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

May 12, 2010

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 05000305/2010002

Dear Mr. Heacock:

On March 31, 2010, 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 April 1, 2010, with Mr. Stephen 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, one NRC-identified and two self-revealed findings of very low safety significance were identified. The three 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, two licensee-identified violations are listed in Section 4OA7 of this report.

If you contest the subject or severity of an NCV, 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 and its enclosure 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 05000305/2010002
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServe

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-305
License No:	DPR-43
Report No:	05000305/2010002
Licensee:	Dominion Energy Kewaunee, Inc.
Facility:	Kewaunee Power Station
Location:	Kewaunee, WI
Dates:	January 1, 2010, through March 31, 2010
Inspectors:	R. Krsek, Senior Resident Inspector R. Ruiz, Senior Resident Inspector (Acting) K. Barclay, Senior Resident Inspector (Acting) J. Cassidy, Senior Health Physicist D. McNeil, Senior Operations Engineer R. Winter, Reactor Inspector R. Edwards, Reactor Engineer J. Jandovitz, Project Engineer A. Dahbur, Senior Reactor Inspector J. Bozga, Reactor Inspector
Approved by:	Michael A. Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000305/2010002; 01/01/2010 – 03/31/2010; Kewaunee Power Station; Follow-Up of Events and Notices of Enforcement Discretion.

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

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance and associated Non-Cited Violation of Technical Specification 3.8.a.5 was self-revealed when the licensee loaded fuel into the reactor with reactor coolant system boron sample results less than the minimum boron concentration as specified in the core operating limits report. Once the licensee believed the boron concentration samples were accurate and that boron concentration was below the required minimum, operators stopped moving fuel until the boron concentration was restored to acceptable limits. The licensee entered the issue into the corrective action program as Condition Report (CR) 351923. The licensee conducted an apparent cause evaluation and proposed long-term corrective actions, including procedure enhancements, operator training on the event, and conservative decision making training.

  This finding was determined to be more than minor because it was associated with the Initiating Events Cornerstone attribute of human 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 operations. Specifically, the licensee did not believe the initial boron sample results and continued to move fuel with actual boron concentrations below the minimum value specified in the core operating limits report. The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations SDP." The inspectors used Checklist 4 contained in Attachment 1 and determined that the finding did not require a phase 2 or phase 3 analysis and screened as very low safety significance (Green). This finding has a cross-cutting aspect in the area of human performance, decision-making, because the licensee failed to use conservative assumptions when making decisions and did not demonstrate that nuclear safety was an overriding priority (H.1(b)). (Section 4OA3)

- Green. A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was self-revealed for the failure to establish adequate measures to identify and control design interfaces and coordinate among participating design organizations. Specifically, the licensee failed to adequately control all required tertiary auxiliary transformer relay inputs/settings that interfaced with the existing plant design. This adversely impacted associated equipment and caused an unanticipated system response. The licensee promptly cleared tags on the reserve auxiliary transformer to restore a normal offsite power source to one of the two 4160-volt safeguards buses. The licensee performed a root cause evaluation and implemented corrective actions, some of which included: modifying the design change process to ensure that all programmable digital device setpoints and inputs were identified; documenting the basis for each setpoint or input in the design change documentation; and providing programmable digital device training for design engineering and maintenance personnel. The licensee entered the issue into its corrective action program as CR 352878.

The finding was determined to be more than minor because the finding was associated with the Initiating Events Cornerstone attribute of design control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to adequately control all required tertiary auxiliary transformer relay inputs/settings adversely impacted the associated equipment, which caused an unanticipated system response and challenged core shutdown cooling. The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, “Shutdown Operations SDP.” The inspectors used Checklist 4, contained in Attachment 1, and determined that the finding required a Phase 2 analysis because it degraded the ability to recover the decay heat removal system. The Region III senior reactor analyst performed a phase 2 and subsequently a phase 3 analysis and determined the finding was of very low safety significance (Green). This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee did not maintain complete, accurate, and up-to-date design documentation (H.2(c)). (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified by the inspectors for inadequate surveillance calibration procedures. Specifically, calibration surveillance procedure SP-06-034B-1, “Steam Generator Flow Mismatch and Steam Pressure Instrument Channel 1,” failed to have the correct negative ramp curve. The curve was required to ensure that the low steam line pressure safety injection lag circuitry unit did not exceed the Technical Specification setpoint value. This condition also existed in calibration procedures for channels 2, 3, and 4. The licensee subsequently entered the issue into its corrective action program as CR 367826 and CR 367932. The licensee conducted an apparent cause evaluation and corrective actions were in progress at the conclusion of the inspection period.

The finding was determined to be more than minor because it was associated with the Mitigating System Cornerstone attribute of procedure quality and adversely affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to ensure that the low steam line pressure safety injection lag circuitry units did not exceed the Technical Specification value of less than or equal to 2 seconds. The finding was of very low safety-significance (Green) based on a phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The finding has a cross-cutting aspect in the areas of human performance, work practices, because the licensee failed to ensure that the calculation upon which the surveillance procedure was based, was approved prior to issuance of the procedure (H.4(b)). (Section 4OA3)

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's corrective action program. These 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, with one exception. On January 22, 2010, Kewaunee experienced an unplanned power change when a feedwater heater drain pump failed and caused the licensee to reduce power to 88 percent. The licensee repaired the pump and returned to full power on January 24, 2010.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis maximum probable flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report (USAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site, which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one external flooding sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

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:

- “B” emergency diesel generator (EDG);
- component cooling water train “B” after pump replacement; and
- auxiliary feedwater train (AFW) “B” with the turbine-driven AFW pump out-of-service.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, Technical Specification (TS) requirements, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events, or impact the capability of mitigating systems or barriers. Lastly, the inspectors verified the licensee entered the issues 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 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 SC-70A, screen house, “A” train of service water;
- fire zone SC-70B, screen house, “B” train of service water;
- fire zone TU-92, TU-93, “B” EDG and day tank rooms;
- fire zone AX-32, service rooms; and
- fire zone TU-95B, 480-volt switchgear bus 1-61 and 1-62 room and AFW pump area.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that: adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and implemented adequate

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

These activities constituted five quarterly fire protection inspection samples 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 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. 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 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 flooding zone 226, the control room, to look for sources of internal flooding that were not analyzed or properly maintained and to also verify that the licensee complied with its commitments. 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.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On February 22, 2010, 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 the licensee's conduct of operations procedure 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.

