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

November 13, 2009

EA-09-134

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Energy Kewaunee, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION INTEGRATED INSPECTION REPORT 05000305/2009004

Dear Mr. Heacock:

On September 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Kewaunee Power Station. The enclosed report documents the inspection findings, which were discussed on October 7, 2009, with Mr. Steve Scace 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, four NRC-identified findings of very low safety significance (Green) 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, which was the subject of an investigation by the NRC Office of Investigations, was reviewed by the inspectors and is listed in Section 4OA7 of this report. This Severity Level IV violation substantiated that a lead chemistry technician deliberately violated training requirements by completing a radiological liquid waste release discharge permit knowing that his qualifications were not current. You have entered this issue into your corrective action program and have taken appropriate corrective actions.

If you contest the subject or severity of these NCVs, 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 Kewaunee Power 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 Kewaunee Power 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael A. Kunowski, Chief
Branch 5
Division of Reactor Projects

Docket No. 50-305
License No. DPR-43

Enclosure: Inspection Report 05000305/2009004
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-305
License No:	DPR-43
Report No:	05000305/2009004
Licensee:	Dominion Energy Kewaunee, Inc.
Facility:	Kewaunee Power Station
Location:	Kewaunee, WI
Dates:	July 1, 2009, through September 30, 2009
Inspectors:	S. Burton, Senior Resident Inspector K. Barclay, Resident Inspector J. Cassidy, Senior Health Physicist T. Bilik, Regional Inspector E. Sanchez-Santiago, Reactor Engineer S. Bakhsh, Lead ISFSI Inspector M. Learn, Reactor Engineer S. Atwater, Senior Project Inspector, Region II V. Meghani, Reactor Inspector J. Neurauter, Senior Reactor Inspector J. Pearson, Senior Safety Inspector E. Love, Safety Inspector
Approved by:	Michael A. Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000305/2009004; 07/01/2009 – 09/30/2009; Kewaunee Power Station; Post-Maintenance Testing; and Preoperational Testing of an Independent Spent Fuel Storage Installation (ISFSI) at Operating Plants.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Additionally, the report covers a pre-operational testing and initial loading inspection by inspectors of the ISFSI activities at the Kewaunee Power Station. Three Green findings and one Severity Level IV finding were identified by the inspectors. The findings were considered Non-Cited Violations (NCVs) 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 NCV of Title 10 Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the licensee’s failure to provide an adequate single failure proof design basis analysis for the 105-ton transfer cask-lifting beam. The licensee entered this issue into their corrective action program as condition report CR339267. The licensee revised the design calculation for the 105-ton transfer cask-lifting beam and demonstrated compliance with single failure proof acceptance criteria.

  The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of equipment performance and adversely 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 finding was determined to be of very low safety significance by the NRC’s significance determination process because the transfer cask-lifting beam had not been previously used at the Kewaunee Power Station. This finding has a cross-cutting aspect in the area of human performance, work practices, because the licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported, in that, the licensee failed to perform an effective owner’s review to assure that appropriate design methods are used in calculations that demonstrate nuclear safety (H.4(c)). (Section 4OA5.1)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety-significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to perform a power system analysis calculation that would have identified that the fast transfer design feature/scheme was deficient, in that, it allowed an unanalyzed electrical power system alignment where both redundant 4.16-kiloVolt safety-related buses were being supplied by an offsite source via the same transformer. Use of this electrical configuration could have resulted in an out-of-phase transfer, loss of available offsite power to the buses and potential damaging effects on redundant safety-related equipment, during a design basis event such as initiation of safety injection signal. When identified, the licensee entered this issue into their corrective action program and implemented interim actions to prohibit use of the fast transfer feature or manually aligning two safety-related buses to be fed from the same transformer during plant operation.

This performance deficiency was more than minor because the failure to perform the required calculation resulted in a condition where the plant was being operated in an unanalyzed configuration where there was reasonable doubt as to the operability of redundant safeguard loads; this concern resulted in issuance of a Licensee Event Report (LER) 2007-007-00 on May 21, 2007. Consequently, the potential for damage or loss of power to safety-related loads during an event could have led to unacceptable consequences. The finding screened as being of very low safety-significance (Green) for the Initiating Events Cornerstone because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions will not be available. The inspectors did not identify a cross-cutting aspect associated with this finding because the cause of the performance deficiency was related to a historical design issue and not indicative of current licensee performance. (Section 1R21.b)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to have adequate procedures that ensured technical specifications were entered and followed for containment isolation valves. The licensee entered the issue into their corrective action program as CR344856 and CR350526A, and provided additional guidance to operations personnel. At the end of the inspection period, the licensee continued to perform a causal analysis.

  The inspectors determined that the issue was more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, not entering the appropriate technical specification action requirements, when necessary, would lead to more significant safety concerns. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Barrier Integrity Cornerstone. The inspectors answered no to the Barrier Integrity questions and screened the finding as having very low safety significance (Green). The finding has a cross-cutting aspect in the area of human performance, resources, because the licensee did not have complete, accurate and up-to-date design documentation, procedures and work packages (H.2(c)). (Section 1R19.b)

Cornerstone: Not Applicable/Other

- Severity Level IV. The inspectors identified a Severity Level IV NCV of 10 CFR 72.150, "Instructions, Procedures, and Drawings," during the Independent Spent Fuel Storage Installation loading campaign. The licensee failed to follow procedure OP-KW-NOP-ISF-001, "Dry Shielded Canister Loading." The inspectors determined that the licensee's failure to follow step 5.2.6 of Procedure OP-KW-NOP-ISF-001 to perform a crane brake check was contrary to 10 CFR 72.150. The licensee immediately evaluated the situation and discussed the need to check the crane brakes when lifting loads approaching the rated loads with the refueling crew to prevent missing this step in the future.

The inspectors determined that the violation had more than minor safety significance because the failure to check the crane brakes, results in not knowing if the brakes are functioning properly, which may lead to a failure of the brakes while lifting a loaded spent fuel canister. The issue was addressed by traditional enforcement since 10 CFR Part 72 is not risk based and is not covered under the reactor oversight process. Because this violation was of very low safety significance, was non-repetitive and non-willful, and was entered into the corrective action program, this violation is being treated as a NCV of 10 CFR 72.150 consistent with Section VI.A.1 of the Enforcement Policy.
The inspectors determined that there was no cross-cutting aspect associated with this finding. (Section 4OA5.3)

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by the licensee have been reviewed by the inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee’s corrective action program. The violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Kewaunee operated at full power for the entire inspection period except for brief downpowers to conduct planned maintenance and surveillance activities until September 24, 2009, when they began a coast-down for the refueling outage. Kewaunee reduced power to approximately 65 percent power on September 25, 2009, to conduct steam generator safety testing before shutting down on September 26th. Kewaunee was shutdown the remainder of the quarter for the refueling outage.

1. REACTOR SAFETY

Cornerstones: 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:

- safety injection train "B" after system surveillance;
- internal containment spray; and
- spent fuel pool (SFP) train "B" with train "A" out for maintenance.

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, the Updated Safety Analysis Report (USAR), Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports (CRs), 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 correctly aligned 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 to this report.

These activities constituted three partial system walkdown samples as defined in Inspection Procedure (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:

- fire zone AX-24/AX-27, waste handling area;
- fire zone AX-30, relay room and loft;
- fire zone TU-94, carbon dioxide storage tank room; and
- fire zone AX-22/AX-33/AX-39, condensate storage and reactor make-up water storage room and adjacent areas.

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 to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use, 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 CAP. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On July 7, 2009, the inspectors observed a fire brigade activation after a simulated flammable liquid fire started in the warehouse flammable materials storage room. Based on this observation, the inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies; openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate firefighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre planned strategies; (9) adherence to the pre planned drill scenario; and (10) drill objectives. Documents reviewed are listed in the Attachment to this report.

These activities constituted one annual fire protection inspection sample as defined in IP 71111.05-05.

b. Findings

No findings of significance were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the USAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area(s) to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- flood zone 301, control room air conditioning room.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P)

From September 26, 2009, through September 30, 2009, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection Program for monitoring degradation of the reactor coolant system (RCS), risk-significant piping and components and containment systems.

The inspection described in Sections 1R08.1 below constituted the resident inspection portion of the inservice inspection sample for outage activities as defined in IP 71111.08-05.

.1 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

On September 26, 2009, the inspectors observed the licensee's BACC visual examinations for portions of the reactor coolant and emergency core cooling systems and verified whether these visual examinations emphasized locations where boric acid leaks can cause degradation of safety-significant components.

Documents reviewed 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 17, 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:

- radiation monitoring;
- safety injection; and
- fire protection.

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 (SSCs)/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 that were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted three 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:

- risk management during un-modeled auxiliary feedwater flow instrument calibration;
- risk management during routine and planned activities; and
- risk management during discovery and emergent repair of the “B” containment fan coil unit (CFCU) service water leak.

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. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

Technical Specification Action Requirements During a Leak in a CFCU Unit Service Water Line

Introduction: The inspectors identified that the licensee responded differently for two similar CFCU leaks and were concerned that the licensee was in a condition prohibited by TSs during an August 15, 2008, CFCU service water leak.

Discussion: The inspectors identified that the licensee entered TS action requirement 3.0.c, standard shutdown sequence, for a leak inside containment on a CFCU service water line on September 13, 2009. The inspectors reviewed a similar leak that occurred on August 15, 2008, and found that the licensee did not enter the same TS action requirement for that leak. The inspectors asked the licensee to explain the different responses to the two events and the licensee was still evaluating the issue at the conclusion of the inspection period. This issue is unresolved pending additional information required to determine if there is a performance deficiency.
(Unresolved Item (URI) 05000305/2009004-01)

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- diesel fuel oil tank vent seismic qualification;
- electrical containment penetration qualification;
- pressurizer power operated relief valve PR-2B, control cable spurious operation;
- power lead to containment fan “A” found damaged;
- pins worn on back-draft damper;
- possible calculational error for condensate storage tank maximum temperature; and
- containment fan “A” motor remains operable after work performed by nonsafety-related vendor.

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 TSs and USAR 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 seven samples as defined in IP 71111.15-05.

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 tests of the following components to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- emergency diesel generator (EDG) “A” starting air check valve replacement;
- steam generator blowdown heat exchanger;

- MS-100A control power transformer and fuse replacement;
- post-maintenance test of 480-Volt breakers MCC52E-C3 and D-4 after maintenance; and
- post-maintenance test of the "A" residual heat removal train after a system vent valve installation.

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 TSs, the USAR, 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 five post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

(1) Containment Isolation Valve Inoperable With No Technical Specification Action Requirement Entry

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the licensee's failure to have adequate procedures to ensure TSs were entered and followed for containment isolation valves. The specific example that inspectors identified was that the containment isolation valve MS-100A, the steam generator 1A steam supply isolation valve to the turbine-driven auxiliary feedwater (TDAFW) pump, was not designated as a containment isolation valve in procedures or the USAR. As a result, operators failed to enter the appropriate TS action requirement on multiple occasions.

Description: On August 12, 2009, the inspectors observed a post-maintenance test of MS-100A, the steam generator 1A steam supply isolation valve to the TDAFW pump, after the control power transformer was replaced. The inspectors expected to find the valve closed prior to the start of the test, but when the inspectors arrived in the control room the valve was open. The inspectors questioned why the valve was in the open position with the control power transformer removed for maintenance because the valve could not be closed remotely. The licensee stated that TS 3.4.b.4.c applied to the plant configuration with one steam supply to the TDAFW pump being inoperable. The inspectors then asked why the containment isolation TS action requirement was also not entered; the licensee reiterated that the only action requirement necessary was TS 3.4.b.4.c.

The inspectors reviewed the USAR and found that it included a table of containment penetrations and associated containment isolation valves. However, containment penetration 6W, the “A” main steam header penetration, only listed the main steam isolation valves and the associated bypass valves as containment isolation valves. The inspectors also found that the list had a statement at the top, which cautioned that the list was not all inclusive and more containment isolation valves may exist.

The inspectors reviewed the inservice testing (IST) basis valve data sheet for MS-100A and found that it did not have the containment isolation valve designator checked at the top of the form. The inspectors reviewed the valve safety-function section of the sheet and found the valve performed an active safety function in the closed direction. Valve MS-100A must be capable of closure by a remote manual switch subsequent to a main steam line break upstream of the main steam isolation valve MS-1A. Additionally, the basis document indicated that remote manual closure of MS-100A isolates the affected steam generator after a steam generator tube rupture, thereby preventing an uncontrolled release of radioactive steam to the atmosphere from the TDAFW pump exhaust. The IST basis further stated that MS-100A met the Class 4 containment isolation criteria, as defined in USAR 5.3.2 and was required to close for containment integrity.

The inspectors presented the information to the licensee and the licensee agreed that they should have entered the containment isolation TS action requirement 3.6.b.3.c when MS-100A was inoperable. The licensee also reviewed the last three years of operation and found other examples where MS-100A or B were inoperable and the containment isolation action requirement was not entered. The licensee did not identify any examples where they exceeded the TS allowed action requirement times.

The licensee entered this issue into their corrective action process.

<u>Analysis</u>: The inspectors determined that the failure by the licensee to identify MS-100A as a containment isolation valve in either procedures or the USAR, and the subsequent failure to enter the appropriate technical specification action requirement when that valve was inoperable was a performance deficiency warranting further review. The inspectors determined that the issue was more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, not entering the appropriate TS action requirements, when necessary, could result in exceeding the TS allowed outage time and not taking the appropriate TS required actions. The inspectors concluded this finding was associated with the Barrier Integrity Cornerstone.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 4a for the Barrier Integrity Cornerstone. The inspectors answered no to the Barrier Integrity questions and screened the finding as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee did not have complete, accurate and up-to-date design documentation, procedures and work packages. Specifically, MS-100A was not designated as a containment isolation valve in procedures or the USAR. Additionally, the IST basis valve data sheet for the valve was inaccurate in that did it not have the containment isolation valve designator checked at the top of the form, and the WO plant impact review section only discussed the inoperability of the steam supply to the TDAFW and failed to discuss that the containment isolation function of the valve would be inoperable. (H.2(c))

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on August 12, 2009, the licensee failed to have appropriate instructions and procedures for activities affecting quality. Specifically, MS-100A was not designated as a containment isolation valve in procedures, and therefore the appropriate TS were not entered when the valve was taken out-of-service. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR 344856 and CR 350526, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 5000305/2009004-02)

The licensee took immediate corrective actions that included establishing barriers for operations personnel to ensure that MS-100A was identified as a containment isolation valve, establishing a note in the tagging system for the same, and developing a comprehensive list of all containment isolation valves for the next USAR revision. At the end of the inspection period, the licensee continued to perform an apparent cause evaluation to identify the cause and any additional corrective actions needed to address this finding.

