dates:	July 01, 2009, through September 30, 2009
Inspectors:	B. Bartlett, Senior Resident Inspector J. Robbins, Resident Inspector S. Bakhsh, Lead ISFSI Inspector J. Bozga, Reactor Inspector J. Cassidy, Senior Health Physicist T. Go, Health Physicist D. Jones, Reactor Inspector M. Learn, Reactor Engineer E. Love, Safety Inspector V. Meghani, Reactor Inspector M. Mitchell, Health Physicist C. Morell, Safety Inspector J. Neurauter, Engineering Branch 1 R. Ng, Project Engineer J. Pearson, Senior Safety Inspector C. Tilton, Senior Reactor Engineer C. Thompson, Resident Inspector, Illinois Department of Emergency Management
Approved by:	R. Skokowski, Chief Branch 3 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000454/2009004, 05000454/2009004; July 01, 2009 – September 30, 2009; Byron Station, Units 1 & 2; Identification and Resolution of Problems, and Preoperational Testing of an Independent Spent Fuel Storage Facility Installation (ISFSI) at Operating Plants

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green finding was identified by the inspectors. The findings were considered Non-Cited Violations of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety-significance and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for failure to perform an adequate evaluation of seismic restraint on the Fuel Handling Building (FHB) crane trolley. Specifically, for evaluation of the seismic restraint in their single failure proof trolley analysis, the licensee failed to use adequate seismic acceleration values and failed to evaluate the connections for resulting reaction forces. Subsequent review found that the restraint was inadequate. The licensee documented the condition in Issue Report (IR) 934467 and initiated actions for calculation revision and installation of a field modification.

  The inspectors determined that the failure to perform an adequate analysis for the seismic restraint and its connections for seismic loads was contrary to American Society of Mechanical Engineers (ASME) NOG-1-2004, requirements and was a performance deficiency. The FHB crane is designed to Seismic Category I requirements and the licensee used compliance with ASME NOG-1-2004, as the design basis for their upgrade to a single failure proof crane. The finding was more than minor because it was associated with the Initiating Events cornerstone attribute of Equipment Performance, Refueling/Fuel Handling equipment, and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors evaluated the finding using IMC 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and based on a "No" answer to all the questions in the Initiating Events column of Table 4a, determined the finding to be of very low safety-significance (Green). This finding has a cross-cutting aspect in the area of Human Performance, Work Practices (H.4(c)) because the licensee did not provide adequate oversight of work activities, including contractors, such that nuclear safety is supported. (Section 4AO5.1)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance and associated NCV of 10 CFR 50, Appendix A, Criterion 2, "Design Basis for Protection Against Natural Phenomena," and Criterion 4, "Environmental and Natural Effects Design Bases," was identified by the inspectors for the failure to seismically support and protect the Emergency Diesel Generator (EDG) fuel oil storage tank vent lines from tornado generated missiles. Specifically, the licensee installed the vent lines as non-safety related and as such they were not seismically supported nor protected from tornado generated missiles. In response to the issue, the licensee performed an operability determination and concluded that the EDGs remained operable.

This performance deficiency was more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring availability of the EDG to respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance (Green) because the inspectors determined that the finding was a design deficiency confirmed not to result in loss of operability or functionality and the finding screened as Green using the Significance Determination Process Phase 1 screening worksheet. The inspectors did not identify a cross-cutting aspect associated with this finding because the performance deficiency occurred over 30 years ago and was not current. (Section 4OA3)

B. Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee has been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full power through most of the inspection period. On September 13, 2009, at 11:00 pm the licensee brought Unit 1 off-line to begin a scheduled refueling outage. At the end of the report period, the unit was still off-line.

Unit 2 operated at full power through most of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 1 Train A Containment Spray while Unit 1 Train B Containment Spray was Out-of-Service; and
- Unit 1 Train A Safety Injection while Unit 1 Train B Safety Injection was Out-of-Service.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

These activities constituted two partial system walkdown samples as defined in Inspection Procedure (IP) 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On September 22, 2009 the inspectors performed a complete system alignment inspection of the Unit 1 Residual Heat Removal System to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Unit 1 & 2 Trains A & B Essential Service Water Pump (SX) Rooms;
- Unit 2 Train A the Emergency Diesel Generator (EDG) and Day Tank Rooms; and
- Turbine Building 369' Elevation General Area.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted three quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the Unit Common Component Cooling Heat Exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this document.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P)

From September 8, 2009, through September 17, 2009, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, steam generator tubes, emergency feedwater systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, R08.3, IR08.4 and 1R08.5 below constituted one inservice inspection sample as defined in Inspection Procedure 71111.08-05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed and reviewed records of the following nondestructive examinations mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Ultrasonic Examination of Residual Heat Removal (RH) Pipe to Elbow Weld 1RH01CA-16 (Report No. B1R16-UT-001);
- Ultrasonic Examination of RH Reducer to Nozzle Weld 1RH02AA-8 (Report No. B1R16-UT-002);
- Ultrasonic Examination of RH Pipe to Elbow Weld 1RH03AB-8 (Report No. B1R16-UT-003);
- Ultrasonic Examination of RH Elbow to Pipe Weld 1RH02AB-8 (Report No. B1R16-UT-004); and
- Ultrasonic Examination of Safety Injection (SI) Elbow to Pipe Weld 1SI05AB-8 (Report No. B1R16-UT-005).

The inspectors reviewed records of the following nondestructive examinations conducted as part of the licensee's industry initiative inspection program for primary water stress corrosion cracking to determine if the examination was conducted in accordance with the licensee's augmented inspection program, industry guidance documents and associated licensee examination procedures and if any indications and defects were detected, to determine if these were dispositioned in accordance with approved procedures and NRC requirements.

- Liquid Penetrant Examination of RH Heat Exchanger to Support Skirt Weld 1RH-02-AB-RHES-01 (Report No. B1R16-PT-001).

The inspectors reviewed the following examinations completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine if acceptance was in accordance with the ASME Code Section XI or an NRC approved alternative.

- Magnetic Particle Examination of replacement steam generator manway study No. 16.

The inspectors reviewed the following pressure boundary weld repair completed on a risk-significant system since the beginning of the last refuelling outage to verify that the welding and any associated non-destructive examinations were performed in accordance with the Construction Code and ASME Code, Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure(s) were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Replacement of an ASME Section III, Class 1, 6' Section of 1FW03DA-16" Feedwater line, Work Order 00955081-01 / 02.

a. Findings

No findings of significance were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 1 vessel head, no examination was required for this refueling outage. Therefore, no NRC review was completed for this inspection procedure attribute.

The licensee did not perform any welded repairs to vessel head penetrations since the beginning of the preceding outage. Therefore, no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors observed and reviewed records of the licensee's initial Boric Acid Corrosion Control visual examinations and verified whether these visual examinations emphasized locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following licensee evaluations of reactor coolant system components with boric acid deposits to determine if degraded components were documented in the corrective action system. The inspectors also evaluated corrective actions for any degraded reactor coolant system components to determine if they met the component Construction Code, ASME Section XI Code, and/or NRC approved alternative.