.2 Conformance with Operator License Conditions (71111.11B)

a. Inspection Scope

The inspectors reviewed the licensee's procedures for, and response to, detecting an operator under the influence of alcohol in the control room. The inspectors compared the facility response with requirements described in 10 CFR 50.9, "Completeness and accuracy of information," with 10 CFR 55.21, "Medical examination," with 10 CFR 55.25, "Incapacitation because of disability or illness," with 10 CFR 50.74(c), "Notification of change in operator or senior operator status," with 10 CFR 26.719, "Reporting requirements," and with American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.4 – 1996, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants." Documents reviewed are listed in the Attachment to this report.

b. Findings

In August 2008, the licensee tested a control room operator on a program of increased alcohol testing frequency due to a November 2007 arrest for driving while intoxicated. The licensee determined the operator was under the influence of alcohol while performing licensed operator duties in the control room. The licensee removed the operator from the control room and re-checked all activities performed by the operator for accuracy. No errors in the operator's performance were discovered. The licensee notified the NRC in accordance with 10 CFR 26.719(b). The licensee subsequently requested the NRC expire the operator's NRC operating license. It was determined that the licensee complied with all applicable requirements when the licensee discovered the operator under the influence of alcohol in the control room.

Based on the above discussion, unresolved item (URI) 05000305/2008005-01 is considered closed.

.3 Examination Security (71111.11B)

a. Inspection Scope

The inspectors reviewed several condition reports, corrective action program items, and apparent cause evaluations related to examination security issues at the station. The inspectors reviewed the licensee's overall licensed operator requalification examination security program related to examination physical security (e.g., access restrictions and simulator considerations). The inspectors reviewed the facility licensee's examination security procedure, "KPS Simulator Security Checklist (Job Aid: 04-009)," and corrective actions related to past and present examination security problems at the facility. These items were reviewed to verify compliance with 10 CFR 55.49, "Integrity of examinations and tests." Documents reviewed are listed in the Attachment to this report.

b. Findings

One violation of NRC requirements of very low safety significance involving a compromise of a licensee administered requalification operating test was identified by the licensee. See Section 4OA7 of this report for additional details.

Based on the above discussion, URI 05000305/2009005-02 is considered closed.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues for the nuclear instrumentation system.

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 were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly maintenance effectiveness sample 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:

- emergent work on auxiliary building special ventilation damper and flux mapping troubleshooting activities;
- troubleshooting of a potential service water leak behind control panels in the control room; and
- EDG surveillance, service water pump testing with an offsite power line contingency.

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

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- CR 025779; battery room A/B exhaust flow low; and
- CR 364561; service water pump "B2" had no gland flow after startup.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and 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 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 two samples as defined in IP 71111.15-05.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modification 2009-05, "Installation of temporary supports to facilitate replacement of component cooling water pump 1B."

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the USAR, and the TSs, as applicable, to verify that the modification did not affect the operability or availability of the affected system. The inspectors, as applicable, performed field verifications to ensure that: the modifications were installed as directed, the modifications operated as expected, modification testing adequately demonstrated continued system operability, availability, reliability, and operation of the modifications did not impact the operability of any interfacing systems. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- testing of “B” component cooling water pump after replacement;
- testing of “B” EDG after air start hose replacement; and
- testing of “B” reactor trip breaker after replacement.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against 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 three post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- nuclear power range channel 4 (yellow) monthly test (routine);
- reactor protection system channel 2 (white) instrument test (routine);
- “B” EDG monthly availability test (routine);
- bus 1-5 under voltage relay test (routine);
- residual heat removal (RHR) pump and valve test (inservice testing sample); and
- reactor coolant system (RCS) leak rate check (RCS leak detection sample).

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 inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers 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 four routine surveillance testing samples, one inservice testing sample, and one reactor coolant system leak detection inspection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constitutes a partial sample as defined in IP 71124.01-5.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Radiation Safety Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors assessed whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last two radiological surveys from three to six selected plant areas. The inspectors evaluated if the thoroughness and frequency of the surveys was appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas, to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation.

- ultrasonic testing (UT) in north penetration room; and
- clean out and decontamination of steam generator blowdown tank.

For these work activities, the inspectors assessed the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (This evaluation may include any licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and whether the placement of monitors were representative of actual work areas. The inspectors verified that the licensee had a program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected three to five containers holding nonexempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, "Labeling containers," or met the requirements of 10 CFR 20.1905(g).

The inspectors reviewed the following radiation work permits (RWPs) used to access high radiation areas (HRAs) and evaluated the specified work control instructions or control barriers.

- RWP 10-0049, "Support And Perform UT Gas Void Monitoring To Address NRC Generic Letter 2008-1, Task 2 Operations Support For SI And RHR UT Evolutions. Including Routine Drain Down Evolutions And Required Support To Evaluate/Mitigate Any Gas Voids Found During Testing";
- RWP 10-0019, "Cleaning of SGBT Monitor/Holdup Tanks, Sump/Sludge Intercept/Waste Hold-Up Tanks, 1A/B Laundry Tanks, 1A/B Waste Condensate Tanks, Waste Area Sump And RCA Trenches And To Include Disposal Of Filters In Drumming Room If Necessary, Task 1 Clean SGBT Monitor/Holdup Tanks And Disposal Of Filters"; and
- RWP 10-0018, "NRC, INPO And/Or WANO Evaluations, Task 2 NRC Inspection/Evaluation In Higher Dose/Dose Rate Areas."

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter (EPD) alarm set points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's EPD noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the corrective action program and whether dose evaluations were conducted as appropriate. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiologically controlled area, and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures. The inspectors also reviewed whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the types of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected two to three sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact (i.e., they were not leaking their radioactive content).

The inspectors verified that any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.5 Radiological Hazar Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of EPDs in high noise areas as HRA monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or whether the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 10-0019, "Cleaning of SGBT Monitor/Holdup Tanks, Sump/Sludge Intercept/Waste Hold-Up Tanks, 1A/B Laundry Tanks, 1A/B Waste Condensate Tanks, Waste Area Sump and RCA Trenches and To Include Disposal Of Filters In Drumming Room If Necessary, Task 1 Clean SGBT Monitor/Holdup Tanks and Disposal Of Filters."