1R20 Outage Activities (71111.20)

.1 Refueling Outage (RFO) Activities

a. Inspection Scope

The inspectors reviewed the outage safety plan (OSP) and contingency plans for the RFO that began on September 26, 2009, and continued through the end of the inspection period, 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 cool-down processes and monitored licensee controls over the outage activities listed below:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service;
- 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;
- 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;
- maintenance of secondary containment as required by TS; and
- licensee identification and resolution of problems related to RFO activities.

Documents reviewed during the inspection are listed in the Attachment to this report.

This inspection constituted a partial RFO sample as defined in IP 71111.20-05 because the outage continued into the fourth quarter. The sample will be completed in the fourth quarter.

b. Findings

No findings of significance were identified.

1R21 Component Design Bases Inspection (CDBI) (71111.21)

a. Inspection Scope

Unresolved Item (URI) 05000305/2007006-03 was opened during the Component Design Bases Inspection (CDBI), in February 2007, due to concerns identified with the lack of a power systems analysis for the automatic fast transfer design feature/scheme. Specifically, the CDBI team questioned the licensee's ability to successfully perform a fast transfer in accordance with licensing and design basis requirements. The team was mainly concerned that use of the fast transfer scheme, under certain scenarios, with both 4.16-kiloVolt (kV) Buses connected to the Reserve Auxiliary Transformer (RAT) could have resulted in an out-of-phase transfer and consequent damage or loss of safety-related equipment. During this inspection period, the inspectors reviewed related documents and discussed the licensing and design basis with the Office of Nuclear Reactor Regulation (NRR) staff to verify and confirm Kewaunee's licensing and design basis requirements. Specific documents reviewed are listed in the Attachment to this report.

This review did not represent an inspection sample.

b. Findings

(1) No Analysis for Out-of-Phase Transfer that Could Result in Damage to or Loss of Safety-Related Equipment.

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety-significance (Green) involving the licensee's failure to perform a power system analysis calculation that would have identified that the automatic fast transfer design feature allowed electrical system alignment where both redundant 4.16-kV safety-related Buses 1–5 and 1–6 were supplied by the same offsite source, which could have caused a degraded voltage condition on the safety-related buses, an out-of-phase transfer and consequent damage or loss of safety-related equipment.

Description: The normal operating alignment at Kewaunee consisted of 4.16-kV Bus 1-5 aligned to the Tertiary Auxiliary Transformer (TAT) and 4.16-kV Bus 1–6 aligned to the Reserve Auxiliary Transformer (RAT). Both transformers were fed from offsite power sources. Each of the redundant 4.16-kV Buses could be supplied from either of the two required circuits from the offsite power supply. The electrical distribution system featured a voltage restoration scheme that would automatically transfer a safety-related 4.16-kV Bus to an alternate source of power if an under-voltage condition developed, and voltage was available on an alternate source. The inspectors noted that Technical Specification 3.7.a, and USAR Section 8.2.2.2, stated that either power source can supply both redundant buses. In addition, station procedures did not prohibit alignment of both safety buses to either the RAT or the TAT.

Of particular concern to the inspectors was the configuration where both Buses 1-5 and 1-6 were aligned to the RAT. Upon a loss of the normal offsite power source to 4.16-kV Bus 1-5 via the TAT, a fast transfer to the alternate available qualified source of offsite power (via the RAT) was to occur per design and licensing basis. This alignment could be realized automatically by virtue of an automatic transfer of Bus 1-5 loads to the RAT due to an under-voltage condition or loss of power feed to Bus 1-5. This alignment could also be accomplished manually by manipulation of the 4.16-kV Bus feed breakers. The licensee did not have any calculation that demonstrated that either the RAT or the TAT that feed the safety-related buses was capable of supporting the start of large motors during a design basis event. Additionally, when both engineered safety features (ESF) buses are aligned to the RAT and a safety injection (SI) occurs, the automatic sequencing of both trains of ESF equipment would not meet design basis requirements.

Besides the capability issue of the RAT to support both the safety-related buses, the inspectors' review of the fast transfer scheme also identified various deficiencies in the voltage restoration scheme. The automatic transfer of Bus 1-5 from the TAT to the RAT included the potential for an out-of-phase transfer, which could result in potential damage to the safety-related loads fed from Buses 1-5 and 1-6. The inspectors noted that there was a time delay designed into the bus transfer scheme between the opening of feeder breaker (1-501) from the TAT to Bus 1-5 and the closing of feeder breaker (1-503) from the RAT to Bus 1-5. As a result of this time delay, it was determined that the transfer of Bus 1-5 from the TAT to the RAT could result in out-of-phase transfer and cause damage to the safety-related loads. The inspectors also determined that because the RAT was not able to support the loss-of-coolant accident (LOCA) loads of both Buses 1-5 and 1-6 the voltage on the buses would degrade resulting in actuation of the degraded voltage relays and transfer of the buses to the diesel generators.

The inspectors determined that an evaluation or calculation was not available for review to ensure that an out-of-phase transfer will not occur during a degraded voltage condition on the 4.16-kV safety Buses 1-5 and 1-6, and an SI event, when safeguards loads are slowing down (providing counter-electromagnetic fields) and are being transferred to the EDG following degraded voltage relay (DVR) actuation. This scenario could result in closure of the EDG output breaker prior to complete loss of bus voltage and a potential out-of-phase transfer of the safeguards loads to the EDG. Subsequently, on August 11, 2009, the licensee informed the inspector that a preliminary analysis and operability evaluation performed by the licensee had shown that the expected bus residual voltage at EDG breaker closure time is sufficiently low such that the transfer would be acceptable and the safeguards equipment would be expected to successfully sequence onto the EDG. On August 10, 2009, CR 344370, "Fast Transfer to EDG from

DVR not Evaluated," was issued to document and address this issue. Corrective Action (CA) 143491 was initiated for the engineering design group to complete an analysis in order to formalize the licensee's residual voltage expectation.

The CDBI team also identified that the electrical distribution system was also susceptible to transfers between unsynchronized sources because an analysis was not performed to determine required degraded grid voltage values on the 345-kV offsite power supply. Specifically, when the Bank No. 10 transformer (normally feeds the TAT) was disconnected from the 138-kV switchyard, and the redundant safety buses were aligned to their normal sources, degraded voltage could occur on Bus 1-5 causing an out-of-phase transfer to the RAT (see NCV 05000305/2007006-01).

According to the licensee's operational record they have never had a fast transfer of Bus 1-5 from the TAT to the RAT. However, the licensee had manually aligned the two safety buses to the RAT for short periods of time, seven times in the three years prior to the CDBI. The inspectors determined that the probability of occurrence of a LOCA during the short periods of time that the licensee had aligned both the buses to the RAT was very minimal.

During the CDBI in February 2007, the licensee had committed to implement an electrical calculation reconstitution project. As part of this commitment, Calculation C11721, "Bus Fast Transfer Calculation," Revision 0, dated July 30, 2008, was performed. The calculation developed a transient model of the Alternating Current (AC) Electrical Distribution System and analyzed the bus transfer performance of the system. This model was used to investigate the transient effects during the bus transfers of 4.16-kV safety and nonsafety-related Buses. The calculation evaluated a number of cases and scenarios to observe bus transfer during most limiting operating conditions. The analysis verified the ability to maintain operation of energized equipment during and after each transfer, the impact on system voltage, voltage angle and voltage stress (V/Hz) on the energized equipment and the ability of the 4.16-kV and 480-Volt major motors on the buses to reaccelerate.

Some concerns identified under the operating scenarios considered in the calculation model related to: 1) adverse effects on the Class 1E system when using the fast bus transfer scheme under certain operating scenarios considered; 2) the stresses on the motors were significant as the loads were transferred to an alternate source; 3) a number of safety-related motors fed from safety-related buses failed to meet the specified acceptance criteria set for motor reacceleration; 4) voltages did not recover to meet the set criterion following certain bus transfer scenarios; and 5) voltages on the safety-related buses did not recover above the specified acceptance criteria prior to degraded voltage relay time delay expiration.

Calculation Section 7.2, "Recommendations," stated, in part, that the safety-related Bus 1-5 transfer, due to a TAT failure or a degraded voltage trip following a LOCA, are the only transfers which needed to be further evaluated and investigated in order to prevent under voltage failures following the transfer due to excessive loading and voltage angles. This includes determining the actual bus dead time for the circuit breakers at Bus 1-5. The inspectors reviewed pertinent portions of the calculation including the results and recommendations and concluded that the licensee was planning to evaluate the noted recommendations and implement corrective actions to address the issues identified.

During the CDBI, the inspectors noted that there was no guidance in operations procedures to prohibit operating in this unanalyzed electrical configuration. To address this concern, on February 13, 2007, the licensee initiated a Night Order to: 1) preclude alignment of Bus 1-5 and Bus 1-6 to the same transformer; 2) to ensure that the 345-kV and 138-kV systems are tied together; and 3) to declare offsite power inoperable at grid voltage lower than 140-kV (minimum substation voltage). During the inspection, the licensee initiated interim compensatory measures by revising operating procedure N-EHV-39, "4160-Volt (V) Alternating Current (AC) Supply and Distribution System Operation," dated April 8, 2007, to administratively prohibit aligning both safety-related 4.16-kV Buses to the RAT and to require that control switch 43 for breaker 1-503 (RAT to 4160-V Bus 1-5 feed) be placed in Manual position prior to exceeding 200° Fahrenheit (F) in the RCS; thereby bypassing the auto transfer feature. Corrective Action Program (CAP) 041804 and CR 096058 were initiated to document and address this issue. These interim measures were effective in resolving offsite power availability concerns relating to this finding.

This finding resulted in issuance of a Licensee Event Report (LER) 2007-007-00 on May 21, 2007. The LER described this event as an operating condition that could have prevented the fulfillment of the safety function of SSCs needed for safe shutdown and was not allowed by the Technical Specifications. It also stated that analysis had shown that the receipt of a SI with both Buses 1-5 and 1-6 aligned to the RAT might result in a loss of available offsite power to the ESF buses. In addition, the LER stated that this condition was not in full compliance with GDC 17. Corrective Action CA 029876 of CAP 041804 was linked to this LER.

In December 2008, the licensee performed a follow-up review of Calculation C11450, "AC System Analysis-Load Flow, Voltage Drop and Short Circuit" Revision 1. The licensee determined that simultaneous starting of two CFCUs and an Internal Containment Spray (ICS) pump on the 480-Volt Bus, during an SI sequence and degraded voltage condition, will trip the DVR on high inrush current. This concern applied when the 4kV Buses were energized from the RAT. The licensee's analysis to support ICS operability concluded that spray pump operability required the pump discharge valve to initially be shut and at least one CFCU fan in the same train be running. The licensee initiated compensatory measures to have one CFCU fan running and the spray pump discharge valve initially shut. Condition Reports 320115, 320794 and OD 251 were issued to document the non-conforming condition and required Compensatory measures.

In May 2009, the inspectors informed the licensee that TS 3.7.a, and USAR Section 8.2.2.2, statements that either power source can supply both redundant buses were contrary to the current plant design. Subsequently, the licensee issued CR 335612, UFSAR requires clarifications, dated May 20, 2009, and CR 335822, TS 3.7, Auxiliary Electrical Systems, Clarifications, dated May 22, 2009, to evaluate and address these anomalies.

<u>Analysis</u>: The inspectors determined that the failure to evaluate the effects of aligning the two redundant 4.16-kV safety-related Buses 1-5 and 1-6 to a single transformer being fed from the offsite source was a performance deficiency. The finding was determined to be more than minor because the performance deficiency was associated with the initiating Events Cornerstone attribute of equipment performance in that the performance deficiency adversely affected the cornerstone objective of limiting the

likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, this unanalyzed alignment could have caused a degraded voltage condition on the safety-related buses and caused a plant-centered loss of offsite power event during plant response to a safety injection.

Using IMC 0609, Attachment 0609.04, "Phase 1 Initial Screening and Characterization of Findings," Transient Initiators, the inspector determined that the design deficiency did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment will not be available. Once the plant-centered loss of offsite power occurred, the bus loads would be transferred to the associated EDGs and mitigation equipment would still be functional. Because of this, the finding is screened as Green.

The team determined that the issue was within the licensee's ability to foresee and correct because the licensee had previously identified the lack of motor starting analyses in CAP 024970, issued in January 2005.

The inspectors determined that there was no cross-cutting aspect to this finding because the cause of the performance deficiency was related to a historical design issue and not indicative of current licensee performance.

Enforcement: Title 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, prior to February 16, 2007, the licensee's design control measures failed to evaluate the adequacy of design of the electrical distribution system. Specifically, the inspectors identified that the licensee had failed to perform a power systems analysis calculation that would have identified that the fast transfer design feature/scheme was deficient in that it allowed unanalyzed electrical power system alignments where both redundant 4.16-kV safety-related Buses 1-5 and 1-6 were being supplied by offsite source via the same transformer. Use of this electrical configuration, could have resulted in the loss of available offsite power to the buses and potential for damaging effects on redundant safety-related equipment during a design basis event SI. When identified, the licensee entered this issue into their CAP and implemented interim actions to prohibit use of the fast transfer feature or manually feeding the two safety-related buses from the same transformer during plant operation.

Because this violation was not willful, was of very low safety-significance, and it was entered into the licensee's corrective action program as CAP 041804, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000305/2009003-03).

Based on the above discussion URI 05000305/2007006-03 is also considered closed.

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:

- auxiliary feedwater flow instrument calibration;
- service water header pressure switch calibration;
- auxiliary building special ventilation operability test;
- safety injection pump and valve test (IST); and
- step change in unqualified RCS leakage.