- IR 894912, Boric Acid On 1PR06J Piping; and
- IR 949267, Minor Boric Acid Accumulation on 1SI097.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- IR 956010, Dried Boric Acid On 1SI8811A Packing; and
- IR 956014, Dried Boric Acid On 1RH8735 Packing.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

For the Unit 1 steam generators, no examination was required pursuant to the TSs during this refueling outage. Therefore, no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related-problems entered into the licensee's corrective action program and conducted interviews with licensee staff to determine if;

- the licensee had established an appropriate threshold for identifying ISI/SG related-problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On August 4, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Unit 1 Train A Reactor Containment Fan Cooler Fan Motor Vibration; and
- Unit 1 Reactor Coolant Loop D Loop Stop Isolation Valve Guide Failure.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Unit 1 Train B EDG while Unit 1 Train A EDG Inoperable due to Air Filter Failure;
- Unit 2 Train B Auxiliary Feed Water Pump while Unit 2 Train A was Inoperable due to Emergent Work on Valve 2AF017A; and
- Unit 1 Train A and Unit 2 Train A SX Pumps during Replacement of Unit 1 Train B SX Pump with Flooding Barriers Impaired.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

These maintenance risk assessments and emergent work control activities constituted three samples as defined in IP 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Unit 1 Pressurizer Spray Piping Support 1RY06121S
- Unit 2 Train A Solid State Protection System (SSPS) Surveillance with Unit 2 Train B SSPS Jumper Installed;
- Ultimate Heat Sink Capability with Failure of SX Fans;
- Unit 2 Pipe Support 426' Level of the Turbine Building due to Non-Standard Attachment Method; and
- Unit 2 Train A Auxiliary Feedwater with 2AF017A Open and De-energized.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted five samples as defined in IP 71111.15-05

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification:

- Unit 2 Auxiliary Feedwater Valve 2AF017A Placed in the Open Position and De-energized.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system. The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this document.

This inspection constituted one temporary modification sample as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Unit 0 0SX007 Valve Replacement;
- Unit 0 0SX161B, Valve Repairs Requiring Use of Freeze Seal;
- Unit 1 Train A EDG Governor Potentiometer Replacement;
- Unit 1 Valve 1FW039C Check-Valve Replacement;
- Unit 1 Train A Residual Heat Removal Comprehensive Test;
- Unit 1 Train B Residual Heat Removal Comprehensive Test; and
- Unit 2 Train A Charging Pump Deflector Ring Setscrew Replacement.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted seven post-maintenance testing sample as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 1 refueling outage (RFO), licensee designated B1R16, which started September 13, 2009, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. At the end of the inspection period B1R16 was still ongoing. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Refueling activities.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO activities.

This inspection constituted one RFO sample as defined in IP 71111.20-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Unit 0 Train A and Train B Fire Pump Annual Alarm Test;
- Unit 1 Containment Isolation Valve 1SA033 Local Leak Rate Testing;
- Unit 1, Division 11, Battery 111 Load Testing; and
- Unit 2 Train A, SSPS Surveillance.

The inspectors observed in plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;

- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on July 29, 2009, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator control room and the technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's Occupational Exposure Control Cornerstone performance indicator (PI) to determine whether the conditions resulting in any PI occurrences had been evaluated and whether identified problems had been entered into the licensee's CAP for resolution.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas, and airborne radioactivity areas in the plant to determine if radiological controls including surveys, postings, and barricades were acceptable:

- Auxiliary Building;
- Containment Building; and
- Spent Fuel Pool.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed the radiation work permits (RWPs) and work packages used to access these areas and other high radiation work areas. The inspectors assessed the work control instructions and control barriers specified by the licensee. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. The inspectors interviewed workers to verify that they were aware of the actions required if their electronic dosimeters noticeably malfunctioned or alarmed.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors walked down and surveyed (using an NRC survey meter) these areas to verify that the prescribed RWP, procedure, and engineering controls were in place; that licensee surveys and postings were complete and accurate; and that air samplers were properly located.

This inspection constituted one sample as defined in IP 71121.01-5.

The inspectors reviewed RWPs Byron Unit-1 2009 station outage for airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g. high-efficiency particulate air ventilation system operation) and to determine if there was a potential for individual worker internal exposures in excess of 50 millirem committed effective dose equivalent. There were no airborne radioactivity work areas during the inspection period.

Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and had provided appropriate worker protection.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates in excess of 25 R/hr at 30 centimeters or in excess of 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures exceeding 100 millirem total effective dose equivalent (or 5 rem shallow dose equivalent or 1.5 rem lens dose equivalent) were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following job that was being performed in radiation areas, for observation of work activities that presented the greatest radiological risk to workers during the Unit 1 refueling outage. The inspectors reviewed radiological job requirements for these activities, including RWP requirements and work procedure requirements and attended radiation protection (RP) pre-job briefings.

This inspection constituted one sample as defined in IP 71121.01-5.

Job performance was observed with respect to the radiological control requirements to assess whether radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors evaluated the adequacy of radiological controls, including required radiation, contamination, and airborne surveys for system breaches; RP job coverage, including any applicable audio and visual surveillance for remote job coverage; and contamination controls.

This inspection constituted one sample as defined in Inspection Procedure 71121.01-5.

The inspectors reviewed radiological work in high radiation work areas having significant dose rate gradients to evaluate whether the licensee adequately monitored exposure to personnel and to assess the adequacy of licensee controls. These work areas involved areas where the dose rate gradients were severe, thereby increasing the necessity of providing multiple dosimeters or enhanced job controls.

This inspection constituted one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate, High Radiation Area and Very High Radiation Area Controls

a. Inspection Scope

The inspectors conducted plant walkdowns to assess the posting and locking of entrances to high dose rate high radiation areas and very high radiation areas.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation safety work requirements. The inspectors evaluated whether workers were aware of any significant radiological conditions in their workplace, of the RWP controls and limits in place, and of the level of radiological hazards present. The inspectors also observed worker performance to determine if workers accounted for these radiological hazards.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated RP technician performance with respect to radiation safety work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

This inspection constituted one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the outage work scheduled during the inspection period and associated work activity exposure estimates for the following five work activities, which were likely to result in the highest personnel collective exposures:

- Walkdown and Permanent Scaffolding Installation;
- Unit 1 Reactor Coolant Loop D, Cold Leg, Loop Stop Isolation Valve (LSIV) Bonnet Lift Inside the Missile Barrier;
- Unit 1 LSIV Interference, Scaffold, Insulation and Support Activities;
- B1R16 Emergent 1SI800C and 1SI8900D Repair and Replacement; and
- Shielding Related Activities.

This inspection constituted one required sample as defined in IP 71121.02-5.

a. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors assessed the integration of ALARA requirements into work procedures and radiological work planning documents to assess whether the licensee was implementing actions in radiological job planning in order to reduce dose.