For this RWP, the inspectors evaluated airborne radioactive controls and monitoring, including potentials for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed whether barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operations for selected airborne radioactive material areas were adequate.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors inspected the posting and physical controls for selected HRAs and very high radiation areas (VHRAs), to verify conformance with the Occupational Exposure Control Effectiveness PI. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.6 Risk-Significant HRA and VHRA Controls (02.06)

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk HRAs and VHRAs. The inspectors assessed whether any changes to licensee procedures substantially reduced the effectiveness and level of worker protection.

The inspectors reviewed special areas that have the potential to become VHRAs during certain plant operations (e.g., pressurized-water reactor thimble withdrawal into the reactor cavity sump; boiling-water reactor traversing in-core probe movement; boiling-water reactor drywell fuel transfer slot area; spent fuel pool, cavity, or pit diving). The inspectors discussed these areas with first-line health physics (HP) supervisors (or equivalent positions having backshift HP oversight authority) to assess whether the communication beforehand with the HP group would allow for corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization. The inspectors evaluated licensee controls for VHRAs, and areas with the potential to become a VHRA, and ensured that an individual was not able to gain unauthorized access to the VHRA. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the significant radiological conditions in their workplace and the RWP controls/limits in place and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed a maximum of 10 radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technician with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed a maximum of 10 radiological problem reports since the last inspection that found the cause of the event to be a radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee corrective action program. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to the plant. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index (MSPI) - Emergency Alternating Current Power Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Emergency Alternating Current Power Systems performance indicator for the first quarter through the fourth quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, condition reports, event reports, and NRC integrated inspection reports, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's condition report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI emergency alternating current power system sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 MSPI - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - High Pressure Injection Systems performance indicator for the first quarter through the fourth quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, were used. The inspectors reviewed the licensee’s operator narrative logs, condition reports, MSPI derivation reports, event reports, and NRC integrated inspection reports for the period of January 2009 through December 2009, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s condition report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI high pressure injection system sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 MSPI - Residual Heat Removal Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - RHR Systems performance indicator for the first quarter through the fourth quarter 2009. To determine the accuracy of the PI data, definitions and guidance contained in the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, were used. The inspectors reviewed the licensee’s operator narrative logs, condition reports, MSPI derivation reports, event reports and NRC integrated inspection reports, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s condition report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI residual heat removal system sample as defined in IP 71151-05.

b. <u>Findings</u>

No findings of significance were identified.

4OA2 <u>Identification and Resolution of Problems</u> (71152)

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

.1 <u>Routine Review of Items Entered into the Corrective Action Program</u>

a. <u>Inspection Scope</u>

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

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

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Daily Corrective Action Program Reviews</u>

a. <u>Inspection Scope</u>

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 as required by procedure and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

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

.1 (Closed) Licensee Event Report (LER) 05000305/2009-003-00: Containment Spray Pump 1A Inoperable At Degraded Voltage Protection Setpoint

This LER describes an engineering evaluation that determined the degraded voltage protection relay for containment spray pump 1A could trip under conditions where the pump was required to be operable. This issue was identified by the licensee during the completion of corrective actions for NCV 05000305/2007006-02, “No Motor Starting Analysis for Offsite Power Supply.” Specifically, during the 2007 Component Design Bases Inspection, the inspectors were concerned that under postulated conditions (i.e., degraded grid voltage coincident with a loss of coolant accident), the simultaneous starting of the containment spray pump motor with other motors could cause stalling and tripping of the motor.

As an immediate corrective action, the licensee adjusted the containment spray pump motor protective relay settings to provide adequate protection. The inspectors reviewed the corrective actions and did not identify any issues or other violations of NRC requirements. The inspectors also determined that the issues previously described in NCV 05000305/2007006-02 remained of very low safety significance per Appendix H “Containment Integrity Significance Determination Process,” of IMC 0609, because the containment spray pumps impact late containment failure and source term, but do not impact the large early release frequency; and the pumps were not considered in the Level 1 internal events analysis.

As discussed in inspection report 05000305/2009004, a URI was previously opened pending further review of the conditions described in LER 2009-003-00, related specifically to the potential impacts of the identified condition on the safety system functional failure performance indicator. This aspect will be reviewed and tracked per the resolution of URI 05000305/2009004-04. Documents reviewed 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.

.2 (Closed) LER 05000305/2009-006-00: Protection Instruments Not Calibrated to Individual Technical Specification Setpoint Limits

On April 16, 2009, the licensee identified during review of an instrument calibration procedure that the prescribed method for calibrating low steam line pressure safety injection (SI) lead/lag circuitry units was inadequate to ensure that TS setpoint requirements were met. The calibration methodology used a vendor-supplied calibration graph of a composite output from the lead-lag circuit module to determine acceptance criteria. However, Kewaunee Power Station TSs specified individual setting limits for the lead and lag time constant. Consequently, all six channels for the Low Steam Pressure Line SI Signal and the Overtemperature Delta T(OTΔT) instruments were declared inoperable.

The instrument trip setpoints of the steam line pressure and OTΔT instruments themselves were not affected by this condition and had been properly calibrated to TS requirements. Only the anticipatory trip associated with these instruments (controlled by the lead/lag time constants), which initiates a trip prior to the setpoint being reached was affected.

The licensee implemented immediate corrective actions that included developing new acceptance curves per calculations C11874, "Determination of Ramp Acceptance Curves for Steam Pressure Lead/Lag Dynamic Box Calibration," and C11875, "Determination of Ramp Acceptance Curves for OTΔT Lead/Lag Dynamic Box Calibration." The new acceptance criteria were put into the calibration procedures for the affected instruments. The instruments were successfully calibrated using the revised procedures and acceptance criteria and all channels were returned to service.

The inspectors determined that the failure to ensure that the calibration of the low steam line pressure and OTΔT lead/lag instrumentation was a violation of TS 3.5 identified by the licensee. The violation was of very low safety significance (Green) and met the criteria of the NRC Enforcement Policy for being dispositioned as a licensee-identified NCV (see Section 4OA7 of this report).