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 USAR, 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 for IST activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (ASME) code, and reference values were consistent with the system design basis;
- 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 sample(s), one inservice testing sample(s), and one reactor coolant system leak detection inspection sample(s) as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

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 August 25, 2009, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities.
The inspectors observed emergency response operations in 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: 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 USAR 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 USAR 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 (DAW), 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 between 2007 and 2009. The shipment documentation reviewed consisted of:

- low specific activity (LSA-II) and surface contaminated object (SCO-II) shipments to waste processors; and
- type A shipment to vendor for repair.

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 in IP 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. Also, lesson plans for safety training and function specific training for radiation protection technicians and for hazardous material level two employees were reviewed for compliance with the hazardous material training requirements of 49 CFR 172.704. Additionally, the hazardous material training test and the test results for selected radiation protection staff were reviewed by the inspectors for adequacy.

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

b. Findings

No findings of significance were identified.

.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 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures PI for the period from the second quarter 2008 through the second quarter 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, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance WOs, 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. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

Potential Unreported Safety System Functional Failures (SSFFs)

Introduction: During a review of recent LERs for SSFFs, the inspectors identified two LERs that may also be reportable as SSFFs. These items remain open pending resolution of the inspectors' questions regarding reportability.

Discussion: While reviewing LER 2008-001, which was reported as an unanalyzed condition, the inspectors questioned why the LER wasn't also reported as a SSFF. The LER describes a condition where during a design basis fire in the relay room, the control cabling for pressurizer PORV PR-2B was found to be vulnerable to spurious operation due to hot shorts as defined in NRC guidance and NRC endorsed NEI guidance for circuit analysis. The inspectors believed that the loss of pressure and inventory control, caused by the spurious opening of PORV PR-2B during a design basis fire, could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition. The inspectors explained their observations to the licensee; however, the licensee believes that the condition did not result in a SSFF. The inspectors also reviewed LER 2009-003, which was reported as a condition prohibited by plant TSs, and questioned why the LER wasn't also reported as a SSFF. This LER describes a condition where the 1A containment spray pump could have potentially tripped during a degraded voltage condition. The licensee believes that a degraded voltage condition is an environmental condition that can be historically reviewed and determined to have not occurred. The inspectors believed the degraded voltage condition is associated with the initiating event. These items remain open pending additional information to determine if there is a performance deficiency. (URI 05000305/2009004-04)

.2 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for the period from the third quarter 2008 through the second quarter 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 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 one reactor coolant system leakage sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for RCS Specific Activity performance indicator for the period from the fourth quarter 2008 through the second quarter 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 RCS chemistry samples, TS requirements, issue reports, event reports and NRC Integrated Inspection Reports for the period of the fourth quarter 2008 through the second quarter 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. None were identified.

This inspection constituted one reactor coolant system specific activity sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.4 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the period from the fourth quarter 2008 through the second quarter 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 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 radiation protection 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.

.5 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 the fourth quarter 2008 through the second quarter 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 the fourth quarter 2008 and the second quarter 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.

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; timeliness was commensurate with the safety significance; 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 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 listed in the Attachment to this report.

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 inspection samples.

b. Findings

No findings of significance were identified.

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

.1 Response To Unplanned Or Non-Routine Events

a. Inspection Scope

The inspectors reviewed the plant's response to the following non-routine event:

- increasing seal water leakage on from the reactor coolant pump "A" seal #1.

The inspectors reviewed control room logs and licensee procedures, and interviewed licensee personnel following this event. This event follow-up review constituted one sample as defined in IP 71153-05. Documents reviewed as part of this inspection are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 (Discussed) LER 05000305/2009-003-00: Containment Spray Pump "A" Inoperable At Degraded Voltage Protection Setpoint

This LER describes an engineering evaluation which determined that the degraded voltage protection relay for containment spray pump 1A could trip under conditions where the pump was required to be operable. Full review of the LER will be completed after resolution of URI 05000305/2009004-004 documented in section 4OA1 to complete the inspection. Documents reviewed as part of this inspection are listed in the Attachment to this report.

.3 (Closed) Licensee Event Report 0500305/2009-005-00: Nonfunctional Steam Exclusion Door Results in Postulated Inoperability of Safety Systems

This event report, which was submitted after a design reconstitution review determined that steam exclusion door 140 was not capable of withstanding the pressures generated by a postulated high energy line break, was withdrawn. The licensee conducted pressure testing of equivalent doors and the evaluation of the tests determined that door 140 was capable of fulfilling its steam exclusion function. The inspectors observed the actual tests and did not identify any discrepancies. The inspectors also verified that no performance deficiencies or violations of NRC requirements had occurred. Documents reviewed as part of this inspection are listed in the Attachment to this report. This LER is closed.

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

4OA5 Other Activities

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

a. Inspection Scope

(1) Heavy Loads

The inspectors reviewed the licensee's crane and heavy loads program with regards to ISFSI operations. Inspectors completed both a document review of the Auxiliary Building Fuel Handling (ABFH) crane as well as an operational walk-down of the crane. The inspectors reviewed topics associated with the reactor 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 reactor building crane.

The original ABFH crane was manufactured by the Whiting Corporation under the Electric Overhead Crane Institute - 1961 (EOCI-61) standards as a 125-ton, non-single failure proof crane. In order to meet single failure-proof criteria for spent fuel cask handling, the licensee installed an American Crane and Equipment Corporation (ACECO) single failure-proof trolley on top of the original Whiting bridge. The single failure-proof criteria were met using NUREG-0554 with alternatives from NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, Appendix C. The NUREG-0612, Appendix C matrix was submitted to the NRC and the crane was approved for use under License Amendment 200. Amendment 200 committed the licensee to American Society of Mechanical Engineers (ASME) Code B30.2-1976. The crane 125-ton load rating was not changed; however, stiffeners were added to the bridge for seismic considerations.

The AFBH crane was inspected and maintained in accordance with ASME Standard B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," and the crane manufacturer's instructions. Operational testing prior to placing the crane in service was performed under ACECO Procedure REP 20776-010; Monthly crane, wire rope, and hook inspections were performed under Mechanical Maintenance Procedure MA-KW-MPM-CRN-007; Annual crane and wire inspections were performed under Mechanical Maintenance Procedure MA-KW-MPM-CRN-002 and Electrical Maintenance Procedure MA-KW-EPM-CRN-001. Non-Destructive testing of the bridge critical welds is performed every four years in accordance with American Welding Society (AWS) 01.1-2004 Structural Welding Code – Steel and Preventive Maintenance Task RE89105. The inspectors discovered that some of monthly and annual crane inspection procedures did not include some of the inspection criteria specified in ASME B30.2, specifically on the sections pertaining to the hoist wire rope and hook attachment and securing means. The licensee documented this issue in CR 337889 and revised the procedures prior to initial fuel loading activities.

The critical load drop protection features of the ABFH crane were intact and operable. The features included the drum drop structure, dual wire rope-reeving systems, dual hoist holding brakes, load hangup protection system, and two-block protection system. Provisions for manual operation of the crane were also tested and operable.

The ABFH crane was load tested in accordance with ASME B30.2 and the crane manufacturer's instructions. The crane was loaded to 313,100 pounds (125 percent). The two weight limiting switches activated by a wire rope strain sensor were tested independently. The ambient temperature at the crane girders was 82.1° - 88.4° F during the load test and the licensee has established the minimum crane operating temperature at 87° F. Non-Destructive Examinations (NDE) (Magnetic Particle) of the bridge girder critical welds were performed following the static load testing. No rejectable indications were identified.

The crane was also loaded to 250,000 pounds (100 percent) and dynamic load testing was performed prior to placing the crane in service. The 150-ton duplex hook was load tested by the vendor to 300 tons (600,000 pounds). No permanent deformation in throat opening was observed and the ultrasonic testing and magnetic particle testing showed no rejectable indications. Two-block testing was performed, during the test, the hoist overweight switches and upper travel limit switches were bypassed. The main hoist slip clutch slipped on two-blocking, as designed. Load hangup testing was performed, the hoist motor stopped when the overload setpoint was reached.

Inspectors observed that the ABFH crane was operated in accordance with ASME B30.2 and the crane manufacturer's instructions. At the beginning of each shift, the crane operator performed a functional check of the hoist primary upper limit switch. During pre-operational testing of the crane, inspectors identified that the licensee did not indicate a minimum lift height in the ISFSI heavy loads procedures which was a manufacturer's specifications requirement that provided adequate distance for the brakes to engage during an uncontrolled descent, before the load would impact the floor. The licensee documented this issue in CR 337849 and established a minimum lift height of four inches above the floor prior to loading fuel. The safe load path was defined in the licensee's procedure and was direct from the SFP, to the decontamination room, and to the transfer trailer. Crane movement was limited by bridge and trolley travel limit switches.

Inspectors observed that the wire rope slings used for setting the shield plug met the requirements of ASME standard B30.9, "Slings," and that they were properly rated for the lift. Inspectors reviewed procedures for frequent and periodic sling inspections. Inspectors verified that slings shall be removed from service if any conditions listed in ASME B30.9, Section 9-2.9.4 were found to be present. Inspectors also observed procedures that governed general rigging practice for consistency with ASME B30.9, Section 9-6.10.4.

The inspectors verified that the licensee had developed procedures to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. The operators had been trained and qualified to operate the crane. The inspectors identified that the licensee had not adequately documented the training which the inspectors verified the operators received. The licensee documented this issue in CR 338428. The program consisted of classroom training and On-the-Job Training (OJT). All certified crane operators had received this training prior to installation of the new ACECO trolley. Changes to crane operation and maintenance, caused by installation of the new ACECO trolley, were presented by ACECO using Lesson Plan NU-KPS-MMC-08-LP006. This lesson plan enveloped ACECO Procedure REP-20776-016, "Operator Training for Auxiliary Building Crane" and was followed by a written examination. The lesson plan also included a functional operating session in the field. Eleven crane operators completed this course and were certified on January 31, 2008, and March 13, 2008.

(2) Dry Run Demonstrations

Inspectors observed the licensee's dry run demonstration in preparations to load fuel at the Kewaunee Power Station (KPS) the week of August 3, 2009. The inspectors reviewed the loading and unloading procedures to ensure that they contained commitments and requirements specified in the license, TSs, the FSAR, and the Title 10 CFR Part 72. The inspectors observed the licensee's pre-job briefs and focused pre-briefs prior to the applicable evolutions. The licensee conducted these meetings in a professional manner where the necessary pertinent tasks and safety items were discussed. Radiation protection staff attended pre-job briefs and gave insight into working conditions and As-Low-As-Reasonably-Achievable (ALARA) practices. The staff was interactive and questions were addressed, as well as suggestions considered by supervisors to gain additional insight.

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 loading and unloading of "dummy" fuel bundles into the storage canister basket. The licensee demonstrated removal of dummy fuel assemblies from the SFP storage racks, placed them into different locations in the canister, and returned them from the canister to the SFP racks. The NRC inspectors noticed a lot of personnel on the refuel floor and on the refuel bridge during dry run demonstrations and questioned the licensee's ALARA practices. Licensee management reviewed the worker's roles during the evolutions and ensured that during the actual loading the personnel on the refuel floor and refuel bridge would be limited to the ones that were critical for the evolution.

The inspectors observed crane operation to ensure that heavy loads could be safely lifted and transferred. Due to unresolved NRC concerns related to the single-failure-proof design of the transfer cask-lifting beam at the 105-ton rated lift load, the licensee used the transfer cask-lifting beam at a derated crane lift load. During the dry run, the licensee demonstrated installation of the shield plug with the derated crane lift load controlled through the heavy loads procedure. Down ending of the transfer cask containing a storage canister filled with dummy assemblies from the refueling floor to the transfer trailer was observed as well as lifts from the transfer trailer to the refueling floor. The inspectors also observed the lift of the transfer cask out of the SFP and into to the cask decontamination pit. The inspectors verified that lifts were performed in accordance with appropriate industry standards and followed the designated safe haul path.

The inspectors observed the installation of the "spyder" which was a substitute for the inner and outer lid of the dry shielded canister (DSC), installation of the transfer cask lid, as well as removal of the lid at the Horizontal Storage Module (HSM). The inspectors observed the successful transfer of the storage canister to the ISFSI. During the dry run demonstration the inspectors observed successful insertion and retraction from an HSM. Proper controls were in place during the transfer of the canister from the reactor building to the HSM on the ISFSI. These controls included health physics coverage, adherence to the heavy haul path, and appropriate security oversight. The inspectors verified 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. The inspectors reviewed a completed fuel selection package for the two casks to be loaded during the campaign to verify that the licensee used the criteria specified in the TSs to verify the acceptability of assemblies to be loaded in a cask. The inspectors observed the licensee's methods to independently verify and document fuel assemblies. The licensee did not plan to load any damaged fuel assemblies during this campaign.

(4) Radiation Protection

The inspectors evaluated the licensee's radiation protection 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 ensure dose rate limits and surveillance requirements of the TSs were met. 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.

(5) Training

The inspectors reviewed the licensee's training program which consisted of classroom and OJT to ensure involved staff was adequately trained for the job they were responsible to perform. The licensee had a dry fuel storage qualification matrix which documented each workers completed training courses and qualifications. The inspectors interviewed select individuals who were responsible for performance of specific tasks during loading to evaluate their knowledge regarding the campaign activities, the cask loading process, and use of the equipment.

The inspectors reviewed training records of welders and other personnel who the licensee authorized to perform NDE inspections to ensure that these individuals had current training and qualifications.

(6) Emergency Preparedness and Fire Plan

The inspectors reviewed the licensee's emergency preparedness plan required by 10 CFR Part 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.

(7) Structural Modifications and Associated Design Documentation

The inspectors reviewed the licensee's documentation associated with the control of heavy loads. Specifically, the inspectors reviewed design documentation that demonstrated a safe load path for the licensee's designated haul path that moved the dry cask storage system components from the SFP through the auxiliary building.

The inspectors reviewed design calculations, modifications, and other related documentation that demonstrated the crane special lifting device and existing structures affected by movement of the heavy load for the designated haul path remained within the design and licensing bases.

b. Findings

(1) Inadequate Design Analysis for 105-Ton Transfer Cask-Lifting Beam

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the licensee’s failure to provide an adequate single failure proof design basis analysis for the 105-ton transfer cask-lifting beam. The licensee entered this issue into its corrective action program and revised the design calculation for the 105-ton transfer cask-lifting beam to demonstrate compliance with single failure proof acceptance criteria.