This inspection constituted one optional sample as defined in IP 71121.02-5. The inspectors evaluated the licensee's process for constructing or placing shielding in high dose rate areas. The inspectors reviewed the shielding requests initiated by the RP group to evaluate the estimated dose rate reduction. The inspectors also evaluated the responses of the engineering staff to the shielding requests, as applicable.

This inspection constituted one optional sample as defined in IP 71121.02-5.

The inspectors evaluated if the licensee's planning for radiological significant work activities included consideration of the benefits of dose rate reduction activities, such as shielding (provided by water filled components/piping), job scheduling, and shielding and scaffolding installation and removal activities.

This inspection constituted one optional sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

.3 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors attended work briefings and observed ongoing work activities to determine if workers received appropriate on-the-job supervision to ensure the ALARA requirements are met. The inspectors assessed whether the first-line job supervisor ensured that the work activity was conducted in a dose efficient manner by minimizing work crew size and by ensuring that workers were properly trained and that proper tools and equipment were available when the job started.

This inspection constituted one optional sample as defined in IP 71121.02-5.

The inspectors reviewed exposures of individuals from selected work groups to evaluate any significant exposure variations among workers and to determine whether any significant exposure variations were the result of worker job skill differences or whether certain workers received higher doses because of poor ALARA work practices.

This inspection constituted one optional sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed corrective action reports related to the ALARA program and interviewed staff members to verify that follow-up activities had been conducted in an effective and timely manner commensurate with their importance to safety and risk using the following criteria:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;

- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation / consideration of risk significant operational experience feedback.

This inspection constituted one optional sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the UFSAR for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit program with regard to radioactive material processing and transportation programs to verify that it met the requirements of 10 CFR 20.1101(c).

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radwaste processing systems to verify that the systems agreed with the descriptions in the UFSAR and the Process Control Program and to assess the material condition and operability of the systems. The inspectors reviewed the status of radwaste processing equipment that was not operational and/or was abandoned in place. The inspectors reviewed the licensee's administrative and physical controls to ensure that the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed changes to the waste processing system to verify that the changes were reviewed and documented in accordance with 10 CFR 50.59 and to assess the impact of the changes on radiation dose to members of the public. The inspectors reviewed the current processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized. The inspectors also reviewed the licensee's methods for waste concentration averaging to determine if representative samples of the waste product were provided for the purposes of waste classification, as required by 10 CFR 61.55.

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams, including dry active waste, spent resins, and filters. The inspectors also reviewed the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The reviews were conducted to verify that the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure that the waste stream composition data accounted for changing operational parameters and thus remained valid between the annual sample analysis updates.

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipment Manifests

a. Inspection Scope

The inspectors reviewed the documentation of shipment packaging, radiation surveys, package labeling and marking, vehicle inspections and placarding, emergency instructions, determination of waste classification/isotopic identification, and licensee verification of shipment readiness for five non-excepted material and radwaste shipments made in 2007 through 2009. The shipment documentation reviewed consisted of one Low Specific Activity-II, one Type A, and three Type B shipments.

For each shipment, the inspectors determined if the requirements of 10 CFR Parts 20 and 61 and those of the Department of Transportation (DOT) in 49 CFR Parts 170-189 were met. Specifically, records were reviewed and staff involved in shipment activities was interviewed to determine if packages were labeled and marked properly, if package and transport vehicle surveys were performed with appropriate instrumentation, if radiation survey results satisfied DOT requirements, and if the quantity and type of radionuclides in each shipment were determined accurately. The inspectors also determined whether shipment manifests were completed in accordance with DOT and NRC requirements, if they included the required emergency response information, if the recipient was authorized to receive the shipment, and if shipments were tracked as required by 10 CFR Part 20, Appendix G.

This inspection constitutes one sample as defined by Inspection Procedure 71122.02-5.

Selected staff involved in shipment activities were interviewed by the inspectors to determine if they had adequate skills to accomplish shipment related tasks and to determine if the shippers were knowledgeable of the applicable regulations to satisfy package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172 Subpart H. Additionally, lesson plans for safety training and function specific training for RP technicians and for hazardous material (HAZMAT) level two employees were reviewed for compliance with the hazardous material training requirements of 49 CFR 172.704. Additionally, the HAZMAT training test and the test results for selected RP staff were reviewed by the inspectors for adequacy.

This inspection constitutes one sample as defined by Inspection Procedure 71122.02-5.

b. Findings

Introduction: The inspector identified an Unresolved Item (URI) associated with the licensee's characterization of the quantities and types of radionuclides in selected shipments.

Discussion: The inspectors reviewed several shipments that occurred at the end of 2007 and beginning of 2008. Based on the initial assessment by the inspectors, there appeared to be discrepancies with radionuclide activities reported on the shipping manifests and the associated 10 CFR Part 61 analysis for the most appropriate waste stream for the contents of the shipments. The discrepancy could not be immediately resolved by the licensee. Therefore, the inspectors could not evaluate whether the packages were correctly characterized for shipment and ultimate burial. Consequently, this issue remains under review by the NRC to determine if it represents a performance deficiency and is categorized as an URI (URI 05000454/2009004-01; 05000455/2009004-01).

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports, audits and self-assessments that addressed radioactive waste and radioactive materials shipping program deficiencies since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized and corrected. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors reviewed corrective action reports from the radioactive material and shipping programs since the previous inspection, interviewed staff and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of NCVs tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

This inspection constituted one sample as defined in IP 71122.02-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the Unit 1 and Unit 2 Reactor Coolant System (RCS) Specific Activity PI for the period from the April 2008 through June 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, issue reports, event reports and NRC Integrated Inspection Reports for the period of April 2008 through June 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a RCS sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS specific activity samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the period from April 2008 through June 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator

Guideline," Revision 5, was used. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with RP staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational radiological occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences performance indicator for the period of April 2008 through June 2009. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5 to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates between April 2008 and June 2009 to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RETS/ODCM radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.4 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures PI for both units for the period from the first quarter 2008, through the first quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in Revision 5 of the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" were used. The inspectors reviewed control room logs, action requests, event reports and NRC Inspection Reports from January 1, 2008, through March 31, 2009, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's corrective action program database to verify that any problems regarding the PI data had been entered into the licensee's corrective action program with the appropriate characterization and significance.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted two safety system functional failures sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.5 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage performance indicator for Unit 1 and Unit 2 for the period from the beginning of the third quarter of 2008 until the end of the second quarter of 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period of July 2008 through June of 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two reactor coolant system leakage samples as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Items Entered into the Corrective Action Program

a. Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the attached List of Documents Reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-Up Inspection: Unit 1 Loose Part Monitor

a. Scope

The inspectors selected the following for an in-depth review:

- Unit 1 Loose Part Monitor Indicates Source of Noise as Near LSIV 1RC8002D.