The inspectors reviewed the corrective actions and identified that the output voltage values for the negative ramp curve in the calibration procedures SP-06-034B-1, -2, -3, -4, "Steam Generator Flow Mismatch and Steam Pressure Instrument Channel 1, -2, -3, -4 Calibration," did not match the values in calculation C11874. The inspectors determined that this performance deficiency was a violation of 10 CFR 50, Appendix B, Criterion V "Instructions, Procedures, and Drawings." Upon discovery, the licensee verified that the last performed surveillance to set the Foxboro lead/lag boxes in question met the TSs. The licensee also demonstrated that the conservatism built in the calculation was adequate to show that the mismatched curves remained conservative with respect to the TS values.

The licensee entered this issue into the CAP as CR 367826 and CR 367932. Documents reviewed 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.

a. Findings

(1) Incorrect Curve Was Incorporated into Calibration Surveillance Procedures

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 incorporate the correct negative ramp curve into calibration surveillance procedure SP-06-034B-1, "Steam Generator Flow Mismatch and Steam Pressure Instrument Channel 1." The curve was required to ensure that the low steam line pressure SI lag circuitry unit did not exceed the TS setpoint value. The wrong curves were also included in the calibration surveillance procedures for Channels 2, 3, and 4.

Description: On April 16, 2009, Kewaunee Power Station staff identified that the procedure for calibrating low steam line pressure SI lead/lag circuitry units was not adequate to ensure that TS setpoint requirements were met. The licensee reported this condition in LER 2009-006. Specifically, the licensee identified that the original calibration methodology used a vendor-supplied calibration graph of a composite output from the lead and lag circuit module to determine acceptance criteria. However, Kewaunee's TS Table 3.5-1, Item 4, specified individual setting limits for the lead and lag time constants for the low steam pressure input to the engineered safety features initiation instrument.

Upon discovery, the licensee determined that the instrument trip setpoints of the steam line pressure itself, of ≥ 500 pounds per square inch gauge, was not affected by the condition and had been properly calibrated to TS requirements. Only the anticipatory trip associated with the instrument (controlled by the lead/lag time constants), which initiated a trip prior to the setpoint being reached, was affected. The purpose of lead/lag circuitry is to provide an anticipatory trip for the parameter being monitored, in advance of the setpoint actually being reached.

As corrective action, the licensee developed a new methodology for calibrating the steam pressure instrument lead/lag circuitry units that ensured compliance with TS setting limits. New acceptance criteria were generated by this methodology as described in Calculation C-11874, "Determination of Ramp Acceptance Curves for Steam Pressure Lead/Lag Dynamic Box Calibrations." The licensee then incorporated the new acceptance criteria into procedures SP-06-34B-1, -2, -3, and 4.

During a routine review of the corrective actions associated with LER 2009-006, the inspectors verified that the output voltage values for the positive ramp calibration curve specified in the procedures matched the values in calculation C-11874. However, the inspectors identified that the output voltage values for the negative ramp calibration curve in the procedures did not match the values in calculation C-11874.

Based on the inspectors finding, the licensee performed a multidepartment review, included licensing, engineering, maintenance, and procedures group and verified that the existing plant setting for all six Foxboro lead/lag boxes were in compliance with the TS values. The licensee also placed the affected procedures SP-06-034B-1, -2, -3 and -4 on administrative hold to prohibit their use until the issue was resolved. Subsequently, during the review of documentation related to the affected procedures, the licensee identified that a procedure writer had identified in October of 2009 that the numbers in the procedures did not match the calculation, but failed to notify the appropriate organization.

The licensee determined the cause of the incorrect values being used in the procedure was due to that preliminary calculation results were given to the procedures group so that the revision of Steam Pressure lead/lag calibration procedures could be performed in parallel with the approval of the calculation.

The licensee entered this issue into their corrective action program as CR 367826, "NRC Questioned Corrective Actions in LER 2009-006," and CR 367932, "Supplemental Upgrade Writer Failed to Initiate a CR or Communicate Issues."

<u>Analysis</u>: The inspectors determined that the licensee's failure to ensure that the appropriate curve was incorporated into surveillance procedures was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V, and was a performance

deficiency. Specifically, the licensee failed to incorporate the correct negative ramp curve as specified in calculation number C-11874 into surveillance procedures SP-06-034B-1, -2, -3 and -4.

The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, because it was associated with the Mitigating System Cornerstone attribute of procedure quality and adversely affected the cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to ensure that the negative ramp curve, used to calibrate the low steam line pressure SI Lag circuitry units, could not exceed the TS value of less than or equal to 2 seconds.

The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Tables 3b and 4a for the Mitigating Systems Cornerstone, dated January 10, 2008. The inspectors determined that the finding was of very low safety-significance (Green) because the finding did not involve a design or qualification deficiency, there was no actual loss of safety function, no single train loss of safety function for greater than the TS allowed outage time occurred, and there was no risk due to external events. Specifically, upon the inspectors' discovery of the issue, the licensee was able to demonstrate operability and adequate margin existed to ensure that wrong curve would not exceed the TS value.

The inspectors determined that cause of this finding was related to the work practices, procedural compliance aspect of the human performance cross-cutting area because the licensee failed to ensure that the calculation was approved prior to the issuance of the procedure (H.4(b)).

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality 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 this, on April 17, 2009, the licensee inappropriately revised Surveillance Procedures SP-06-034B-1, -2, -3, and -4 and, failed to include the adequate curves. Specifically, the licensee failed to incorporate the correct negative ramp curve as specified in calculation number C-11874 into surveillance procedures SP-06-034B-1, -2, -3, and -4. The curves were required to ensure that the TS value would not be exceeded. Because this violation was of a very low safety-significance and because it was entered into the licensee's corrective action program as CR 367826 and CR 367932, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000305/2010002-01; Incorrect Curve Was Incorporated into Calibration Surveillance Procedure)

The licensee conducted an apparent cause evaluation and corrective actions were in progress at the conclusion of the inspection period.

.3 (Closed) LER 05000305/2009-008-00: Inadequately Controlled Reactor Coolant System Dilution Results in Violation of Technical Specification

On October 10, 2009, the licensee violated the TSs when operators loaded fuel into the core with boron concentration levels below the minimum concentration specified in the Core Operating Limits Report (COLR). The inspectors reviewed the licensee's evaluation of this TS violation and actions taken. Documents reviewed 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.

a. Findings

(1) Fuel Loading Occurs With Boron Concentration Below Required Minimum

Introduction: A finding of very low safety significance and associated NCV of TS 3.8.a.5 was self-revealed when the licensee loaded fuel into the reactor with RCS boron sample results less than the minimum boron concentration specified in the COLR.