Description: On May 21, 2009, the licensee completed their owner acceptance review of Millstone calculation ISFSI-040336CG, “Design of 105-Ton Transfer Cask-Lifting Beam,” Revision 0, which accepted ACECO calculation CAL-20602-SE-700, Revision 2. The licensee entered the accepted document into the Kewaunee Power Station (KPS) document system as calculation ISFSI-040336CG, Revision 0. The KPS personnel utilized the Millstone transfer cask-lifting beam since the KPS utilized the same transfer cask as Millstone.

The inspectors identified a non-conservative application of a standard engineering formula that was used to calculate bending stress in the evaluation of the lift pins in calculation CAL-20602-SE-700. Specifically, while calculation CAL-20602-SE-700 indicated that the crane hook would distribute the transmitted lift load at the lift pin across a one inch length, the evaluation that calculated bending stress in the lift pins distributed the transmitted load across a seven-inch length. The inspectors noted that distributing the applied load across a seven-inch length was non-conservative to distributing the applied load across a one inch length. The inspectors further noted that CAL-20602-SE-700 did not include the effect of the four-ton weight of the transfer cask-lifting beam. Although calculation CAL-20602-SE-700 had sufficient design margin to accommodate an approximate four percent load increase, this non-conservatism needed to be considered in conjunction with the resolution of the load application concern.

In Addendum B of Revision 2 of Dominion calculation ISFSI-040336CG, “Evaluation of Bending Stresses in 6” Diameter Pins for the 105-ton Cask-Lifting Beam,” the licensee performed a detailed finite element analysis of the lifting pins that accurately modeled the applied load across a one inch length and also included the effect of the four-ton weight of the transfer cask-lifting beam. This calculation addendum demonstrated the lifting pin component of the cask-lifting beam met the stress limits for single failure proof special lifting devices. However, in order to demonstrate that the calculated bending stress in the lift pin met the acceptance limit for single failure proof acceptance limits for single failure proof lifting devices, the licensee performed an additional analysis to demonstrate the dynamic loading factor specified in their purchase order for the 105-ton cask-lifting beam could be relaxed based on the very low crane lift speed for movement of the transfer cask. The inspectors further noted that the detailed finite element analysis confirmed that the methodology used in CAL-20602-SE-700, distributing the applied load across a seven-inch length, was non-conservative; i.e., the finite element analysis of the lift pin resulted in a larger bending stress magnitude.

Analysis: The inspectors determined that the licensee's failure to conservatively calculate bending stress in the lift pins in accordance with the standard engineering formula was contrary to American National Standards Institute (ANSI) N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kilograms) or More," and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of equipment performance and adversely 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 licensee used compliance with the ASNI N14.6 design requirements in conjunction with Section 5.1.6, "Single Failure Proof Handling Systems," of NUREG-0612 as a means to demonstrate safe load handling of heavy loads by providing a single failure proof special lifting device. An appropriate application of the standard engineering formula would have resulted in a calculated bending stress in the lift pin in excess of the single failure proof stress criteria for special lifting devices.

The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with IMC 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 105-ton transfer cask-lifting beam had not been previously used at the KPS.

This finding has a cross-cutting aspect in the area of human performance, work practices, because the licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. Specifically, the licensee failed to perform an effective owner's review to assure that appropriate design methods are used in calculations that demonstrate nuclear safety. (H.4(c))

Enforcement: Title 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 May 21, 2009, the licensee failed to provide for the design adequacy of the 105-ton cask-lifting beam consistent with ANSI N14.6 and NUREG-0612 single failure proof design requirements. Specifically, licensee calculation ISFSI-040336CG used a non-conservative application of a standard engineering formula to calculate bending stress in the lift pins for the 105-ton transfer cask-lifting beam. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as CR 339267, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000305/2009004-05)

As a corrective action, the licensee revised the design calculation for the 105-ton transfer cask-lifting beam and demonstrated compliance with single failure proof acceptance criteria.

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

a. Inspection Scope

The licensee holds a general license for operation of its ISFSI. The licensee documented the results of the required evaluations in Technical Report NE-1567, "Kewaunee Power Station Independent Spent Fuel Storage Installation 10 CFR 72.212 Evaluation Report." The inspectors reviewed the technical report and a sample of referenced supporting documents to determine its acceptability and compliance with conditions set forth in the Certificate of Compliance (CoC), the FSAR, and 10 CFR Part 72 requirements in regards to the NUHOMS 32PT cask system.

Based on the review of the evaluation, the inspectors assessed that overall, the evaluation report was comprehensive and adequately addressed the areas required to be evaluated under 72.212(b)(2) through (4). No significant oversights or concerns were identified. The inspectors noted that with regard to the review of reactor site parameters for ISFSI operation, the impact of seiches had not been evaluated even though the ISFSI is located in close proximity to Lake Michigan; Kewaunee personnel subsequently added a section to the final evaluation report addressing this issue.

The inspectors reviewed several supporting documents referenced in the evaluation report, in particular the ISFSI Fire Hazards Analysis report SL-009169, Revision 1, dated February 10, 2009. This report contained the results of the fire and explosion hazard analysis for the ISFSI haul path and storage location and prescribes required standoff distances for various hazards` as well as any other physical or administrative controls required for ISFSI operations. The inspectors reviewed the above documents and determined that the supporting engineering evaluations for the fire/explosion hazards analyses were detailed and used a systematic approach to evaluate all potential fixed and transient fire/explosion hazards. The inspectors interviewed staff regarding control of combustibles on and near the ISFSI pad.

Conservative and appropriate assumptions involving administrative controls were placed in the appropriate ISFSI operating procedures. The inspectors noted minor discrepancies with regard to the following issues:

- ensuring that the guidance document for walkdowns of the haul path prior to transfer cask movement had sufficient detail to provide for the identification of any new fire or explosion hazards introduced into the area since the performance of the original fire hazard evaluation.
- the use of acetylene tanks in a weld shop near the haul path had not been identified or considered in the fire hazard evaluation report; and
- the use of propane-powered forklifts or other similarly powered equipment within the ISFSI boundaries had not been considered or their use restricted.

These issues were discussed with Kewaunee personnel who addressed the observations prior to initial loading.

The licensee provided a basis for not performing a welding demonstration on site due to previous demonstrations and performances. The licensee took credit for the welding demonstration at Millstone due to the following similarities: (1) the cask systems/components used at Millstone and Kewaunee are identical; (2) Kewaunee used the same contractor, PCI Energy Services (PCI), as Millstone in a weld demonstration that was previously observed by NRC inspectors; (3) PCI used welding procedures identical to the ones used at Millstone to weld the casks at Kewaunee; (4) Kewaunee staff observed the welding at Millstone in April; (5) Kewaunee staff and PCI performed walkdowns/logistics review for accomplishing the welding at Kewaunee and performed the necessary equipment checks; and (6) Millstone personnel, including mechanics and radiation protection personnel, assisted at Kewaunee during the dry run and loading campaign.

Kewaunee addressed their basis for satisfying the CoC requirement for dry run demonstrations without performing an actual demonstration onsite in the 72.212 evaluation.

b. Findings

No findings of significance were identified.

.3 Operation of an ISFSI at Operating Plants (60855.1)

a. Inspection Scope

The inspectors observed and evaluated the licensee's loading of the first canister during the campaign to verify compliance with the CoC, TSs, and associated procedures. The inspectors observed loading of select fuel assemblies, lifting of the transfer cask from the SFP, decontamination and surveying, welding of the lids, non-destructive weld examinations, vacuum drying, downending and placement of transfer cask onto transfer trailer, and transfer of the cask to the ISFSI pad.

During performance of the activities, the inspectors verified the licensee staff's familiarity with procedures and its steps, adequate supervisory oversight, and communication and coordination between the groups. The inspectors reviewed the loading procedures and evaluated the licensee's adherence to the procedures. The inspectors also verified that the contamination and radiation levels from the transfer cask were well below the regulatory and TS limits. The inspectors attended various pre-job briefs to assess the licensee's ability to identify critical steps of the evolution, potential failure scenarios, and tools to prevent errors.

The inspectors reviewed a number of condition reports and the associated follow-up actions that were generated in response to some unexpected conditions encountered during the loading campaign. The projected dose for the loading and transfer of the first cask was 500 millirem (mrem) whereas the resulting accumulated dose for the cask was 608 mrem. The additional dose contributor was the manual welding that was performed on the keyway of the inner lid due to an issue with the welding machine and the silver dollars on the inner lid due to the weld head equipment that was coming in contact with a camera used for remote welding (CR 345475 and CR 345343).

The inspectors reviewed the licensee's fuel selection process to verify that the process incorporated all of the physical, thermal, and radiological fuel acceptance parameters specified in the current CoC and TSs. The inspectors reviewed the fuel selection procedures and the qualification records and ensured damaged fuel was not loaded. The inspectors reviewed the licensee's surveillance and maintenance program associated with storage of fuel. Each fuel assembly was loaded under an approved loading plan. The identity of each spent fuel assembly was confirmed prior to insertion into the canister, and was verified to be in the correct location following loading. The total canister decay heat load was 15.639 kilowatts (kW), which was below the 24.0 kW limit. The transfer cask dose rates were within the TS limits of 200 and 500 mrem/hour respectively. During welding operations the licensee was required by procedures to monitor for hydrogen generation to prevent an ignition event. The space below the shield plug was monitored for hydrogen concentration before and during welding. The loading procedure required an inert gas purge of the space below the shield plug if the hydrogen concentration reached 2.4 percent, which was not approached during welding of the inner lid. The TS vacuum drying time limit of 31 hours was met. The TS helium backfill pressure of 2.5 to 3.5 pounds-force per square inch gauge was met.

During transfer of the first cask to the ISFSI pad, the licensee implemented appropriate security measures. The licensee commenced daily and startup thermal monitoring of the loaded Horizontal Storage Module, in accordance with TS requirements. The temperatures observed were consistent with the design calculations.

b. Findings

(1) Failure to Follow ISFSI Loading Procedure Step

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 72.150, "Instructions, Procedures, and Drawings," during the ISFSI loading campaign. The licensee failed to follow a step of Procedure OP-KW-NOP-ISF-001, "Dry Shielded Canister Loading." The licensee entered this issue into its corrective action program and discussed the need to check the crane brakes when lifting loads approaching the rated loads with the refueling crew to prevent missing this step in the future.

Description: During the Kewaunee ISFSI loading campaign, the inspectors observed that the licensee crane operator did not check the hoist brakes when the load approached the rated load for the crane. Checking the brakes meant lifting the load a short distance and applying the brakes and verifying that any slippage is minimal. Specifically, this check was required by step 3.2.11 of Procedure "Overhead Cranes/Hoists," which was referenced in step 5.2.6 of Procedure OP-KW-NOP-ISF-001, "Dry Shielded Canister Loading."

Analysis: The inspectors determined that the licensee failed to follow step 5.2.6 of Procedure OP-KW-NOP-ISF-001, "Dry Shielded Canister Loading." The issue was addressed by traditional enforcement since 10 CFR Part 72 is not risk based and is not covered under the Reactor Oversight Process.

The inspectors reviewed the examples in the Enforcement Policy, Supplement I and determined that the failure to follow the procedure was a violation that had more than minor safety or environmental significance but did not rise to a Severity Level I, II, or III violation. The inspectors determined that the violation had more than minor safety significance because failure to check the brakes results in not knowing if the brakes are functioning properly, which may lead to a failure of the brakes while lifting a loaded spent fuel canister.

The inspectors determined that there was no cross-cutting aspect associated with this finding.

Enforcement: Title 10 CFR Part 72.150, "Instructions, Procedures, and Drawings," requires, in part, that the licensee, applicant for a license, certificate holder, and applicant for a CoC shall prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed.

Step 5.2.6 of Procedure OP-KW-NOP-ISF-001, "Dry Shielded Canister Loading," requires, in part, that the use of cranes, hoists, rigging, and lifting of loads shall be in accordance with procedure MA-AA-OCR-101, "Overhead Cranes/Hoists."

Procedure MM-AA-OCR-101 requires, in part, if a load approaching the rated load is to be handled, then the following checks of the hoist brakes are required at least once each shift: a) raise the load a short distance; b) verify the brakes engage automatically when the hoist is stopped; and c) verify that any slippage is minimal.

Contrary to the above, during the weeks of August 17 and August 24, 2009, the licensee failed to follow an instruction, procedure, and drawing, as specified in 10 CFR 72.150 for an activity affecting quality. Specifically, during the period, the licensee lifted dry fuel storage casks with loads approaching the rated load using the ABFH crane and the licensee failed to: a) raise the load a short distance; b) verify the brakes engage automatically when the hoist is stopped; and c) verify that any slippage is minimal when a load approaching the rated load was lifted.

The licensee immediately evaluated the situation and discussed the need to check the crane brakes when lifting loads approaching the rated loads with the refueling crew to prevent missing this step in the future. Because this violation was of very low safety significance, was non-repetitive and non-willful, and was entered into the corrective action program (CR 347138), this violation is being treated as a Severity Level IV NCV of 10 CFR 72.150 consistent with what Section VI.A.1 of the Enforcement Policy. (NCV 05000305/2009004-06)

.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 (Discussed) Unresolved Item URI 0500305/2009003-01: Emergency Diesel Generator Fuel Oil and Day Tank Vent Line Design Qualification

The inspectors identified additional issues related to this unresolved item, specifically regarding the seismic qualifications of the EDG fuel oil storage vent and fuel oil day tank vent lines. While reviewing this unresolved item associated with the tornado missile qualifications of the diesel generator fuel storage and day tank vents the inspectors questioned the licensee about the seismic qualifications of the vent lines. The licensee initiated CR347730 and revised the tornado operability determination (OD-169) to address the seismic qualification aspects of the vent lines. The inspectors reviewed the operability determination and found that the basis for the immediate operability appeared satisfactory. However, the evaluation did not assess the operability requirements against the design requirements and current licensing basis, but rather used an engineering evaluation to support operability until the licensing basis could be evaluated. Because the licensee was reconstituting the licensing basis for the seismic requirements concurrent with the tornado qualification issues identified in URI 0500305/2009003-01, the inspectors considered resolution of the seismic requirements for the EDG fuel oil storage and day tank vents unresolved pending reconstitution of the licensing basis by the licensee. Therefore, additional information was required regarding this unresolved item to determine if there is a performance deficiency.