The inspectors discussed the evaluations and associated corrective actions with licensee personnel and verified the following attributes during their review of the root cause evaluation:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause and previous occurrences;
- classification and prioritization of the resolution of the problem, commensurate with safety significance;
- identification of the root and contributing causes of the problem; and
- identification of corrective actions, which were appropriately focused to correct the problem.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings of significance were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report (LER) 05000455/2009-001-00: Late Entry Into TS Condition Associated with Reactor Coolant System Leakage Characterization Resulting in a Condition Prohibited by TSs

Introduction: On June 24, 2009, during a routine containment entry at power licensee personnel identified a pinhole leak (one drop every five minutes) on a welded connection inside of Unit 1 containment (IR 934800). The welded connection was on line 2PS01BB and the line is 3/8 inch in diameter. This line is a pressurizer liquid sample line and is a non-safety related non-ASME code, Class D pipe. The licensee verified that valve 2PS9350B upstream (between the leak and the RCS pressurizer) was closed and that both containment isolation valves downstream were closed. Based on the upstream valve being closed and in the Shift Manager's opinion being isolated, and with the leakage being not significant, the leak was not considered by licensee personnel to be RCS pressure boundary leakage.

The NRC inspectors consulted regional management and headquarters personnel and on June 26, 2009 at 4:30 pm, the licensee was informed that in the NRC's opinion, the leak was RCS pressure boundary leakage and that TS 3.4.13.B should have been entered. The licensee acknowledged the NRC opinion and immediately entered TS 3.4.13.B.

The licensee had begun repair efforts earlier in the day on June 26, 2009. The repair was completed; post maintenance testing was performed and the TS was exited at 8:07 pm on June 26. The NRC inspectors were informed of the completion of the licensee's repair efforts.

In NRC Inspection Report 05000454/2009003 the inspectors performed a SDP of the licensee's failure to comply with TS using IMC 0609, Attachment 0609.04. The inspectors determined the finding fell under the Initiating Events Cornerstone as a primary system LOCA initiator, did not represent a transient initiator contributor, did not represent a fire initiator contributor, and was not an internal/external flooding initiator contributor. The inspectors determined that assuming the worst case degradation could the finding result in exceeding the TS limit for any RCS leakage was yes as the TS limit for RCS pressure boundary leakage was none and there was one drop every five minutes. The inspectors then performed a Phase 2 SDP using the risk informed inspection notebook. The Phase 2 result was green.

Because of the very low safety significance of the issue and because the issue has been entered into the licensee's CAP (IR 934800); the issue was treated as an NCV, consistent with Section VI.A.1, of the NRC Enforcement Policy.
(NCV 05000454/2009003-01).

At the close of this inspection period, the licensee had submitted a response which respectfully disagreed with the NRC position that this was pressure boundary leakage. NRC personnel are following the applicable procedures for a disputed violation.

The inspectors reviewed the LER and concluded it was completed in accordance with 10 CFR 50.73. Therefore, this LER is closed.

This event follow-up review constituted one sample as defined in IP 71153-05.

.2 Findings

(Closed) URI 05000454/2009003-02; 05000455/2009003-02: Diesel Oil Storage Tank Vent Lines Regulatory Compliance

Introduction: A finding of very low safety significance and associated NCV of 10 CFR 50, Appendix A, Criterion 2, "Design Basis for Protection Against Natural Phenomena," and Criterion 4, "Environmental and Natural Effects Design Bases," was identified by the inspectors for the failure to seismically support and protect from tornado generated missiles the EDG diesel oil storage tank (DOST) vent lines.

Description: As documented in NRC Inspection Report 2009003, issued on August 7, 2009, the inspectors concluded that the DOST vent piping was non-safety related and was located in a non-safety related structure. Subsequent inspector questions focused on the DOST's ability to vent if the vent lines were crimped during a seismic or tornado generated missile event. Therefore, the inspectors identified this issue as an URI pending further NRC review of the installed configuration and assessment of 10 CFR 50.109(a)(4) to determine if a modification was necessary to bring the facility into compliance with the rules or orders of the Commission
(URI 05000454/2009003-02; 05000455/2009003-02).

Analysis: The inspectors determined that the failure to seismically support or protect from tornado generated missiles the EDG DOST vent lines was contrary to 10 CFR 50, Appendix A and was a performance deficiency.

The performance deficiency was determined by the inspectors to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated December 4, 2008, because it was associated with the Mitigation System Cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring availability of the EDG to respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, General Design Criteria (GDC) 2, requires in part that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes and tornadoes. In addition, GDC 4, requires in part that structures, systems and components important to safety be designed to be appropriately protected against dynamic effects, including missiles. During a design basis event, the EDGs are postulated to start and continue operating as required for as long as 7 days. During EDG operation, the DOSTs would be significantly depleted and air would be required to enter the tanks in order to maintain the tanks at atmospheric pressure. Failure to maintain the tanks at or near atmospheric pressure could result in the failure of the fuel oil transfer pumps to maintain suction and/or could result in structural failure of the storage tanks. The EDG DOST vents are required in order for the EDGs to perform their required safety-related functions.

The inspectors determined that the finding could be evaluated using the SDP in accordance with Table 4a of IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings,", date January 10, 2008, for the Mitigation Systems Cornerstone. The finding screened to be of very low safety significance (Green) because the finding was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee performed an operability determination and concluded that even with the vent lines significantly degraded (crimped) or in the event that a seismic event caused the lines to fail completely, sufficient air would enter the tanks to replace the approximately one cubic foot per minute of fuel that would be used.

The inspectors did not identify a cross-cutting aspect associated with this finding because the performance deficiency occurred approximately 30 years ago; therefore, the finding is not reflective of current performance.

Enforcement: 10 CFR Part 50, Appendix A, GDC 2, requires in part that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes and tornadoes. In addition, GDC 4, requires in that part structures, systems and components important to safety be designed to be appropriately protected against dynamic effects, including missiles.

Contrary to the above, on August 7, 2009, the NRC inspectors determined that the licensee failed to install the EDG DOST vent lines in accordance with applicable design requirements. Specifically, the licensee failed to seismically support or protect from tornado generated missiles the EDG DOST vent lines. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (IR 877430), this violation is being treated as a NCV, consistent with

Section VI.A.1 of the NRC Enforcement Policy (NCV 050004542009004-02; 05000455/2009004-02).

URI 05000454/2009003-02; 05000455/2009003-02 is closed.

4OA5 Other Activities

.1 Preoperational Testing of an Independent Spent Fuel Storage Facility Installation (ISFSI) at Operating Plants (60854.1)

a. Inspection Scope

(1) Control of Heavy Loads

The inspectors reviewed the licensee's crane and heavy loads program with regards to ISFSI operations. The inspectors reviewed topics associated with the fuel handling building crane's hoisting system, wire rope, bridge and trolley, controls, crane inspection and maintenance, load testing, limit switches, operation, and safe load paths. The inspection consisted of documentation review, interviews with staff, and an inspection of the fuel handling building (FHB) crane.