Description: On October 10, 2009, the licensee was in a refueling outage making preparations to reload fuel into the core. The licensee needed to raise the reactor cavity water level prior to moving fuel, so the operators calculated a water addition to the reactor cavity that raised water level but also lowered reactor cavity boron concentration and they also managed boric acid tank levels in preparation for subsequent outage activities. The licensee's calculation produced a final reactor cavity boron concentration above 2500 parts per million (ppm), but did not consider that areas of lower concentration may exist if an appropriate amount of mixing time was not allocated. The Blended make-ups (of water and boric acid) occurred between approximately 5:12 a.m. and 7:30 a.m. with shift turnover occurring around 6:00 a.m. on October 10.

In procedure OP-KW-NCL-FH-003, "Pre-Refueling Checklist," RCS boron concentration was verified greater than the COLR, prior to fuel movement, using a reactor cavity and RHR system boron sample. The procedure requires the samples be obtained "prior to initial fuel movement." The sample results used to verify the reactor cavity boron concentration were obtained at 7:49 a.m.; however, the sample results used to verify the RHR system boron concentration were taken prior to the dilutions at 4:45 a.m. The pre-refueling checklist was completed and the shift manager gave permission to move fuel at 7:54 a.m. One minute later, at 7:55 a.m., the chemistry technician obtained RHR boron sample results of 2310 ppm and notified the control room. The shift manager attributed the low results to the large dilution and the limited amount of mixing time. The shift manager did not consider the sample a representative RCS sample and directed chemistry to resample while fuel movement proceeded. The next RHR boron sample result was obtained at 8:25 a.m. with a result of 2323 ppm. Shortly after the results were obtained the control room operators incorrectly concluded that the sample point for the 8:25 sample, in the "A" train of RHR, was located in an isolated portion of the system. The shift manager again did not consider the sample a representative RCS sample and directed chemistry to sample from the "B" RHR train while fuel movement continued. It was later determined that the previous shift had restored flow through the portion of the RHR system in question and failed to include that information in the shift turnover. The chemistry technician obtained "B" train RHR boron results of 2348 ppm at 9:00 a.m. and notified the control room; however, the communication was not relayed to the shift manager and fuel movement continued. The chemistry technician sampled again at 9:38 a.m. and obtained "B" train RHR boron result of 2375 ppm. The technician notified the control room, however, this information was also not relayed to the shift manager and fuel movement continued. The chemistry technician sampled again at 10:04 a.m. and obtained "B" train RHR boron result of 2392 ppm. The technician notified the control room and at 10:38 a.m. the shift manager determined that the boron concentration was outside of the requirements of the TSs and stopped fuel movement.

The licensee conducted a 675-gallon boration which raised the RCS boron concentration to 2773 ppm. Following the acceptable sample results, the shift manager authorized fuel movement which continued through the next two shifts until the core reload was complete on October 11 at 3:32 p.m.

The licensee conducted an apparent cause evaluation for the violation of TS and long-term corrective actions were in-progress at the conclusion of the inspection period.

<u>Analysis</u>: The inspectors determined that loading fuel into the reactor with boron concentrations less than the minimum boron concentration as specified in the COLR was contrary to TSs and was a performance deficiency.

This finding is more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, because it was associated with the Initiating Events Cornerstone attribute of human 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 operations. Specifically, the licensee continued to move fuel with boron concentrations below the COLR.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations SDP," dated February 28, 2005. The inspectors used Checklist 4 contained in Attachment 1 and determined that the finding did not meet the reactivity guidelines because TS 3.8.a.5 was not being met. The inspectors then reviewed the list of findings requiring a phase 2 or phase 3 analysis and determined the finding was not similar to any of the examples listed. Upon further review with the Region III senior reactor analyst (SRA), the inspectors determined that actual boron concentrations were sufficient to ensure that adequate shutdown margin was maintained during fuel movement to preclude criticality. Therefore, the issue did not need a quantitative assessment and screened as Green.

This finding has a cross-cutting aspect in the area of human performance, decision-making, because the licensee failed to use conservative assumptions when making decisions and did not demonstrate that nuclear safety was an overriding priority. Specifically, licensed operators moved and continued to move fuel after receiving multiple boron sample results that were below the minimum COLR limit (H.1(b)).

<u>Enforcement</u>: Technical Specification 3.8.a.5 states, in part, "When there is fuel in the reactor, a minimum boron concentration as specified in the COLR shall be maintained in the RCS during reactor vessel head removal or while loading or unloading fuel from the reactor." The minimum boron concentration specified in the COLR is 2500 ppm.

Contrary to this, on October 10, 2009, the licensee loaded six fuel assemblies into the core with RCS boron concentration less than the minimum as specified in the COLR. Specifically, the licensee loaded fuel from 7:54 a.m. to 10:38 a.m. with boron concentrations in the reactor coolant system below 2500 ppm, the COLR minimum boron limit. Once the licensee realized that the boron concentration was below the required minimum, the operators stopped moving fuel until the boron concentration was restored to acceptable limits. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program (as CR 351923), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000305/2010002-02; Fuel Loading Occurs With Boron Concentration Below Required Minimum).

The licensee conducted an apparent cause evaluation and proposed long-term corrective actions included procedure enhancements, operator training on the event, and conservative decision making training.

.4 <u>(Closed) LER 05000305/2009-009-00: Automatic Start of Emergency Diesel Generator due to Safeguards Bus Power Supply Transformer Trip</u>

On October 15, 2009, with the reactor in cold shutdown mode, power was lost to safeguards bus 5. This resulted in automatic actuation of EDG "A" to re-energize the bus. The power loss was caused by a trip and lockout of the tertiary auxiliary transformer (TAT), which supplies the bus, due to an incorrectly set transformer relay input parameter. The inspectors reviewed the licensee's evaluation of this TS violation and actions taken. Documents reviewed are listed in the Attachment to this report. This LER was incorrectly closed in the 2009005 report and is therefore being addressed in this inspection report. This LER was inspected in this current inspection period and is closed.