.6 (Closed) Unresolved Item URI 05000305/2007006-03: No analysis for out-of-phase transfer that could result in potential damage or loss of safety-related equipment.

This issue is described in Section 1R21 and was resolved to a NCV of 10 CFR Part 50, Criterion III, Design Control. (NCV 05000305/2009004-02)

.7 (Closed) Licensee Event Report 50-305/2007-007-00: Unexpected Safety Injection With Safeguards Buses Connected to The Reserve Auxiliary Transformer.

This issue is described in Section 1R21 and was resolved to a NCV of 10 CFR Part 50, Criterion III, Design Control. (NCV 05000305/2009004-02)

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 7, 2009, the inspectors presented the inspection results to Mr. Steve Scace, 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.

.2 Interim Exit Meetings

Interim exits were conducted for:

- radioactive material processing, transportation, and performance indicator verification with Mr. S. Scace on July 17, 2009.
- results of the inspection review of licensee's actions pertaining to Unresolved Item 05000305/2007006-03 and Licensee Event Report 50-305/2007-007-00 with Site Licensing Manager, Mr. T. Breene, and other members of the licensee's staff via telephone on August 17, 2009.
- licensee identified violation for the completion of a liquid radioactive release permit by an unqualified individual with Mr. M. Hale on September 9, 2009.
- an exit teleconference for ISFSI inspection procedures 60854.1 and 60855.1 was held on September 23, 2009. The inspectors presented the inspection results to members of the licensee management and staff. Licensee personnel acknowledged the information presented. The inspectors asked licensee personnel whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

Interim Debriefs

Interim debriefs regarding the ISFSI inspections were conducted on February 13, February 26, June 11, July 9, August 7, and August 21, 2009. The inspectors presented the inspection results to members of the licensee management and staff. Licensee personnel acknowledged the information presented. The inspectors asked licensee personnel whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Severity Level IV) was identified by the licensee. The violation met the criteria of Section VI of the NRC Enforcement Policy, for dispositioning as an NCV.

.1 Radioactive Liquid Waste Discharge Permit Completed by Unqualified Technician

Technical Specification 6.4 states: "A training and retraining program for the Plant Staff shall be maintained and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI-18.1-1971 and 10 CFR Part 55." General Nuclear Procedure GNP-13.03.01 Revision 23 "Conduct of Training" implements these requirements. This procedure identifies that trainees are responsible for ensuring personal qualifications are current to independently perform assigned work.

Contrary to the above, on July 16, 2008, a lead chemistry technician independently performed an activity without having met the requisite qualification. Specifically, the individual completed a radiological liquid waste release discharge permit after his qualification for performing the activity was revoked during the periodic re-qualification program on July 16, 2008. Based on an Office of Investigations investigation; Case No. 3-2008-027, the NRC staff concluded that a lead chemistry technician deliberately violated the training requirements by completing a radiological liquid waste release discharge permit knowing that his qualifications were not current.

The violation was categorized as a Severity Level IV violation because: 1) the violation had limited actual radiological significance and low potential significance; 2) the violation involved the acts of a low-level individual; 3) the violation resulted from an isolated action without management involvement; 4) there was no economic or other advantage gained as a result of the violation; and 5) adequate remedial action was taken by the licensee. Because the violation is of very low safety significance, it meets the additional criteria in Section VI.A.1 of the NRC Enforcement Policy, and because it has been entered into the corrective action system (CR103866) it is being treated, after consultation with the Director, Office of Enforcement, as an NCV.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Scace, Site Vice-President
M. Crist, Plant Manager
M. Wilson, Director, Safety and Licensing
S. Yuen, Director, Engineering
D. Allison, Radiation Protection Supervisor
M. Aulik, Manager, Design Engineering
T. Breene, Manager of Licensing
J. Stafford, Organizational Effectiveness Manager
C. Chovan, Outage and Planning Manager
W. Henry, Maintenance Manager
J. Madden, System and Component Engineering
T. Brookmire, Supervisor, Dominion Nuclear Spent Fuel Engineering
D. Fried, Operations Fuel Handling Supervisor
J. Gadzala, Licensing Engineer
M. Hale, Radiation Protection Manager
D. Lohman, Project Engineer
R. Repshas, Licensing Engineer
D. Shannon, Radiation Protection General Supervisor
B. Wakeman, Dominion Nuclear Spent Fuel Engineering
D. Allison, Radiation Protection Supervisor

Nuclear Regulatory Commission

M. Kunowski, Chief, Division of Reactor Projects, Branch 5
R. Daley, Chief, EB3
M. Munir, Reactor Engineer, EB3

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000305/2009004-01	URI	Technical Specification Action Requirements During a Leak in a Containment Fan Coil Unit Service Water Line (Section 1R13)
05000305/2009004-02	NCV	Containment Isolation Valve Inoperable With No Technical Specification Action Requirement Entry (Section 1R19)
05000305/2009004-03	NCV	Failure to adequately analyze the automatic fast transfer feature that allowed operation with both 4 kV safety-related Buses 1–5 and 1–6 connected to the RAT (Section 1R21.b)
05000305/2009004-04	URI	Potential Unreported Safety System Functional Failures (Section 4OA1)
05000305/2009004-05	NCV	Inadequate Design Analysis for 105-ton Transfer Cask-Lifting Beam (Section 4OA5.1)
05000305/2009004-06	NCV	Failure to Follow ISFSI Loading Procedure Step (Section 4OA5.3)

Closed

05000305/2009004-02	NCV	Containment Isolation Valve Inoperable With No Technical Specification Action Requirement Entry (Section 1R19)
05000305/2009004-03	NCV	Failure to adequately analyze the automatic fast transfer feature that allowed operation with both 4 kV safety-related Buses 1–5 and 1–6 connected to the RAT (Section 1R21.b)
05000305/2009004-05	NCV	Inadequate Design Analysis for 105-ton Transfer Cask-Lifting Beam (Section 4OA5.1)
05000305/2009004-06	NCV	Failure to Follow ISFSI Loading Procedure Step (Section 4OA5.3)
05000305/2009-005-00	LER	Nonfunctional Steam Exclusion Door Results in Postulated Inoperability of Safety Systems (Section 4OA3)
05000305/2007006-03	URI	No analysis for out-of-phase transfer that could result in potential damage or loss of safety-related equipment (Section 4OA5.6)
50-305/2007-007-00	LER	Unexpected Safety Injection With Safeguards Buses Connected to The Reserve Auxiliary Transformer (Section 4OA5.7)

Discussed

05000305/2009-003-00	LER	Containment Spray Pump A Inoperable At Degraded Voltage Protection Setpoint (Section 4OA3)
0500305/2009003-01	URI	Emergency Diesel Generator Fuel Oil and Day Tank Vent Line Tornado Design Qualification (Section 4OA5)

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.

<u>1R04 Equipment Alignment</u>

-CR 337130; ICS Pump "B" Oil Level Low In Sightglass
-CR 339489; Spent Fuel Pool Piping Is Dripping Condensation That Is Causing A Safety Hazard
-CR 340325; SFP Pump "A" Noise Is Elevated And Oil Is Dark
-CR 345292; Open Scaffold Clamp in Contact with Component Cooling line in North Penetration Room
-CR 345674; Resident Questions Designation of "No Safety-Related Equipment" on Scaffold Checklist
-N-ICS-23-CL; Containment Spray System Prestartup Checklist; Revision 31
-N-SI-33-CL; Safety Injection System Prestartup Checklist; Revision AK
-N-SFP-21-CL; Spent Fuel Pool Cooling And Cleanup System Prestartup CL; Revision T
-Drawing OPERM-218; Flow Diagram; Spent Fuel Pool Cooling And Clean-up System; Revision AD
-Drawing M-217; Flow Diagram; Internal Containment Spray System; Revision AP
-Drawing OPERXK-100-28; Flow Diagram; Safety Injection System; Revision AR
-Drawing OPERXK-100-29; Flow Diagram; Safety Injection System; Revision AE
-Drawing OPERXK-100-132; Flow Diagram; Waste Disposal System; Revision AG

<u>1R05 Fire Protection</u>

-CA 129939; Evaluate OE 28238 for Applicability To KPS And Take Appropriate Actions
-CA 131611; Evaluate FPP-08-02 And 08-10 For Adequate Guidance
-CR 117633; Fire Brigade Drill Areas Noted For Improvement
-CR 340616; Fire Drill Scenario Acceptance Criteria Shortfalls
-CR 347770; NRC Questions Staging Of Cable Rolls In Cardox Room
-FBD 007; Fire Brigade Drill Scenario; Revision 4
-FPP-08-07; Control Of Ignition Sources; Revision 11
-FPP-08-08; Control Of Transient Combustible Materials; Revision 9
-FPP-08-10; Fire Drills; Revision 5
-OP-KW-AOP-FP-001; Abnormal Operating Procedure – Fire; Revision 1
-Drawing A-538; CO2 Storage Tank Room; Revision D
-Drawing A-549; Waste Handling Area; Revision F
-Drawing A-550; Condensate Storage And Reactor Make Up water Storage Room And Adjacent Areas; Revision D
-Drawing A-552; Relay Room And Loft; Revision C
-Fire Drill Pre-Plan 2009-16; Warehouse – Flammable Materials Storage Room; July 7, 2009

1R06 Flood Protection Measures

-CR 349375; Potential Flood Source In Control Room Air Conditioning Room
-OP-KW-AOP-GEN-005; Barrier Control; Revision 2
-TDBD-KPS-FLD; Internal Flooding; Revision 2
-Calculation 2005-05708; Internal Flood Levels Due To Postulated Piping
-Drawing A-586-7; Door Notes; Revision J
-Drawing M-905; Service Water supply control room A/C Chiller 1A; Revision H
-Drawing M-910; Control Room A/C Chiller Pump 1A Suction; Revision A
-Drawing M-1641; 4” Service Water Supply To 1A Control Room A/C Chiller; Revision D
-Drawing OPERM-605-1; Flow Diagram Heating System; Revision K
-eSOMS Station Narrative Log; September 30, 2009
-Internal Flooding – Auxiliary Building; Desktop Guidelines – Walkdown Checklist; Flood Zone 301; Control Room AC Equipment Room; April 26, 2005
-Ruptures In General Pipe Lines In The Auxiliary Building; Revisions 1 and 3
-Kewaunee Nuclear Power Plant Auxiliary Building Internal Flood Evaluation; Revision 0

1R11 Licensed Operator Requalification Program

-LRC-09-SEE03A; Simulator Exercise Guide; Revision A

1R12 Maintenance Effectiveness

-CA 125178; Justification As To Why The Box Fan Is Not Required To Support Functionality
-CA 125179; Perform OP-KW-100-1000 For R-20 Shut Down When R-16 Was Scheduled
-CAP 011254; R-15 Detector Failed Low
-CAP 011372; R-15 Air Ejector Exhaust Failed
-CAP 013860; R-15 Air Ejector Radiation Monitor Failure
-CAP 024129; R-15 Detector Failure
-CAP 034390; R-15 Rad Monitor Failure Causes Auto Actuation Signal
-CR 317524; NRC Resident Questions Regarding Temporary Fan
-CR 317575; R-20 Shut Down When R-16 Was Scheduled
-CR 330028; Perform A Maintenance Rule Evaluation On CR 317575
-CR 334513; R-15 Failure Resulting In Isolation Of SG Blowdown
-CR 335951; R-15 Failure During SP-45-230 Source Check
-CR 344971; Low Flow Alarm On R-22 Sample Pump
-CR 345244; Posting For Contaminated Area Did Not Meet Expectations (Not Consistent)
-MRE 001105; R-11; Eng Elec/I&C Paper Drive
-MRE 006457; R-43; Eng Elec/I&C Light Is Burned Out
-MRE 006461; R-11; Paper Drive Chamber Leaking Containment Air Into Auxiliary Building
-MRE 006541; R-16; Flow Indicator 23160 Is Reading 0.0 gpm
-MRE 006542; R-43; Green Light Is Not Working
-MRE 006577; R-43; Eng Elec/I&C N-16 Monitor Light Is Burned Out
-MRE 006658; R-16; Eng Elec/I&C Source Check Issue
-MRE 007348; R-23; High Alarm
-MRE 010171; R-16; Flow Indicator 23161 Indicates 0 gmp

-MRE 010174; R-17; Fluctuating Counts
-MRE 010181; R-20; Excessive Amount Of Sand found In Chamber
-MRE 010314; R-19; Non-Functional As A Result Of Tubing Failure At SAP
-MRE 010422; Failure Of R-43, S/G B N16 Monitor
-MRE 010529; R-20; Shut Down When R-16 Was Scheduled
-MRE 010578; R-11; Counts Decrease Because Take-Up Spool Loose
-MRE 010653; R-13; ODCM Surveillance For Sample Flow Rate Measuring Device Not Being Met
-MRE 010654; R-14; ODCM Surveillance For Sample Flow Rate Measuring Device Not Being Met
-MRE 010655; R-21; ODCM Surveillance For Sample Flow Rate Measuring Device Not Being Met
-MRE 010665; R-15; Failure Resulting In Isolation Of SG Blowdown
-MRE 010701; R-15; Failure During SP-45-230 Source Check
-MRE 010764; N-16; Monitor R-43 Green Light Off
-MRE 010769; R-9; CPU Board Failed
-MRE 010794; R-9; Letdown Line Monitor Failure
-MRE 010877; R-11; Not functioning Properly
-N-RM-45; Radiation Monitoring System; Revision 58
-Esoms Station Narrative Logs; May 11, 13, 14, and 24, 2009
-SDBD-KPS-RM; Radiation Monitoring; Revision 0
-SP-14-117A; Auxiliary Building Special Vent System Test Train “A”; Revision 2
-SP-14-117B; Auxiliary Building Special Vent System Test Train “B”; Revision 2
-SSC Performance Criteria Sheet; System 45 Radiation Monitoring; Attachment B; Revision 1
-SSC Performance Criteria Sheet; System 08 Fire Protection System; Attachment B; Revision 2
-Dominion Central Reporting System; Condition Reports Request Data for NRC Inspection
-Kewaunee Power Station System Design Basis Document For Auxiliary Building Ventilation Systems; Revision 1
-Kewaunee Power Station USAR; Chapter 5; 5.3 Reactor Containment Vessel Isolation Systems; Revision 21
-Kewaunee Power Station USAR; Chapter 6; 6.5 Leakage Detection And Provisions For the Primary And Auxiliary Coolant Loops; Revision 21
-Kewaunee Power Station USAR; 11.2 Radiation Protection; Revision 21
-Kewaunee Power Station Work Order Overview Report; September 2, 2009
-Kewaunee Power Station Work Order Overview Report; September 17, 2009
-Maintenance Rule Safety Injection Data; 2007 To Present Time
-Maintenance Rule Scoping Questions; KPS System 45 Radiation Monitoring; Attachment A; Revision 0
-Maintenance Rule Scoping Questions; KPS System 08 Fire Protection System; Attachment A; Revision 1
-Maintenance Rule System Basis; 08 Fire Protection; Revision 5
-Maintenance Rule System Basis; 33 Safety Injection; Revision 8
-Maintenance Rule System Basis; System 45 Radiation Monitoring; Revision 8
-Radiation Monitoring System; System 45 Function 01 Data; November, 2007 through April, 2009
-Radiation Monitoring System; System 45 Function 01 Data; February, 2008 through July, 2009