As part of the modifications in preparations to load fuel in the ISFSI, the licensee upgraded the 125-ton capacity overhead crane in the FHB to a single failure proof crane in compliance with the NRC guidance, NUREG 0612, NUREG 0554 and ASME-NOG-1-2004. This involved installation of a new trolley. The inspectors reviewed the procedures and evaluations regarding safe storage and transfer of the spent fuel cask to and from the Spent Fuel Pool (SFP). The review included structural evaluations associated with the seismic design of the new trolley, hoist/reeving equipment, and miscellaneous components; crane bridge girders and supporting structural steel, special lifting devices and rigging components, modifications affecting the operating plant, floor loading in the spent fuel pool and other cask storage areas, and the cask and canister stack up configurations. The associated safety evaluations/screenings were also reviewed.

The inspectors reviewed that the reactor building crane had been static loaded to approximately 125 percent of the maximum critical load on its main hook. The inspectors verified that the default minimum crane operating temperature was defined as 70 degrees Fahrenheit in loading procedures.

The inspectors reviewed the crane's hoist brake system and observed the power control braking system and two holding brakes. Holding brakes were tested to automatically apply the full holding position when power is off, and under overspeed and overload conditions. The inspectors reviewed procedures for emergency positioning of the crane and lowering the load.

The Byron Station FHB crane employs a system of four independent upper travel limit switches to prevent two-blocking (lower block coming in contact with the drum). The hoist drum was equipped with drum capture plates put in place to limit drum drop during a shaft or bearing failure.

The inspectors reviewed the latest annual preventive maintenance program and crane inspection. During ISFSI operations, the FHB crane was categorized as being under normal service. This categorization required a "frequent check" on a monthly basis. The inspectors reviewed the crane's daily inspection list.

The inspectors performed a walk-down of the licensee's FHB crane. The inspectors observed the licensee test electrical interlocks that restrict movement of the crane over the SFP.

The licensee had several very long delays associated with the FHB crane and its upgrade to single failure proof, which prevented its use and delayed the dry run demonstrations as well as the loading campaign. The issues included not meeting the requirements of NUREG-0554 (IR 909647) in addition to several mechanical issues. These issues were mostly vendor issues (relay problems, etc.).

(2) Dry Run Demonstrations

The licensee performed selected dry run demonstrations in preparations to load fuel at the Byron Station. The NRC inspectors were onsite to observe such activities from March 2 through March 5, May 11 through 14, May 19 through 21, June 12, April 9 through 10, and June 12, 2009. The inspectors observed Multi-Purpose Canister (MPC) processing (all activities that were able to be demonstrated outside the reactor building and without use of the crane), reviewed the licensee's 72.212 report, performed crane walk-down inspection, observed welding demonstration and transportation of the transfer cask with empty cask to the ISFSI pad using the crawler, and reviewed documents.

The inspectors reviewed the loading and unloading procedures to ensure that they contained commitments and requirements specified in the license, the Certificate of Compliance (CoC) TS, the UFSAR, and 10 CFR 72. The inspectors observed the licensee's pre-job briefings and determined that the licensee conducted these meetings in a professional manner where the necessary items to enhance safety were discussed. RP staff attended pre-job briefs and gave insight into working conditions and ALARA practices.

The inspectors observed licensee personnel perform a number of activities associated with dry fuel storage to demonstrate their readiness to safely load spent fuel from the SFP into the dry cask storage system. The inspectors observed the use of the loading procedures pertaining to welding, and MPC processing (canister pressure testing, vacuum drying, and helium backfill). In addition, the inspectors observed the use of the licensee's unloading procedures including cut out of vent and siphon ports (video), and MPC re-flood operations.

The inspectors observed welding and weld non-destructive examination (NDE) on a mock-up cask. The inspectors observed fit-up, tack weld, and machine welding methods as well as visual inspections and dry penetrant inspections of welds. Through a review of records, the inspectors verified that welders, weld procedures, and procedure qualification records met the code requirements and were properly qualified. The inspectors sampled weld material and base metal material and verified they met ASME code and fabrication specification requirements.

The inspectors reviewed welding and NDE procedures related to undergoing fabrication to verify that they were in compliance with design and/or code specifications. The inspectors also reviewed controls on calibrated equipment used for both quality control inspection and fabrication activities. During the dry run demonstration, the inspectors identified that the licensee's contractor was not following the welding procedures in the order written on the procedure and provided no explanations to justify any changes. Licensee had a stand down and discussions with the workers on the ISFSI project prior to resuming activities (IR 905447). Since this occurred during the dry run demonstration, there was no violation of NRC requirement.

The inspectors observed transfer of the HI-STORM overpack from the FHB to the ISFSI pad via the haul path and its placement on its proper location on the ISFSI pad. Proper controls were in place during the transfer of the canister from the FHB to the ISFSI. These controls included health physics coverage and adherence to the heavy haul path. The inspectors verified adequate communication and team work between departments and adherence to procedures.

(3) Fuel Selection

The inspectors reviewed the licensee's processes and methods associated with fuel characterization and selection and interviewed Exelon corporate fuel personnel regarding the CASKLOADER database used in selection of the fuel assemblies to be loaded. The inspectors reviewed a completed fuel selection package for the casks to be loaded during the campaign to verify that the licensee used the criteria specified in the CoC TS to verify the acceptability of assemblies to be loaded in a cask. The inspectors observed the licensee's use of the CASKLOADER database. The licensee did not plan to load any damaged fuel assemblies during this campaign. Since the licensee postponed its plans to load fuel, the fuel characterization documents (procedures, loading plan, etc.) will be reviewed again prior to the 2010 campaign to ensure the CoC TS are met and no damaged fuel will be loaded.

(4) Radiation Protection

The inspectors evaluated the licensee's RP program pertaining to the operation of the ISFSI. The inspectors reviewed the licensee's procedures describing the methods and techniques used when performing dose rate and surface contamination surveys and verified that they ensured dose rate limits and surveillance requirements of the TS were met. The licensee's RP staff considered lessons learned from other utilities' spent fuel loading campaigns during development of the radiological controls for the Byron Station loading operations. Based on the review, the inspectors determined that licensee had established procedures, engineering and work controls that were based on sound RP principles in order to achieve occupational exposures that were ALARA. The inspectors interviewed the licensee's personnel to verify their knowledge regarding the scope of the work and the radiological hazards associated with transfer and storage of spent fuel.

The inspectors identified that the procedure used to take survey readings of the overpack on the pad was confusing to the RP staff. The procedure required the RP staff to take four readings at four set locations around the circumference of the overpack. It then had the staff take these same four readings at three different locations along the length of the overpack. The inspectors asked several clarifying questions since there was confusion when the RP staff was notating the survey instrument readings. The manner in which the survey map was organized made it easy to notate the survey reading in the incorrect location on the survey map. The licensee planned to revise the procedure to make the locations of the surveys and survey entries more clear.

(5) Training

The inspectors reviewed the licensee's training program, which consisted of classroom and on-the-job training to ensure involved staff was adequately trained for the job they were responsible to perform. The licensee's contractor prepared a dry fuel storage qualification matrix which documented each worker's training courses completed.

The inspectors reviewed training records of welders and other personnel who the licensee authorized to perform the NDE inspections to ensure that these individuals' training was current. These qualifications will be reviewed again before the licensee proceeds with the remainder of its dry run and loading campaign activities.