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

a. <u>Findings</u>

(1) <u>Incorrect Settings on Differential Relay Results in Loss of Tertiary Auxiliary Transformer (TAT)</u>

<u>Introduction</u>: A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was self-revealed for the failure to establish adequate measures to identify and control design interfaces and coordinate among participating design organizations. Specifically, the licensee failed to adequately control all required TAT relay inputs/settings that interfaced with the existing plant design, which adversely impacted associated equipment and caused an unanticipated system response.

<u>Description</u>: During the fall 2009 outage, the licensee replaced the TAT as part of its ongoing switchyard upgrade project under design change request 3632-1, "TAT Replacement." On October 15, 2009, after the TAT installation was completed, the operators accepted the TAT as operable and it was credited as the TS required power source for the 4160-volt safeguards bus 5. At 12:04 p.m., on that same day, a TAT lockout occurred after differential relay 87-2/TAT1 actuated when SI pump "A" was

started. The TAT lockout caused a momentary loss of bus 5 and subsequent loss of the "A" train of RHR until the "A" EDG automatically started and restored bus 5.

At the time of the TAT lockout, the reserve auxiliary transformer (RAT) was danger tagged out-of-service in preparation for maintenance, and bus 6 was being supplied by the main auxiliary transformer. Kewaunee TSs allow systems, trains, or components with inoperable normal or emergency power supplies to be considered operable, if the corresponding normal or emergency power supply is operable and the redundant system, train, or component is operable. The two normal off site power sources for the 4160-volt safeguards buses are described in the TS basis as the RAT and the TAT. Prior to the event, both RHR trains were considered operable because both emergency diesels and the TAT were operable. Once the TAT lockout occurred neither of the normal off site power sources were operable and the licensee declared both trains of RHR inoperable. With the main auxiliary transformer still supplying power to bus 6, the "B" RHR train was available and the licensee started the 'B' RHR pump shortly after the TAT lockout occurred and restored shutdown cooling. The licensee cleared tags and restored the RAT to operable status, and also stopped and realigned the "A" EDG for auto start. Once the licensee restored the RAT and it was supplying power to bus 5, and "A" EDG was operable, both RHR pumps were declared operable.

The licensee determined that the input parameters for the TAT programmable differential relay were entered incorrectly. Specifically, the phase angle readings were found to be at 120 degrees instead of the required 180 degrees. Further investigation revealed that the specific input parameters in question were not included in the written instructions given to the qualified relay technician for initial relay parameter entry. When the technician initially entered the calculated setpoints and input parameters he believed he had all the information necessary to properly program the relay. The technician had entered the same input parameters that the last four relays required. The incorrect relay inputs caused the relay trip setting to be lower than expected and when the "A" SI pump was started, the additional load on bus 5, caused actual conditions to reach the lower threshold resulting in the TAT lockout.

The licensee's root cause evaluation discovered that the design change documentation failed to provide adequate details for the relay technician to enter the correct input parameters. It was determined that members of the licensee's staff believed that the contractor hired to assist in the creation of the design change would provide all necessary setpoints and inputs; however, the contractor believed the licensee would be proving inputs because the inputs in question were based on the physical TAT characteristics and not on a calculation preformed by the contractor.

The licensee performed a root cause evaluation and implemented corrective actions, some of which included: modifying the design change process to ensure that all programmable digital device setpoints and inputs were identified and the basis for each was described in the design change documentation, and providing programmable digital device training for design engineering and maintenance personnel.

<u>Analysis</u>: The inspectors determined that the failure to establish adequate measures to identify and control design interfaces and coordinate among participating design organizations was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency.

The finding is more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, because the finding was associated with the Initiating Events Cornerstone attribute of design control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to adequately control all required TAT relay inputs/settings adversely impacted the associated equipment, caused an unanticipated system response, and challenged core shutdown cooling.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations SDP," dated February 28, 2005. The inspectors used Checklist 4 contained in Attachment 1 and determined that the finding required a phase 2 analysis because it degraded the ability to recover the decay heat removal system.

The Region III SRA performed the assessment using Appendix G, Attachment 2, "Phase 2 Significance Determination Process Template for PWR during Shutdown." The SRA determined this to be a loss of off site power precursor. The plant operating state (POS) was "POS 2" (RCS vented). The initiating event likelihood using Table 2, "Initiating Event Likelihood for LOOP (loss of offsite power) Precursors," was assumed to be "1," since only a partial LOOP occurred. The main transformer supplied safety Bus 6 continuously throughout this event. The time window was "late," since this event occurred after refueling.

Using Worksheet 4 contained in Attachment 2, a credit of "3" was given for emergency AC power, since both trains of emergency power were available. A credit of "2" was given for recovery of offsite power given successful gravity feed. A credit of "1" was given for recovery of offsite power given unsuccessful gravity feed. This Phase 2 result was determined to be White with a dominant sequence being loss of offsite power, loss of emergency power, and failure to recover offsite power given successful gravity feed.

The SRA determined that this result was overly conservative since operators successfully started the opposite train RHR pump from another offsite power source, the station black out diesel was available, and SI pump B was available. At the time of the event, the reactor had been shut down for over 19 days and time to boil was about 2.5 hours. Considering this information, the SRA performed a Phase 3 analysis with additional credit for this equipment and associated operator actions. This resulted in a finding of very low safety significance (Green).

This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee did not maintain complete, accurate and up-to-date design documentation. Specifically, the detailed requirements for programmable digital device input parameters were not specified in a process or procedure (H.2(c)).

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the identification and control of design interfaces and for coordination and control of design interfaces and for coordination among participating design organizations.

Contrary to this, on October 15, 2009, the licensee failed to establish measures for the identification and control of design interfaces and for coordination among participating design organizations. Specifically, the licensee failed to adequately control all required TAT relay inputs/settings that interfaced with the existing plant design, adversely impacting the associated equipment and causing an unanticipated system response. Because this violation was of very low safety significance and was entered into the licensee's corrective action program (as CR 352878), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000305/2010002-03; Incorrect Settings on Differential Relay Results in Loss of Tertiary Auxiliary Transformer).

The licensee performed a root cause evaluation and implemented corrective actions, some of which included: modifying the design change process to ensure that all programmable digital device setpoints and inputs were identified; describing the basis for each setpoint or input in the design change documentation; and providing programmable digital device training for design engineering and maintenance personnel.