-Radiation Monitoring System; System 45 Function 02 Data; February, 2008 through July, 2009
-Safety Injection Availability Hours Summary Sheet And Graphs; 2007 – 2009
-Safety Injection Condition Reports List; 2007 - 2009
-Wisconsin Public Service Corporation; Second Level Review Report; Design Change Number 553; Radiation Monitoring System; February 5, 1977
-Work Order Overview Report; May 14, 2009

1R13 Maintenance Risk

-CA 021020; Model Reg Guide 1.97 Instrumentation In Safety Monitor As Required
-CR 106141; Service Water Leak Identified on Containment Fan Coil Unit 'B'
-CR 348133; Increased Containment Sump Inleakage
-ACE 13890; Service Water Leak Identified on Containment Fan Coil Unit 'B'
-CAP 029147; NRC Resident Concern With Components Not Modeled In Safety Monitor
-MA-KW-ISP-AFW-215; Auxiliary Feedwater To Steam Generator 1A Flow Loop Calibration; Revision 0
-eSOMS Station Narrative Logs; July 22 And 23, 2009; August 4 And 5, 2009
-eSOMS Station Narrative Logs; August 15, 2008 and September 13, 2009
-Cross-Cutting Issues Data For Third Quarter 2008, Fourth Quarter 2008, First Quarter 2009, And Second Quarter 2009
-Emergent Work Risk Evaluation; Week Of July 20, 2009; Revision 2
-Kewaunee Nuclear Status Reports; July 28, 29, And 31, 2009
-Maintenance Rule 10 CFR 50.65(a)(4) Risk Projection For Week Starting August 3, 2009
-Planned Daily Activities Data For July 23, July 24, July 28, July 29, July 30, July 31, August 3, August 5, August 6, August 7, And September 14-16, 2009
-Work Week Major Activities Data For July 20, 2009; July 27, 2009, July 31, 2009, And August 3, 2009
-Work Week Major Activities Data for September 13, 2009 through September 19, 2009

1R15 Operability Evaluations

-CA 020057; PORC Question On EDG Fuel Oil Tank Vent Seismic Qualification
-CA 137422; OE28798 – NRC NOV Concerning Condensate Storage Tank Temperature Limits
-CA 145961; Track APC Issue (DCR 3765, D/G Fuel Oil Storage And Day Tank Vents)
-CAP 027146; Action Item From April 20, 2005 RIII Presentation – License Amendment Requests
-CE 017349; 95002 Readiness Team Findings Associated With EDG Ducts And FO Tank Vents
-CM-AA-401; Specification Development And Control; Revision 2
-CM-AA-PRJ-1006; Project Procurement And Vendor Interface; Revision 1
-CR 101300; Place-Keeper For the Remaining Assignments From CAP 027832
-CR 327659; Spare Containment Fan Coil Unit Motor Has Stuck Pipe Plug
-CR 328655; SSC Affected By The Degraded Or Non-Conforming Condition
-CR 336202; OE28798 – NRC NOV Concerning Condensate Storage Tank Temperature Limits
-CR 339810; SSC Affected By The Degraded Or Non-Conforming Condition
-CR 342991; Evaluation Results Of OE28798

-CR 346205; CRAC Fan “A” T1 Power Lead Found Damaged In MCC Breaker Bucket
-CR 346731; Pins Worn On Back-Draft Damper
-CR 346732; Component ASV-BD-4 Show Signs Of Worn Back Pins. Replacement Should Be soon
-CR 347730; NRC Requests That Revised OD 000169 Address Seismic Qualification Of EDG Vents
-CR 347741; NRC Informs DEK That Using TORMIX For Operability Determination Is Inappropriate
-CR 347771; APC Generated For DCR 3765, D/G Fuel Oil Storage And Day Tank Vents
-CR 347879; NOD Issue; Design Control For Offsite Repairs Of Safety-Related Equipment
-DCR 3765; Provide Separated And Redundant EDG Fuel Oil Vent Lines
-50.59/72.48 Screen for DCR 3765; Provide Separated And Redundant EDG Fuel Oil Vent Lines
-DNES-AA-GN-4002; Procurement Specifications; Revision 6
-KW 100503527; Spare Containment Fan Coil Unit Motor Has Stuck Pipe Plug
-MA-AA-102; Foreign Material Exclusion Evaluation for KW 100503527; Revision 5
-MA-KW-ECM-EHV-001; Electrical Corrective Maintenance; Revision 3
-OD 000169; EDG Exhaust Duct Operability
-OD 000276; Containment Integrity Requirements For Electrical Penetrations
-OD 000327; SSC Affected By The Degraded Or Non-Conforming Condition
-MS-AA-PRO-401; Procurement Control; Revision 1
-Badger Electric Motor Inc. Correspondence Of Purchase Order 70195377; March 20, 2009
-Kewaunee Power Station System Design Basis Document; 4160-Volt Electrical Supply System; Table 4.1-1 Safety-Related Functional Requirements
-Kewaunee Power Station USAR; Appendix B; Special Design Procedures; Revision 20
-Kewaunee Power Station USAR; Appendix B; Section B.4 Loads; Revision 21
-Kewaunee Power Station USAR; Appendix B; Section B.5 Protection Of Class I Items; Revision 21
-Kewaunee Power Station USAR; Appendix B; Section B.6 Design Criteria For Structures; Revision 21
-Kewaunee Power Station USAR; Tornado Missiles; Revision 21
-Kewaunee Power Station USAR; Figure 2.7-1; Climate of Kewaunee Site Region; Revision 21
-Kewaunee Power Station USAR; Section 2.7.5 Outside Air Temperature; Revision 21.3

1R19 Post-Maintenance Testing

-ANS-56.2; ANSI N271-1976; Containment Isolation Provisions For Fluid Systems
-CR 328160; EDG “A” Starting Air Reserve Air Receiver Leakage Unsat (OSP-DGE-006A)
-CR 332872; Work Orders To Implement DCR 3760 (Non-Outage)
-CR 335286; EDG “A” Starting Air Reserve Air Receiver Leakage Unsat (OSP-DGE-006A)
-CR 336049; Excessive Leakage On “B” DG Startup Air Receivers
-CR 341265; EDG “A” Starting Air Reserve Air Receiver Leakage Unsat (OSP-DGE-006A)

-CR 344356; During Return To Service Of The “A” Starting Air Reserve Air Receiver, An Air Leak Was Noted At The Threaded Valve Cap Area Of SA-2001A-R
-CR 344856; NRC Resident Questioned TS Application Of MS-100A
-CR 344768; Work Order Package Contains Incorrect Reference To ASME B31.1 Code
-CR 347934; Mild Water Hammer During RHR Pump Start
-KPS Pressure Test Report; ANSI B31.1/ASME Section I And VIII/NFPA; July 28, 2009
-KW100505204; EDG “A” Starting Air Reserve Air Receiver Leakage Unsat (OSP-DGE-006A)
-KW100537057; Implement DCR 3760 (Online)
-KW100537057; Asset Number Usage Data; August 24, 2009
-KW100537057; Foreign Material Exclusion Evaluation; July 28, 2009
-KW100537057; Inter and Intra Panel Wiring; System 33; August 24, 2009
-KW100537063; Implement DCR 3760 (Online)
-KW100537063; Asset Number Usage Data; August 24, 2009
-KW100537063; Foreign Material Exclusion Evaluation; August 15, 2009
-KW100537063; Inter And Intra Panel Wiring; System 33, August 24, 2009
-KW100537063; Pre-Selected Starter Buckets For Removal And Inspection; 480-Volt Starter Removal/Installation; August 24, 2009
-KW100558359; Relief Valve BT-1065A Lifting
-KW100558359; Foreign Material Exclusion Evaluation; July 24, 2009
-KW100559001; Replacement Of Steam Generator Blowdown Heat Exchanger
-KW100559001; Foreign Material Exclusion Evaluation; July 22 And 23, 2009
-KW100559001; Hot Work Permit 09-040; July 27, 2009
-KW100559001; Mechanical Work Package Checklist
-KW100559001; Rigging Lift Plan Data; July 23, 2009
-KW100559001; Tagout Tag List Data; July 23, 2009
-KW100559001; Torque Wrench Calculation Sheet Data; July 23, 24, And 25, 2009
-KW100559691; Craft Walkdown Checklist; July 23, 2009
-KW100559691; Foreign Material Exclusion Evaluation; July 23, 2009
-KW100559691; Install Coupling On Replacement Heat Exchanger Per D&Z NPS Repair Plan
-KW100559691; Piping Or Mechanical system Welded Repair Check Sheet General Requirements; July 23, 2009
-MS-100A; KPS Inservice Testing Basis Valve Data Sheet; Revision 6
-MS-101A; KPS Inservice Testing Basis Valve Data Sheet; Revision 6
-PMP-01-08; AS-Diesel Generator Startup Air Compressor Relief And Check Valve Testing And Inspection
-SP-33-098A; Train “A” Safety Injection Pump And Valve Test – IST; Revision 10
-SP-34-099A; Train “A” RHR Pump And Valve Test – IST; Revision 23
-SW-903A; KPS Inservice Testing Basis Valve Data Sheet; Revision 6
-Control Room Log Enteries; May 10, 2007, May 16, 2007, June 23, 2007, October 13, 2007, January 5, 2008, March 28, 2008, May 21, 2008, October 18, 2008, And August 12, 2009
-Day and Zimmermann NPS; Quality Control Documentation Package; Heliflow Heat Exchanger Coupling ASME Section VIII Code Repair
-Drawing E-607; Wiring Diagram; Motor Control Center 1-52E (Sh.1); Revision AX

-Drawing E-1367; Schematic Diagram M.C.C. 1-52E; Motor 1-149; Revision N/3760-1
-Drawing E-1369; Schematic Diagram M.C.C. 1-52E; Motors 1-379 And 1-387; Revision N/3760-1
-Drawing E-1375; Schematic Diagram M.C.C. 1-52E; Motors 1-104 and 1-415; Revision R
-Kewaunee Nuclear Power Plant Weld Withdrawal Slip; GMP 102; July 24, 2009
-Kewaunee Power Station Asset Report For 135-511; July 22, 2009
-Welder Performance Qualification Matrix; June 17, 2009
-Wisconsin Public Service Corporation Correspondence To U.S. NRC; Lessons Learned Response; October 19, 1979
-Wisconsin Public Service Corporation Correspondence To U.S. NRC; Lessons Learned Response; December 31, 1979

1R20 Outage Activities

-ER-KW-BAC-101-1001; KPS Site Specific Boric Acid Corrosion Control Program (BACCP) Inspection And Evaluation Requirements; Revision 1
-GNP-02.07.01; Refueling Operations – Logkeeping, Watchstanding, And Shift Turnover; Revision 3
-GNP-08.06.02; Containment Hot Shutdown Walkdown; Revision 4
-OP-KW-AOP-EHV-008; Loss Of All AC Power During Shutdown Conditions; Revision 1
-OP-KW-AOP-RHR-001; Abnormal Residual Heat Removal (RHR) System Operation; Revision 3
-OP-KW-AOP-RHR-002; Shutdown Loss Of Coolant Accident; Revision 4
-OP-KW-AOP-RHR-003; Loss Of RHR While Operating At Reduced Inventory Conditions; Revison 0
-50.59 Applicability Review of OP-KW-AOP-RHR-003; Loss Of RHR While Operating At Reduced Inventory Conditions; Revison 0
-OP-KW-AOP-SFP-001; Abnormal Spent Fuel Pool Cooling And Cleanup System Operation; Revisions 1 And 2
-CR 350088; Barrier Impairment Permit Not Posted With Barrier Breached

1R21 Component Design Bases Inspection

-C11721; Bus Fast Transfer Calculation; July 30, 2008
-CA041804; NRC Concern with Transferring Safety-related Buses to Alternate Source; February 16, 2007
-CA029876; Take Corrective Actions to Resolve Conditions Identified in CAP 041804; February 19, 2007
-CA029877; Take Corrective Actions to Resolve Conditions Identified in CAP 041804; February 19, 2007
-CA029877; Take Corrective Actions to Resolve Conditions Identified in CAP 041804; February19, 2007
-CA030398; Take Corrective Actions to Resolve Conditions Identified in CAP 041804; March 9, 2007
-CR030628; NRC Concern with Transferring Safety-Related Buses to Alternate Source; February 20, 2007
-CA031463; Create Procedure to Load EDG; April 24, 2007
-EFR 03128; Perform Effectiveness Review of 0DM030393; April 14, 2007
-CA081121; Bus Transfer Onto RAT; August 14, 2008