(6) Quality Assurance

The inspectors reviewed the licensee's Quality Assurance program, as it applied to the ISFSI. The inspectors also reviewed procedures pertaining to the receipt inspection of MPCs. The inspectors observed that gauges were within their calibration date, and that 99.999 percent pure helium was used during backfilling.

(7) Emergency Preparedness and Fire Protection

The inspectors reviewed the licensee's emergency preparedness plan required by 10 CFR 50.47 for conformance with 10 CFR 72.32(c). The inspectors verified that the licensee incorporated Emergency Action Levels to the plant emergency plan to address the possible emergency scenarios, their classification, and recovery actions associated with the ISFSI.

b. Findings

(1) Inadequate Seismic Restraint on the Fuel Handling Building Crane Trolley

Introduction: A finding of very low safety-significance and associated NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for failure to perform an adequate evaluation of seismic restraint on the FHB crane trolley. Specifically, for evaluation of the seismic restraint in their single failure proof trolley analysis, the licensee failed to use adequate seismic acceleration values and failed to evaluate the connections for resulting reaction forces. Subsequent review found that the restraint design was inadequate. The licensee documented the condition in IR 934467 and initiated actions for calculation revision and installation of a field modification.

Description: During review of Calculation 36272-12, "Exelon/Byron and Braidwood Single Failure Proof Trolley Seismic Analysis," Revision 9, the inspectors identified deficiencies in the evaluation of the trolley seismic restraint in accordance with the requirements of ASME NOG-1-2004. In determining the transverse load applied to the restraint, the force required to overcome friction was deducted from the total seismic transverse load acting at the wheel-rail interface. The calculation did not take into consideration the upward seismic force in calculation of friction and thus overestimated the frictional effect. This resulted in a non-conservative design of the restraint due to a smaller load being applied. In addition, the calculation did not address the shear lugs and bolts connecting the restraint to the trolley frame structure. Upon identification by the inspectors, the licensee documented the condition in IR 934467. Subsequent review by the licensee concluded that the restraint and the connection welds were overstressed by 97 percent and 67 percent, respectively, when subjected to seismic loads. The licensee initiated actions for calculation revision and installation of a field modification to the seismic restraint.

Analysis: The inspectors determined that the failure to use adequate seismic accelerations values and to evaluate the connections for seismic restraint for single failure proof trolley analysis was contrary to ASME NOG-1-2004 design requirements and was a performance deficiency. The FHB crane is designed to Seismic Category I requirements and the licensee used compliance with ASME NOG-1-2004 as the design basis for their upgrade to a single failure proof crane.

The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of Equipment Performance, Refueling/Fuel Handling equipment, and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the NRC has endorsed compliance with the NOG-1-2004 design requirements as a means to demonstrate safe load handling of heavy loads over the reactor core or over safety-related systems by providing a single failure proof crane.

The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Initiating Events cornerstone. Based on a "No" answer to all the questions in the Initiating Events cornerstone column of Table 4a, the finding was determined to be of very low safety-significance. Specifically, the crane had not been used in operation as a single failure proof crane.

This finding has a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not provide adequate oversight of work activities, including contractors, such that nuclear safety is supported. Specifically, the licensee's owner review process failed to identify that the contractor's evaluation did not address the design of the seismic restraint connections and did not account for effect of vertical seismic acceleration in determining adequacy of the trolley seismic restraint. (H.4(c))

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, on June 3, 2009, the licensee's design control measures failed to verify adequacy of the design for the FHB crane trolley seismic restraint. Specifically, in Calculation 36272-12, the licensee failed to evaluate the seismic restraint components and connections for design basis loading in accordance with ASME-NOG-1-2004, design requirements. The FHB crane is designed to Seismic Category I requirements and the licensee used compliance with ASME NOG-1-2004, as basis for its upgrade to a single failure proof crane. Because this violation was of very low safety-significance and it was entered into the licensee's corrective action program as IR 934467, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000454/2009004-03; 05000455/2009004-03)

(2) Use of Friction in Seismic Analysis for the FHB Crane Trolley May Not Meet Design Requirements

Introduction: An URI was identified by the inspectors for the licensee's use of friction in their seismic analysis methodology that may not be consistent with the design requirements of ASME NOG-1-2004. Specifically, in the seismic analysis of the FHB trolley and components, the licensee used friction force developed at the trolley wheel/rail interface to reduce the horizontal seismic loads applied to the trolley components. Since Table 4154.3-1 of ASME NOG-1-2004 specifies analytical boundary conditions that would prevent sliding between the trolley and the rails, the licensee's use of friction is contrary to the requirements of the ASME NOG-1-2004 Rules for Construction of Overhead and Gantry Cranes. The NRC staff is currently reviewing the appropriateness of the licensee's application of friction with respect to the design requirements specified in ASME NOG-1-2004.

Description: Seismic evaluations of the new single failure proof crane trolley and its components are included in Calculations 36272-02, 36272-12, 36272-13, and 36272-17. The inspectors identified that for evaluation of trolley components in Calculation 36272-17, instead of using seismic loads based on accelerations obtained from the crane dynamic analysis performed in accordance with ASME NOG-1-2004, the licensee used much smaller loads limited by the frictional force at the rail surface. This resulted in a significant load reduction for qualification of the trolley components, and is contrary to the ASME NOG-1-2004, requirements for the boundary conditions to be used in dynamic analysis model.

The use of friction was first identified during review of calculations at Exelon's LaSalle County Station and is currently being reviewed by the inspectors and NRR staff. The issue is captured in IR 957014 for LaSalle. The licensee subsequently issued IR 966184 for addressing the impact at Byron. Per IR 966184, the license identified that Calculations 36272-12 and 36272-17 utilized friction and the licensee is in the process of revising these and any other affected calculations to remove the use of friction methodology. All revised calculations issued to resolve this concern need to be reviewed. Note that since the licensee has decided to revise the affected calculation, the NRC review may be performed as part of the URI described in Section 4AO5.1.b(3), "Review of FHB Crane Trolley Upgrade Design Document Not Completed."

This issue will be a URI pending the inspectors' reviews of the licensee's resolution of the use of friction in their design of structures and components in the upgrade of the FHB crane to single failure proof (URI 005000454/2009004-04; 05000455/2009004-04).

(3) Review of FHB Crane Trolley Upgrade Design Documents Not Completed

Introduction: A URI was identified by the inspectors for their incomplete review of the design documents related to the FHB crane upgrade to single failure proof. Specifically, not all of the licensee's design and modification documents required to complete the inspection were complete at the conclusion of the inspection.

Description: The licensee performed the following calculations to demonstrate that the crane trolley seismic design is in accordance with ASME NOG-1-2004.

- 36272-02; Exelon/Byron and Braidwood Hoist Reeving Equipment Calculation;
- 36272-12; Exelon/Byron and Braidwood Single Failure Proof Trolley Seismic Analysis;
- 36272-13; Exelon/Byron and Braidwood Single Failure Proof Trolley Critical Weld Calculations; and
- 36272-17; Exelon/Byron and Braidwood Single Failure Proof Trolley Misc. Item Seismic Analysis.