4OA5 Other Activities

.1 (Closed) URI 5000305/2008005–01: "Licensee Response to Operator's Violation of NRC Requirements"

The inspectors reviewed the circumstances surrounding the facility licensee's response to finding an operator actively performing the functions of a licensed operator in the control room while under the influence of alcohol. It was determined that the licensee had complied with applicable regulations in responding to this situation.

See Section 1R11, "Licensed Operator Requalification Program," for details. This URI is closed. Documents reviewed are listed in the Attachment to this report.

.2 (Closed) URI 05000305/2009005-02: "Licensed Operator Requalification Examination Security Issues"

The inspectors reviewed the circumstances surrounding the discovery of an uncontrolled removable media storage device containing licensed operator requalification testing material, and the subsequent discovery of uncontrolled simulator files containing licensed operator requalification program testing material. It was determined that a violation of NRC requirements had occurred. A licensee-identified violation was documented. See Section 4OA7 for details of the violation. This URI is closed. Documents reviewed are listed in the Attachment to this report.

.3 (Closed) URI 05000305/2009002-02: "Inappropriate Application of a Dedicated Operator During a CCW Surveillance"

The inspectors evaluated the licensee's historical review of out-of-service times for systems that input into both the maintenance rule and performance indicators. The licensee found instances where it needed to correct its mitigating systems performance indicator submittals; however, the changes did not result in an indicator crossing a threshold. The inspectors verified that the changes to the MSPI were completed and correct.

This URI is closed. Documents reviewed are listed in the Attachment to this report.

.4 (Closed) URI 05000305/2009005-04: "Residual Heat Removal Pipe Support RRHR-H2: Seismic Category I Requirements"

The licensee provided the inspectors a more detailed analysis that determined that pipe support standard component hardware for pipe support RRHR-H2 conformed to seismic category I design basis requirements. The inspectors determined that no performance deficiency existed.

This URI is closed. Documents reviewed are listed in the Attachment to this report.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 1, 2010, the inspectors presented the inspection results to Mr. S. 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:

- The results of an inspection of the facility's response to a licensed operator found with a high blood alcohol content in the control room with D. Laing, Kewaunee Power Station Training Manager, on March 9, 2010;
- The results of an inspection of the facility's licensed operator examination security program and licensee-identified violation with D. Laing, on March 9, 2010; and
- The results of Radiological Hazard Assessment and Exposure Controls inspection with the Site Vice-President, Mr. S. Scace, on February 12, 2010.

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.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) or Severity Level IV were identified by the licensee and are violations of NRC requirements that meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations.

.1 Annual Licensed Operator Requalification Examination Material Was Administered to Licensed Operators After the Examination Material Was Uncontrolled.

Part 10 CFR 55.49, "Integrity of Examinations and Tests," states, in part, that no one shall engage in any activity that compromises the integrity of any test or examination required by 10 CFR Part 55. The integrity of the test or examination is considered compromised if any activity, regardless of intent, affected, or, but for detection, would have affected the equitable and consistent administration of the test or examination.

Contrary to this, on November 10, 2009, licensed operators were administered an annual operating test required by 10 CFR Part 55 containing job performance measure (JPM) test material that was later discovered to be uncontrolled on the control room simulator computer. The JPM test material was found in some simulator log files that were automatically generated by the simulator computer during validation of the JPMs by station trainers. Because the JPM test material was used to examine licensed operators and had to be replaced with new material, a compromise of operating test material existed. This example of test material compromise was documented in Condition Report 357397. The licensee took immediate action to delete the simulator log files and re-tested the operators with new test material prior to the operators returning to control room duties. Long-term compensatory actions included modifying the simulator software to delete the log files and revising a simulator shutdown checklist to ensure any log files created by the simulator during examination validation were deleted.

.2 <u>Inadequate Calibration Procedure Leads To Violation of Technical Specifications</u>

Technical Specification 3.5, "Instrumentation System," states, in part, if the number of channels for the low steam pressure/line or the OTΔT subsystem falls below the minimum required, a shutdown is required, as soon as practicable.

Contrary to this, on April 16, 2009, the licensee; during review of an instrument calibration procedure; identified that the procedure for calibrating low steam line pressure SI lead/lag circuitry units was inadequate to ensure that TS setpoint requirements were met. The calibration methodology, in place since initial plant operation, used a vendor supplied calibration graph of a composite output from the lead/lag circuit module to determine acceptance criteria. However, Kewaunee Power Station TSs specified individual setting limits for the lead and lag time constant and it was determined that the composite output was incorrect and did not meet the individual settings. Consequently, all six channels for the Low Steam Pressure Line SI Signal were declared inoperable and the minimum operable channels requirement was no longer met. Therefore, the licensee commenced a shutdown and achieved a Hot Shutdown condition at 11:56 p.m.

The inspectors determined that failure to ensure the proper calibration of the low steam line pressure and OTΔT lead/lag instrumentation, from original plant operation until discovery, was a licensee-identified violation of TS 3.5. The instrument trip setpoints of the steam line pressure and OTΔT instruments themselves were not affected by this condition and had been properly calibrated to TS requirements. Only the anticipatory trip associated with these instruments (controlled by the lead/lag time constants), which initiates a trip prior to the setpoint being reached was affected. The licensee entered the issue into its corrective action program as CR 331174.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Scace, Site Vice-President
D. Laing, Training Manager
D. Emery, Supervisor, Initial License Training
A. Fahrenkrug, Senior Instructor
D. Shannon, Radiation Protection General Supervisor
M. Wilson, Licensing Director
T. Breene, Licensing Manager
S. Yuen, Engineering Director
J. Stafford, Organizational Effectiveness Manager
D. Lawrence, Operations Manager
T. Evans, Maintenance Manager
C. Chovan, Outage and Planning Manager
J. Gadzala, Licensing
R. Giuliani, Nuclear Oversight Manager

Nuclear Regulatory Commission

M. Kunowski, Chief, Division of Reactor Projects, Branch 5
P. Tam, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000305/2010002-01	NCV	Incorrect Curve Was Incorporated into Calibration Surveillance Procedures (Section 4OA3.2)
05000305/2010002-02	NCV	Fuel Loading Occurs With Boron Concentration Below Required Minimum (Section 4OA3.3)
05000305/2010002-03	NCV	Incorrect Settings on Differential Relay Results in Loss of Tertiary Auxiliary Transformer (Section 4OA3.4)