-CR096058; CAP 041804 Place Keeper; April 18, 2008
-CR 320115; C11450 R1 Suggests Margin Improvement for Spray Pump Protection; January 15, 2009
-CR 320794; CR 320794; January 21, 2009
-CR 335822; Technical Specification (TS) 3.7, Auxiliary Electrical Systems, Clarifications; May22, 2009
-CR 344370; Fast Transfer to EDG from DVR not Evaluated; August 10, 2009
-E-221; Metering and Relaying Diagram – Generator and 4160-V Equipment; Revision AB
-E-1037; Schematic Diagram - 4160-V Breaker 1-503; Revision W
-E-1872; Schematic Diagram – Bus 5 Voltage Restoring; Revision Z
-E-1874; Schematic Diagram – Bus 5 Voltage Restoring; Revision X
-LER 2007-007-00; Unexpected Safety Injection Response with Safeguard Buses Connected to the Reserve Auxiliary Transformer; March 23, 2007
-Volume RA.KPS.001; Risk Assessment for Removal of Automatic Loading on Bus 5 onto the RAT; Revision 1
-KPS USAR Ch. 8; Electrical System; Revision 20
-IEEE-279; Proposed IEEE Criteria for Nuclear Power Plant Protection Systems; August 1968
-NRC SER; July 24, 1972
-K-93-203; Amendment No. 101 to Facility Operating License No. DPR-43 (TAC No. M84406); October 8, 1993
-N-EHV-39; 4160-V AC Supply and Distribution System Operation; April 8, 2007

1R22 Surveillance Testing

-CA 021020; Model Reg Guide 1.97 Instrumentation In Safety Monitor As Required
-CAP 029147; NRC Resident Concern With Components Not Modeled In Safety Monitor
-CR 343080; Change In Unquantified RCS Leakage is Greater Than 0.05 gpm Above Baseline
-CR 344769; PS-15522J fount Out-Of-Spec During MA-KW-ISP-001A
-KW100537057; Implement DCR 3760 (Online)
-KW100537057; Asset Number Usage Data; August 24, 2009
-KW100537057; Foreign Material Exclusion Evaluation; July 28, 2009
-KW100537057; Inter and Intra Panel Wiring; System 33; August 24, 2009
-KW100537063; Implement DCR 3760 (Online)
-KW100537063; Asset Number Usage Data; August 24, 2009
-KW100537063; Foreign Material Exclusion Evaluation; August 15, 2009
-KW100537063; Inter And Intra Panel Wiring; System 33, August 24, 2009
-KW100537063; Pre-Selected Starter Buckets For Removal And Inspection; 480-Volt Starter Removal/Installation; August 24, 2009
-MA-KW-ISP-AFW-215; Auxiliary Feedwater To Steam Generator 1A Flow Loop Calibration; Revision 0
-MA-KW-ISP-SW-001A; Service Water Header “A” Pressure Switch Calibration; Revision 1
-OP-KW-OSP-RCS-001; Reactor Coolant System Leak Rate Check; Revision 2
-SP-14-026A; Auxiliary Building Ventilation Train “A” Operability Test; Revision 10
-SP-33-098A; Train “A” Safety Injection Pump And Valve Test – IST; Revision 10
-Drawing E-607; Wiring Diagram; Motor Control Center 1-52E (Sh.1); Revision AX

-Drawing E-1367; Schematic Diagram M.C.C. 1-52E; Motor 1-149; Revision N/3760-1
-Drawing E-1369; Schematic Diagram M.C.C. 1-52E; Motors 1-379 And 1-387; Revision N/3760-1
-eSOMS Station Narrative Logs; August 13 And 14, 2009
-Planned Daily Activities Data for July 28, 2009

1EP6 Drill Evaluation

-KPS Exercise; Section 1; Exercise Ground Rules; August 25, 2009
-Dominion Energy Kewaunee; Kewaunee Power Station; Emergency Preparedness Training Drill; Revisions 0 And 1; August 25, 2009

2PS2 Radioactive Material Processing and Transportation

-CR102394; NOD Audit 08-06: Guideline Used To Survey Filter High Integrity Containers; October 30, 2008
-CR025312; Different Gamma Analysis Results For Radwaste Samples; June 24, 2008
-CR017632; Temporary Change To Procedure HP 9.004; April 18, 2008
-CR118618; NOD Recommendation: RP To Develop And Present to Sr. Management A Radwaste Plan; April 13, 2009
-CR105585; 341495; Resin Found In Waste Metering Tank Room During NRC Inspection; July 15, 2009
-CR341489; 2006 10CFR61 DAW Waste Stream Analysis Not Found In Records Vault; July 5 2009
-CR341645; Evaluate Sponge Media Waste In Drumming Room For Shipment Or Interim Storage; July 15, 2009
-CR341658; Initiate Work Order To Repair/Replace Level Indicator For Waste Metering Tank; July 16, 2009
-HP-09.015; High Integrity Containers Operating Guidelines; Revision 6
-HP-09.004; Filling And Dewatering Radwaste Containers; Revision 8
-NAD-01.16; Solid Radioactive Waste Process Control Program (PCP); Revision H
-RP-KW-009-011; Waste Stream Analysis; Revision 0
-Radioactive Waste Shipment Package; Manifest Number SPF0607-07; June 7, 2007
-Radioactive Waste Shipment Package; Manifest Number 041808-1; April 18, 2008
-Radioactive Waste Shipment Package; Manifest Number 050908-1; May 9, 2008
-Radioactive Waste Shipment Package; Manifest Number 102308-2; October 23, 2008
-Radioactive Waste Shipment Package; Manifest Number 061108-1; June 11, 2008
-Radioactive Waste Shipment Package; Manifest Number060309-1; June 3, 2008
-Audit 07-06: Radiological Protection, Process Control Program, And Chemistry; August 4, 2008
-Audit 08-06; Radiological Protection/Process Control Program; September 5, 2008

-2008 – 10 CFR 61 Compliance Data Technical Basis For DAW Waste Stream; July 9, 2009
-2008 – 10 CFR 61 Compliance Data Technical Basis For Resin Waste Stream; July 9, 2009
-2008 – 10 CFR 61 Compliance Data Technical Basis For Filter Media Waste Stream; July 11, 2009
-LP No. RPA-01-LP003; 49CFR Regulatory Awareness Training For Radiation Protection; Revision B

4OA1 Performance Indicator Verification

-ACE 013979; Repeat Occurrences Of A “Lit” LED On The SBV Train “A” Exhaust Damper Servo-Amplifier Card
-ACE 014071; Breaker 14604BKR Was Tripped From The Control Room (With Control Switch) To Unload The Technical Support Center Diesel Generator From Bus 1-46
-ACE 014079; Service Water Pump A2 Breaker Did Not Close
-ACE 017352; SW Pump A1 Red Run Indication Light Not Lit In Control Room
-CA 126135; Evaluate Strategy Of Rotating Spare PM’d 480 VAC Metal Clad And 4160 VAC Vacuum
-CR 104624; SBV Train “A” S/CV 35109 led Found “On” When It Normally Is “Off” When Fan in “Off”
-CR 104768; SBV-10A And SBV-20A Failed To Modulate During Sp-24-107A
-CR 106929; Repeat Occurrences Of A “Lit” LED On The SBV Train “A” Exhaust Damper Servo- Amplifier Card
-CR 109107; SW Pump A1 Red Run Indication Light Not Lit In Control Room
-CR 119353; Breaker 14604BKR Was Tripped From The Control Room (With Control Switch) To Unload The Technical Support Center Diesel Generator From Bus 1-46
-CR 119799; Service Water Pump A2 Breaker Did Not Close
-CR 317178; Auxiliary Building Basement Fan Coil Unit A Red Lamp Failed To Light During SP-45-049.13
-CR 318324; SW Pump A1 Red Run Indication Not Lit In Control Room
-CR 325172; While Stopping SW Pump B2 SER Point 694 Service Water Pump B2 Breaker 1-609
Open Alarmed Then Cleared And Alarmed Again All Within One Second
-CR 347062; ACE 14079 Has Been In Manager Review Since May Of 2009
-KPS Inservice testing basis valve Data Sheet; System 36-RC; Reactor Coolant; Revision 6
-KW 100282718; PM87-134 – Calibrate Relay 802200
-LA 001187; Perform SMPI/SSFF Review For LER 2008-001
-LI-AA-500; NRC/INPO/WANO Performance Indicator And MOR Reporting; Revision 1
-MRE 006905; Controller Failing To Modulate SBV-10A And SBV-20A
-MRE 010195; Breaker 1-506 – SW Pump A1 Red Run Indication Not Lit In Control Room
-NRC Regulatory Issue Summary 2004-03; Risk-Informed Approach For Post-Fire Safe-Shutdown Circuit Inspections; Revision 1
-RP-AA-112; Radiation Safety Performance Indicator Reporting; Revision 2
-OD 000194; Pressurizer PORV-2B; Control Cable Spurious Operation Concern
-ODM 000079; Perform ODM On Breaker Issue Associated With CR 318324
-SP 32B-268; Semi-Monthly Site Boundary Dose Update; January 7, 2009 Through June 5, 2009
-Control Room Log entries Reports; July 22 and 23, 2008, October 10, 2008, November 7 and 17, 2008, December 21, 22, 23, and 29, 2008

-Drawing E-1040; Control Schematic 4160-V Breaker 1-506; Revision Y
-Drawing E-1041; Control Schematic 4160-V Breaker 1-507; Revision Y
-Primary Leak Rate Reactor Coolant System Leakage Data; Third Quarter 2008 Through Second Quarter 2009
-LI-AA-500; NRC/INPO/WANO Performance Indicator and MOR Reporting; Revision 1
-RP-AA-112; Radiation Safety Performance Indicator Reporting; Revision 2
-CDE Audit Log; Maximum I-131 Activity (uCi/gm); dated November 2008 through June 2009
-CDE Audit Log; High Radiation Area Occurrences, Very High Radiation Area, and Unintended Exposure Occurrences; dated November 2, 2008 through May 1, 2009
-SP 32B-268; Semi-Monthly Site Boundary Dose Update; January 7, 2009 through June 5, 2009

4OA3 Follow-up of Events and Notices of Enforcement Discretion

- Door 140 Pressure Test ; Test Plan; Revision 0

4OA5 Other Activities

- 72.48 Screening; MA-KW-ICP-ISF-001, Revision 0
- 72.48 Screening; MA-KW-GMP-ISF-002, Revision 0
- 72.48 Screening; OP-KW-NOP-ISF-009, Revision 0
- 72.48 Screening; OP-KW-NOP-ISF-003, Revision 0
- 72.48 Screening; PI-900729-02, Revision 1
- 72.48 Screening; RP-KW-ISF-001, Revision 0
- Administrative Procedure MA-AA-101; Fleet Lifting and Material Handling; Revision 4
- Administrative Procedure MA-AA-OCR-101; Overhead Cranes/Hoists; Revision 1
- Administrative Procedure MA-AA-RHW-101; Rigging Hardware; Revision 1
- ACECO Procedure REP 20776-009; Factory Load Test Plan for Kewaunee Auxiliary Building Crane; Revision 1
- ACECO Procedure REP 20776-010; Site Functional Test for Kewaunee Auxiliary Building Crane; Revision 3
- ACECO Procedure REP-20776-011; Site Load Test Plan for Kewaunee Auxiliary Building Crane; Revision 4
- ACECO Procedure REP-20776-016; Operator Training for Auxiliary Building Crane; Revision 0
- Airgas Certificate of Analysis; Helium Ultra High Purity; May 12, 2009
- Amendment 200 approved by NRC in a letter dated November 20, 2008
- Amendment 205 approved by the NRC in a letter dated April 20, 2009
- Areva Certificate of Conformance; Spent Fuel Storage Cask; DSC Serial Number KPS32PT-S100-A-HZ001; Revision 2
- Areva Certificate of Conformance; Spent Fuel Storage Cask; DSC Serial Number KPS32PT-S100-A-HZ003; Revision 2
- Dominion Nuclear Fuel Defect Chronology; Version 2009-A; April 30, 2009
- License Amendment Request LAR-227, Attachment 6; NUREG-0554 Compliance Matrix for Upgraded KPS AB Crane
- Calculation C11846; Evaluation of the Auxiliary Building Using Revised Crane Wheel Loads Due to the Fuel Handling Area Crane Upgrade to Single-Failure-Proof; Revision 0
- Calculation C11846; Evaluation of the Auxiliary Building Using Revised Crane Wheel Loads Due to the Fuel Handling Area Crane Upgrade to Single-Failure-Proof; Revision 1
- Calculation 11862-010-ST-01; Spent Fuel Pool Cask Load Path Evaluation Inside Auxiliary Building; Revision 1
- Calculation 11862-010-ST-01; Spent Fuel Pool Cask Load Path Evaluation Inside Auxiliary Building; Revision 2