At conclusion of the inspections, the licensee had not completed all the necessary design activities and there were open technical concerns regarding adequacy of the seismic restraint evaluated in Calculation 36272-12 as described in Section 4OA5.1.b(1), "Inadequate Seismic Restraint on Fuel Handling Building Crane Trolley." In addition, resolution of the open technical concern regarding use of friction to limit seismic loads, as described in Section 4OA5.1.b (2), "Use of Friction in Seismic Analysis…," may impact some of the above calculations. All revised calculations from those listed above and any new design documents including calculations and modifications issued to resolve these technical concerns will need to be reviewed.

This issue will be a URI pending the inspectors' reviews of the licensee's design documents demonstrating resolution of the technical concerns identified during the inspection (URI 05000454/2009004-05; 05000455/2009004-05).

.2 Review of 10 CFR 72.212(b) Evaluations (60856)

a. Inspection Scope

The inspectors evaluated Byron Station's compliance with the requirements of 10 CFR 72.212 and 10 CFR 72.48. The inspection consisted of interviews with cognizant personnel and review of licensee documentation. The licensee is required as specified in 10 CFR 72.212(b)(1)(i) to notify the NRC of the intent to store spent fuel at the Byron ISFSI facility at least 90 days prior to the first storage of spent fuel. Byron Station notified the NRC on January 23, 2009 (Letter No. BYRON 2009-0011) within the 90 days of their intent to store spent fuel into Holtec HI-STAR 100 Cask System according to CoC No. 72-1014, Revision 3, associated Safety Evaluation Reports and HI-Storm 100 Final Safety Analysis Report (FSAR) HI-2002444, Revision 5. The licensee is required as specified in 10 CFR 72.212(b)(1)(ii) to register the use of each cask with the NRC within 30 days of using that cask to store spent fuel. At the time of the inspection Exelon (Byron) had not provided this registration because the ISFSI Dry Run activities were not yet completed.

A written evaluation is required per 10 CFR 72.212(b)(2)(i), prior to use, to establish that the conditions of the CoC have been met. "Byron Nuclear Power Station Units 1 and 2 10 CFR 72.212 Evaluation Report," dated May 7, 2009, documented the evaluations performed by Exelon Generating Company, LLC prior to use of the 10 CFR Part 72 general License. The licensee had performed written evaluations which confirmed that the conditions set forth in the CoC had been met. The review was based on the Units 1 and 2 10 CFR Part 50 Licenses and TSs, FSARs, and other Byron design and licensing basis information, as referenced within the 72.212 Evaluation report.

The inspectors verified that the licensee had performed written evaluations which confirmed that the conditions set forth in the CoC had been met, the ISFSI pad had been designed to support the load of stored casks, and the requirements of 72.104 had been met. The inspectors determined that applicable reactor site parameters, such as fire and explosions, tornadoes, wind-generated missile impacts, seismic qualifications, lightning, flooding and temperature, had been evaluated for acceptability with bounding values specified in the Holtec HI-STORM 100 FSAR and associated analysis.

The inspectors noted that a 50.59 evaluation of the construction and operation of the Byron ISFSI and plant interfaces had been performed to demonstrate that changes to plant TS, or a license amendment were not required and that ISFSI related work activities would not impact the safe operation of Byron Nuclear Power Station.

The inspectors reviewed selected procedure changes related to the emergency preparedness, quality assurance, training, and health physics programs as well as determining their irretrievability and control with the licensee's processes. The emergency plan, quality assurance program, radiological safety program, and training program had been evaluated and their effectiveness were determined not to be decreased by ISFSI activities. However, the licensee acknowledged that enhancements were needed and that certain existing evaluations and/or calculations required changes and that final implementation would not occur until resolution of such issues prior to loading.

The inspectors interviewed the licensee's staff to determine if they were knowledgeable about the impacts of ISFSI activities. The inspectors determined that the appropriate programs had been reviewed and the determinations were found to be acceptable.

The licensee documented the required evaluations and developed an extensive set of procedures to control ISFSI-related work activities. During the review the inspectors noted some minor discrepancies and coordinated with Byron Staff for subsequent correction in the "Byron Station 10 CFR 72.212 Evaluation," dated May 7, 2009.

b. <u>Findings</u>

No findings of significance were identified.

.3 <u>(Open) NRC Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"</u>

a. <u>Inspection Scope</u>

On September 30, 2009, the inspectors completed a walkdown of Unit 1 RH system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspector's independent walkdown (TI 2515/177, Section 04.02.c.3).

In addition, the inspectors verified that the licensee had isometric drawings that describe the Unit 1 RH system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings:

- High point vents were identified;
- High points that do not have vents were acceptably recognizable;
- Other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation;
- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified;
- All pipes and fittings were clearly shown; and
- The drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the CAP for resolution.

The inspectors verified that Piping and Instrumentation Diagrams (P&IDs) accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the P&IDs were documented and entered into the CAP for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the Attachment to this report.

This inspection effort counts towards the completion of TI 2515/177 which will be closed on a later inspection report.

b. Findings

No findings of significance were identified.

.4 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.5 (Closed) VIO[1] (05000454/2008003-07; 05000455/2008003-07); Design Basis Re-Analysis of the Ultimate Heat Sink

As documented in Inspection Reports 05000454/2008008; 05000455/2008008, 05000454/2008003; 05000455/2008003, and 05000454/2009002; 05000455/2009002, the inspectors identified that the licensee did not consider spurious failure/opening of the 4160 volt or 480 Vac as a valid single failure in Amendment No. 95. The inspectors further noted that the NRC did not evaluate the potential for a passive failure of the electrical breakers even though passive failures were required to be evaluated under 10 CFR Part 50, Appendix A. After further review, the inspectors determined that the provisions of 10 CFR 50.109 (a)(4), were applicable and that a modification is necessary to bring a facility into compliance with the rules or orders of the Commission.

On November 3, 2008, the licensee determined that the 10 CFR 50.59 evaluation of the re-analysis results concluded that a TS revision would be necessary. Consequently, on June 30, 2009, the licensee submitted a License Amendment Request to get the necessary NRC approval prior to revising the Ultimate Heat Sink design basis.

The inspectors reviewed the extent of condition evaluation that resulted from the licensee's corrective actions. This extent of condition evaluation included all safety related systems that were allowed to normally operate with a crosstie and had components common to both units. Only two systems met these criteria, Component Cooling system (CC) and Control Room Ventilation system (VC). The licensee identified an issue with the ability to separate the CC system by train while maintaining net positive suction head to the Unit 0 pump due to a surge tank design deficiency. The licensee will address this vulnerability through a modification and resolution of this issue is being tracked under IR 924875. No single failure criteria issues were identified for the VC system.

As a result of the inspectors' review of the extent of condition evaluation and the licensee's submittal of the license amendment request, the inspectors have no outstanding concerns with the issues involved in this backfit. This issue is closed. It should be noted that a separate unresolved item (URI 05000454/2009007-03; 05000455/2009007-03) relating to the application of single failures to the Chapter 15 accident analysis remains open.