- Calculation 11862-010-ST-01; Spent Fuel Pool Cask Load Path Evaluation Inside Auxiliary Building; Revision 3
- Calculation CAL-20776-S05; Calculation 125/15 Ton Auxiliary Building Crane Bridge Girder Modifications for Defective Butt Welds; Revision 0
- Calculation HI-2-73752; WPMR Analysis of Freestanding Rack in North Pool of Kewaunee Power Station; Revision 2
- Calculation ISFSI-04036CG; Design of 105-Ton Cask-Lifting Beam; Revision 0
- Calculation ISFSI-04036CG; Design of 105-Ton Cask-Lifting Beam; Revision 1
- Calculation ISFSI-04036CG; Design of 105-Ton Cask-Lifting Beam; Revision 1, Addendum A
- Calculation ISFSI-04036CG; Design of 105-Ton Cask-Lifting Beam; Revision 2
- Calculation ISFSI-04036CG; Evaluation of Bending Stresses in 6" Pins for the 105 Ton Cask-Lifting Beam; Revision 2, Addendum A
- Calculation ISFSI-04036CG; Evaluation of Bending Stresses in 6" Pins for the 105 Ton Cask-Lifting Beam; Revision 2, Addendum B
- Calculation KPS-70155344-S01; Development of Response Spectra at Auxiliary Building Crane Rail Level; Revision 1
- Calculation KPS-70155344-S02; Evaluation of Auxiliary Building Crane Girder; Revision 1,
- Calculation KPS-70155344-S02; Evaluation of Auxiliary Building Crane Girder; Revision 1, Addendum A
- Calculation KPS-70155344-S03; Evaluation of Auxiliary Building Superstructure for Single-Failure-Proof Crane Upgrade; Revision 0, Addendum A
- Calculation SF-0004; Hook and Yoke Weight of the NUHOMS OS-197H Transfer Cask and 32PT Dry Shielded Canister During the Loading Process; Revision 0
- CR095357; Rust Found on HSM Roof Sections; April 10, 2008
- CR095447; Minor/Cosmetic Damage on HSM Roofs; April 13, 2008
- CR095578; Greater than Cosmetic Damage to Concrete HSM Roof; April 14, 2008
- CR095688; White Residue Found on HSM Heat Shields; April 15, 2008
- CR096604; Damage to HSM Roof; April 23, 2008
- CR096813; NOD ID's Drawing Discrepancy in ISFSI Assembly Procedure; April 25, 2009
- CR097366; High Spot on ISFSI Pad Interferes with Placement of End Wall; April 30, 2009
- CR097492; Unable to Set HSM 7 Roof Within Tolerance Called Out by Installation Spec; May 1, 2008
- CR098369; Minor/Cosmetic Damage on Top Corners of End Walls; May 7, 2008
- CR098372; Minor/Cosmetic Damage Occurred During Installation of Birdscreen on Roof K; May 7, 2008
- CR098488; Corrective Action not Generated for ACE 709; May 9, 2008
- CR099598; HSM Door Lift Rig Doesn't Attach to HSM Doors; May 19, 2009
- CR102714; ISFSI Heavy Haul Path Requires Repair; July 1, 2008
- CR105314; Work Package Not Providing Detailed Information for Adequate Radiation Work Permit; August 5, 2008
- CR106678; DSC Weld Preps May Have Incorrect Dimension, August 21, 2008
- CR111597; Generate WO for SFP Activities Needed to Ready Pool for ISFSI; September 29, 2008
- CR116982; Request Training for EALS Associated with ISFSI; October 31, 2008
- CR128494; Determine Contingency Actions Needed to Enable NW Rack to Be Removed; May 26, 2009
- CR128498; DSC Lid Weld Preps Found out of Spec; February 4, 2009
- CR331141; WOs Requested for ISFSI Major Activities; April 16, 2009
- CR322911; Insufficient Travel on Auxiliary Building Crane Hoist to Remove the NW Rack; February 9, 2009

- CR322924; Lower Lift Rig Legs, Plates, and Bolts Missing Needed for ISFSI SFP Modification; February 11, 2009
- CR332885; Work in SFP Behind Schedule; April 24, 2009
- CR333277; Fuel Rack Lift Not in Accordance with Original Kewaunee Lift Plan; May 2, 2009
- CR333488; SFP Rigging Experiencing Oxidation; May 4, 2009
- CR335187; Delay in Progress in Fuel Pool Due to Tooling Issues; May 18, 2009
- CR337789; NRC Identified that Crane Inspection Procedure Missing Some Required Inspection; June 10, 2009
- CR337849; Minimum Lift Height for AB Crane Should Be in Procedures; June 9, 2009
- CR337872; Frequency for Visual Inspection on Cask Lift Yoke Not Per ANSI Standard; June 10, 2009
- CR337889; Wire Rope Removal from Service Criteria
- CR338428; AB Crane Operator Qualification Documentation Not Found; June 17, 2009
- CR339267; KEWA - NRC Identified Discrepancy in Transfer Cask Lift Yoke Calculation; June 24, 2009
- CR339498; Error Found in Spent Fuel Pool Calculation; dated June 25, 2009
- CR339783; Holtec Field Nonconformances for Rack Installation; June 29, 2009
- CR339895; KEWA - NRC Comments on Calculation 11862-010-ST-01; June 30, 2009
- CR341633; NRC Concern Regarding Calculation C11846; dated July 16, 2009
- CR344788; Auxiliary Building Crane Position Indication Not Working; August 13, 2009
- CR344795; Welding Plug Changed Out on PCI Welding Equipment; August 13, 2009
- CR344804; Distraction During Critical Lift Brief; August 13, 2009
- CR344805; Wasp Nest Next to the Horizontal Storage Modules; August 13, 2009
- CR345252; SFP Retrieval Tools Need to Be Upgraded; August 18, 2009
- CR345326; ISFSI Lost Power to Welding Equipment During Night Shift; August 18, 2009
- CR345343; ISFSI Welder Did Not Weld Automatically Around Keyway; August 19, 2009
- CR345475; PCI Weld Equipment Was Not Properly Set-Up; August 20, 2009
- CR347138; NRC Observations from First Loading Campaign; September 2, 2009
- Crosby Group, Inc. Crane Hook Load Testing Certificate 648855; May 23, 2007
- Document from PCI to Dominion; Out-of-Tolerance Weld Prep on the Kewaunee Inner and Outer DSC Lids; June 10, 2009
- Document of Information Sharing; Mechanical Maintenance Aux. Building Crane Upgrades Briefing; June 30, 2009
- Document No. MA-KW-MCM-ISF-001; HSM Concrete/Grout Repair; March 20, 2008
- Dominion Mechanical Maintenance Training Program; Training Exemption/Alternate Qualifications TR-AA-400 Attachment 3; June 18, 2009
- Drawing D-20602-001; 105 Ton Capacity Lift Beam, General Arrangement; Revision 4
- DSC Physical Inventory NUHOMS-32PT; Cask No. DSC-01; August 18, 2009
- Dummy Fuel Assembly Dry Run Testing Movement Sequence; August 3, 2009
- Electrical Maintenance Procedure MA-KW-EPM-CRN-001; Auxiliary Building Fuel Handling Crane Inspection; Revision Draft
- Fuel Assembly Certification; Canister CoC ID KPS32PT-s100-A-HZ001; August 18, 2009
- Fuel Movement Sequence; Form RE-25-1; August 17, 2009
- Fuel Reliability Indicator Data; 1986-2009
- Fuel Transfer/Inventory Authorization; Dummy Fuel Assemblies; July 14, 2009
- General Nuclear Procedure GNP 08.12.02; Controls for Use of Cranes Within the Protected Area; Revision 19
- Health Physics Procedure RP-KW-ISF-001; Dry Shielded Canister Surveys; Revision 0
- Hitachi Zosen Diesel and Engineering Co LTD Proof Load Test Record; January 10, 2006
- ISFSI Fire Hazard Analysis; Report No. SL-009169; Revision 1

- ISFSI Fuel Assembly Certification and ISFSI Canister Loading Map; Canister CoC ID: KPS32PT-S100-A-HZ001; Revision 0
- ISFSI Fuel Assembly Certification and ISFSI Canister Loading Map; Canister CoC ID: KPS32PT-S100-A-HZ003; Revision 0
- Job Aid 01-003; Needs Assessment Worksheet; EAL change to ISFSI
- Kewaunee ISFSI 10 CFR 72.212 Evaluation Report; August 5, 2009
- Kewaunee ISFSI Fuel Assembly Certification and Loading Map for KPS32-PT-S100-A-HZ003; Revision 0
- Kewaunee Power Station Emergency Action Levels Chart
- Kewaunee Power Station ISFSI Fire Hazards Analysis; Revision 1; February 10, 2009
- Licensee Training Document; Crane Course Completion Status
- Modification DCR 3629; Upgrade Auxiliary Building Crane to Be Single-Failure-Proof; Revision 1
- Modification DCR 3633; Modify Seismic Support Structure in North Spent Fuel Pool; Revision 0
- Mechanical Maintenance Procedure MA-KW-MPM-CRN-002; Auxiliary Building Fuel Handling Crane Annual Maintenance; Revision 1
- Mechanical Maintenance Procedure MA-KW-MPM-CRN-005; Auxiliary Building Crane Main and Auxiliary Hoist Lower Block Pre-Submersion Maintenance; Revision 1
- Mechanical Maintenance Procedure MA-KW-MPM-CRN-006; Auxiliary Building Crane Main and Auxiliary Hoist Lower Block Post-Submersion Maintenance; Revision 1
- Mechanical Maintenance Procedure MA-KW-MPM-CRN-007; Monthly Inspection of Auxiliary Building Fuel Handling Crane; Revision 0
- Normal Operating Procedure OP-KW-NOP-ISF-001; Dry Shielded Canister Loading; Revision 3
- Normal Operating Procedure OP-KW-NOP-ISF-002; Vacuum Drying System Operations; Revision 0
- Normal Operating Procedure OP-KW-NOP-ISF-003; Dry Shielded Canister Insertion Into HSM; Revision 1
- On-the-Job Training; Task Performance Evaluation Guide; Auxiliary Operator; DSC Draining and Drying Operations; Document No. AO-250-OJ011; Revision A
- PCI Procedure GQP-9.2; High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials, and Cladding; Revision 3
- PCI Procedure PI-CNSTR-T-OP-250; Closure Welding of Dry Shielded Canisters at the Millstone and Kewaunee Stations; Revision 0
- Preventive Maintenance Task RE89105; NJP PM53-055; Inspect Critical Welds
- Procedure OP-KW-ARP-47034-24; ISFSI Temperature Monitoring System; Revision 0
- Procedure SP-87-148; Daily Instrument Channel Checks; Revision 34
- Qualification Matrix; Kewaunee Power Station ISFSI tasks; July 20, 2009
- Radiation Protection Procedure RP-AA-202; Radiological Posting; Revision 2
- Radiation Protection Procedure RP-AA-203; Radiological Labeling and Marking; Revision 0
- Radiation Protection Procedure RP-AA-223; Contamination Surveys; Revision 0
- Radiological Survey Records for DSC #1; August 17-22, 2009
- Reactor Engineering Procedure RE-24; Special Nuclear Materials Control; June 30, 2009
- Reactor Engineering Procedure RE-25; Fuel Movement During Non-Refueling Operations; Revision 12
- Reactor Engineering Surveillance Procedure NF-KW-RSP-FH-004; Revision 8
- Supplier Nonconformance Evaluation; CR 096604 Concrete Spall; April 29, 2009
- Supplier Nonconformance Evaluation; CR 097492 HSM Front Face Alignment Out of Tolerance; May 30, 2008

- Supplier Nonconformance Report; DSC Canister Out of Tolerance Field Weld Preparations on Top Cover Plates; June 11, 2009
- Surveillance Procedure C-SP-604.3; Transfer Cask Lift Yoke Inspections; Revision 000-02 (Millstone)
- Unirope LTD Test Certificate No. 86187 Item #1; Hoist Wire Rope Destructive Testing; March 30, 2007
- Vacuum Drying Rate of Rise Test Results; August 7, 2009
- Vacuum Drying Rate of Rise Test Results; August 19, 2009
- Welder Maintenance Log; PCI Energy Services
- Welder Operator Performance Qualification; PCI Energy Services
- Welder Performance Qualification; PCI Energy Services
- Work Order KW100450235; Monthly Auxiliary Building Fuel Handling Crane Inspection; May 12, 2009
- Work Order KW100274498; Auxiliary Building Fuel Handling Crane Inspection; March 2, 2009
- Work Order 53102222800; Transfer Cask Lift Yoke Inspections (Millstone)

4OA7 Licensee-Identified Violations

-CR103866; Radioactive Liquid Waste Discharge Permit Completed by Unqualified Technician; July 18, 2008

LIST OF ACRONYMS USED

ABFH	Auxiliary Building Fuel Handling
AC	Alternating Current
ACECO	American Crane and Equipment Corporation
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Reasonably-Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AWS	American Welding Society
BACC	Boric Acid Corrosion Control
CA	Corrective Action
CAP	Corrective Action Program
CDBI	Component Design Basis Inspection
CFCU	Containment Fan Coil Unit
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CR	Condition Report
DAW	Dry Active Waste
DOT	Department of Transportation
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
DSC	Dry Shielded Canister
DVR	Degraded Voltage Relay
EDG	Emergency Diesel Generator
ESF	Engineered Safety Features
F	Fahrenheit
FSAR	Final Safety Analysis Report
HSM	Horizontal Storage Modules
ICS	Internal Containment Spray
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
IST	Inservice Testing
KPS	Kewaunee Power Station
kV	kilovolts
kW	kilowatts
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
LSA	Low Specific Activity
mrem	Millirem
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ODCM	Offsite Dose Calculation Manual
OJT	On-the-Job Training
OSP	Outage Safety Plan
PARS	Publicly Available Records
PCI	PCI Energy Services

PI	Performance Indicator
RAT	Reserve Auxiliary Transformer
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specification
RFO	Refueling Outage
SCO	Surface Contaminated Object
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SI	Safety Injection
SSC	Systems, Structures, and Components
SSFF	Safety System Functional Failure
TAT	Tertiary Auxiliary Transformer
TDAFW	Turbine-Driven Auxiliary Feedwater
TS	Technical Specification
USAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volt
WO	Work Order

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 Kewaunee Power 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael A. Kunowski, Chief
Branch 5
Division of Reactor Projects

Docket No. 50-305
License No. DPR-43

Enclosure: Inspection Report 05000305/2009004
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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NAME	JCassidy:ls		WSlawinski		SOrth		JJandovitz for SBurton:cms		MKunowski	
DATE	09/28/09		09/28/09		11/13/09		11/12/09		11/13/09	

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**indicates also OE concurrence and OGC no/legal objection*

Letter to D. Heacock from M. Kunowski dated November 13, 2009

SUBJECT: KEWAUNEE POWER STATION INTEGRATED INSPECTION REPORT 05000305/2009004

DISTRIBUTION:
ADAMS (PARS)
RIDSSECYMAILCENTER.Resource
OCA Distribution
Bill Borchardt
Bruce Mallett
Roy Zimmerman
Belkys Sosa
Gerald Gulla
Marvin Itzkowitz
Catherine Marco
Mark Satorius
Cynthia Pederson
Eric Leeds
Bruce Boger
Daniel Holody
Carolyn Evans
William Jones
Mary Ann Ashley
Holly Harrington
Hubert Bell
Guy Caputo
Mona Williams
Jared Heck
Allen Barker
James Lynch
Harral Logaras
Viktoria Mitlyng
Prema Chandrathil
Patricia Lougheed
Paul Pelke
Magdalena Gryglak
OEMAIL Resource
Susan Bagley
RidsNrrDorlLpl3-1 Resource
RidsNrrPMKewaunee
RidsNrrDirsIrib Resource
Steven Orth
Carole Ariano
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports Resource