4OA6 Management Meetings

.1 Exit Meeting Summary

On September 30, 2009, the inspectors presented the inspection results to Mr. D. Enright and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

[1] Note: In Inspection Report 05000454/2008003-07; 05000455/2008003-07, the item was identified as "OTHER." In accordance with IMC 0612, the classification was revised to VIO.

.2 Interim Debriefs

Interim debriefs for the ISFSI dry run readiness inspections were conducted on March 5, April 10, May 11, May 14, May 21, and June 12, 2009. The inspectors presented the inspection results to members of the licensee management and staff. Licensee personnel acknowledged the information presented.

.3 Interim Exit Meetings

Interim exits were conducted for:

- The inspection activities for Radioactive Material Processing and Transportation under the Public Radiation Safety cornerstone with Mr. D. Enright and other members of the licensee's staff on August 07, 2009.

- The inspection activities performed under IP 60854.1 was held on September 9, 2009. The inspectors presented the inspection results to members of the licensee management and staff. Licensee personnel acknowledged the information presented.

- A review of the access control to the radiologically significant areas and ALARA planning and control under the occupational radiation safety cornerstone with Mr. Dan Enright on September 25, 2009.

- The inspection activities performed under Inspection Procedure 71111.08 was held on September 30, 2009. The inspectors presented the inspection results to Mr. D. Enright and other members of the licensee staff. The licensee acknowledged the issues presented.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Cornerstone: Occupational Radiation Safety

- Technical Specification 5.7.1.b states that any individual permitted to enter high radiation areas shall be provided with or accompanied by a radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Contrary to this, a worker entered "Area Five," a posted high radiation area (HRA) area to perform boric acid cleaning on a valve, left his electronic dosimeter (ED) on the bench during a dress-out process. The worker worked in the HRA area for approximately 15 minutes without the ED radiation monitoring device. The worker was identified and escorted out of the radiological controlled are (RCA). This event was entered into the licensee's Corrective Action Program as IR 966917. The RP department immediately took control by escorting the worker from the RCA and removed worker RCA access pending investigation. The licensee conducted a human performance investigation on this event. The issue is of very low safety significance because it did not involve

ALARA planning or work controls, an overexposure, substantial potential for overexposure, or limit the ability to assess radiation dose.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Enright, Site Vice President
J. Anderson, Byron ISFSI Project Manager
B. Barton, Radiation Engineering Manager
E. Blondin, Mechanical Design Manager
G. Contrady, Programs Manager
S. Greenlee, Engineering Director
D. Anthony, NDE Manager
B. Grundmann, Corporate Licensing, Cantera
D. Gudger, Regulatory Assurance Manager
T. Spelde, Byron ISFSI Project Manager
D. Thompson, Radiation Protection Manager

Nuclear Regulatory Commission

R. Skokowski, Branch Chief

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000455/2009-001-00	LER	Late Entry into TS Condition associated with Reactor Coolant System Leakage Characterization resulting in a condition prohibited by TSs (Section 4OA3.1)
05000454/2009-004-01 05000455/2009-004-01	URI	Assigning appropriate 10 CFR 61 waste stream to radioactive waste shipments (Section 2PS2.4)
05000454/2009-004-02 05000455/2009-004-02	NCV	Diesel Oil Storage Vents Not Seismically Qualified or Tornado Resistant (Section 4OA3.2)
05000454/2009-004-03 05000455/2009-004-03	NCV	Inadequate Evaluation of Seismic Restraint on the FHB Crane Trolley (Section 4OA5.1)
05000454/2009-004-04 05000455/2009-004-04	URI	Use of Friction in Design of FHB Crane to Single Failure Proof (Section 4OA5.1)
05000454/2009-004-05 05000455/2009-004-05	URI	Unresolved Technical Concerns on Design of Seismic Restraint on FHB Crane Trolley (Section 4OA5.1)

Closed

0500455/2009-001-00	LER	Late Entry into TS Condition associated with Reactor Coolant System Leakage Characterization resulting in a condition prohibited by TSs (Section 4OA3.1)

05000454/2009-004-02 05000455/2009-004-02	NCV	Diesel Oil Storage Vents Not Seismically Qualified or Tornado Resistant (Section 4OA3.2)
05000454/2009-004-03 05000455/2009-004-03	NCV	Inadequate Evaluation of Seismic Restraint on the FHB Crane Trolley (Section 4OA5.1)
05000454/2009-003-02 05000455/2009-003-02	URI	Diesel Oil Storage Vents Not Seismically Qualified or Tornado Resistant (Section 4OA5.1)
05000454/2008003-07 05000455/2008003-07	VIO	Design Basis Re-Analysis of the Ultimate Heat Sink (Section 4OA5.5)

Discussed

05000454/2009-007-03 05000455/2009-007-03	URI	Application of Single Failures to the Chapter 15 Accident (Section 4OA5.5)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Section 1R04: Equipment Alignment (Quarterly)

Drawing M-61, Sheet 1A, Revision AQ; Diagram of Safety Injection
Drawing M-61, Sheet 3, Revision AH; Diagram of Safety Injection
BOP SI-M1A; Train A Safety Injection System Valve Lineup, Revision 2
BOP SI-M1; Safety Injection System Valve Lineup, Revision 21
BOP SI-E1A; Safety Injection System Train A Electrical Lineup, Revision 2
BOP SI-E1; Safety Injection System Electrical Lineup, Revision 7
BOP CS-M1A; Containment Spray System Train A Valve Lineup, Revision 2
BOP CS-M1; Containment Spray System Valve Lineup, Revision 13
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Weldstar Certificate of Compliance; Weld Material; July 2, 2008
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LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CC	Component Cooling System
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DOST	Diesel Oil Storage Tank
DOT	U.S. Department of Transportation
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report (Holtec)
GDC	General Design Criteria
HAZMAT	Hazardous Materials
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report/Inspection Report
ISFSI	Independent Spent Fuel Storage Installation
ISI	Inservice Inspection
LER	Licensee Event Report
LSIV	Loop Stop Isolation Valve
MPC	Multi-Purpose Canister
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Model
OSP	Outage Safety Plan
P&ID	Piping and Instrumentation Diagram
PARS	Publicly Available Records
PI	Performance Indicator
Radwaste	Radioactive Waste
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specifications
RFO	Refueling Outage
RH	Residual Heat Removal
RP	Radiation Protection
RWP	Radiation Work Permit
SDP	Significance Determination Process
SFP	Spend Fuel Pool
SG	Steam Generator
SI	Safety Injection
SSPS	Solid State Protection System
SX	Essential Service Water
TI	Temporary Instruction
TS	Technical Specification

UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VC	Control Room Ventilation System
WO	Work Order

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Sincerely,

/RA/

Richard A. Skokowski, Chief
Branch 3
Division of Reactor Projects

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

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Letter to C. Pardee from R. Skokowski dated November 5, 2009.

SUBJECT: BYRON STATION, UNITS 1 AND 2 INTEGRATED INSPECTION REPORT 05000454/2009004; 05000455/2009004

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