Closed

05000305/2009-003-00	LER	Containment Spray Pump “A” Inoperable At Degraded Voltage Protection Setpoint (Section 4OA3.1)
05000305/2009-006-00	LER	Protection Instruments Not Calibrated to Individual Technical Specification Setpoint Limits (Section 4OA3.2)
05000305/2010002-01	NCV	Wrong Curve Was Incorporated into Calibration Surveillance Procedures (Section 4OA3.2)
05000305/2009-008-00	LER	Inadequately Controlled Reactor Coolant System Dilution Results in Violation of Technical Specification (Section 4OA3.3)

05000305/2010002-02	NCV	Fuel Loading Occurs With Boron Concentration Below Required Minimum (Section 4OA3.3)
05000305/2009-009-00	LER	Automatic Start of Emergency Diesel Generator due to Safeguards Bus Power Supply Transformer Trip (Section 4OA3.4)
05000305/2010002-03	NCV	Incorrect Settings on Differential Relay Results in Loss of Tertiary Auxiliary Transformer (Section 4OA3.4)
05000305/2008005-01	URI	Licensee Response to Operator’s Violation of NRC Requirements (Sections 1R11 and 4OA5.1)
05000305/2009005-02	URI	Licensed Operator Requalification Examination Security Issues (Sections 1R11 and 4OA5.2)
05000305/2009002-02	URI	Inappropriate Application of a Dedicated Operator During a CCW Surveillance (Section 4OA5.3)
05000305/2009005-04	URI	Residual Heat Removal Pipe Support RRHR-H2: Seismic Category I Requirements (Section 4OA5.4)

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>1R01 Adverse Weather Protection</u>

- CR 336372; Leakage Into Emergency Diesel Generator "B" Room Has Changed
- CR 348081; Possible Insufficient Seal On Seiche Door 182
- CR 348085; Doors 164, 165, And 182 Were Observed To Not Have Door Seals That Maintain A Line Of Contact To The Door
- CR 348087; Doors 164, 165, And 182 Were Observed To Not Have Door Seals That Maintain A Line Of Contact To The Door
- CR 350088; Barrier Impairment Permit Not Posted With Barrier Breached
- CR 353332; Work Evolutions Requiring Evaluations For Possible Seiche Concerns
- CR 353421; Future Plant Modification; Flood Barriers Needed In Screenhouse
- CR 361156; Clean Out Plug In Screenhouse Floor Lifting From Service Water Pumps Backwash
- eSoms Station Narrative Log Data; September 30, 2009
- KW 100452499; DCR 3699 – Replace Service Water Pump 1B1
- OP-KW-AOP-GEN-004; Response To Natural Events; Revision 8
- OP-KW-AOP-GEN-005; Barrier Control; Revision 2
- RAS 000105; Doors 164, 165, And 182 Were Observed To Not Have Door Seals That Maintain A Line Of Contact To The Door

<u>1R04 Equipment Alignment</u>

- CR 365243; NRC Identified Drawing Discrepancy
- CR 373623; NRC Identifies Two Valves With The Same Label During System Walkdown
- N-CC-31-CL; Component Cooling system Prestartup Checklist; Revision 29
- N-FW-05B-CL; Auxiliary Feedwater System Prestartup Checklist; Revision 44
- OP-KW-NCL-DGM-001B; Diesel Generator "B" Prestartup Checklist; Revision 3
- Action Number 37211; Main Data Entry Form for procedure OP-KW-NCL-001B
- Drawing E-2311; Schematic Diagram Fuse Panel RR-176 DC Safeguard 6; Revision Q
- Drawing APXK-100-19; Analytical Part Flow Component Cooling System; Revision L
- Drawing OPERM-205; Flow Diagram-Feedwater System; Revision BF
- Drawing OPERM-213-9; Flow Diagram Diesel Generator Startup Air Compressor A & B; Revision F
- Drawing OPERM-220; Flow Diagram Diesel Generator Fuel Oil; Revision AR

<u>1R05 Fire Protection</u>

- CR 113288; Fire Pump A "Power Available" Light Not Lit At Local Control Station In The Screen House
- CR 339893; Door 1 Damaged; Top Half of Astragal Forced From Door
- CR 366588; Received Fireworks Alarm, Common Trouble Activation
- PMP-41-06; LT-Big Beam Emergency Light Common Train Electrical Maintenance - Appendix "R" And Non Appendix "R"; Revision 21

- Drawing A-520; Fire Zone Boundaries; Mezzanine Floor Elevation 606' 0"; Fig. 4.5-4
- Drawing A-521; Fire Zone Boundaries; Elevation 616' 0"; Fig. 4.5-5;
- Drawing A-522; Fire Zone Boundaries; Operating Floor Elevation 626' 0"; Fig. 4.5-6
- Fire Plan for Fire Zones SC-70A, SC-70B, SC-70C, Screen House
- Fire Protection Program Analysis; AX-32 Service Rooms; Revision 8
- Fire Protection Program Analysis; SC-70A Screenhouse North; Revision 8
- Fire Protection Program Analysis; SC-70B Screenhouse South; Revision 8
- Fire Protection Plan Drawing; PFP-6: TU-92, TU-93, "B" Diesel Generator and Day Tank Rooms; Revision 11/17/04
- Fire Protection Plan Drawing; PFP-9: TU-95B, TU-95C, 480V Switchgear, Bus 1-61, 1-62 and AFW Pump Area; Revision 7

1R06 Flooding

- Auxiliary Building Internal Flood Evaluation; Revision 0
- CR 369685; SW Piping Lagging Wet and Slowly Dripping

1R11 Licensed Operator Requalification Program

- LRC-10-DY101; Simulator Exercise Guide; Revision B
- Accepted Differences Between Simulator And Reference Plant Data; February 25, 2010

1R12 Maintenance Effectiveness

- ER-AA-MRL-100, Implementing Maintenance Rule; Revision 1
- RCE 752; Nuclear Instrumentation Channel N-42 Maintenance Rule (a)(1) Evaluation
- SSC Performance Criteria Sheet; 48 Nuclear Instrumentation, Attachment B; Revision 4
- Licensee Maintenance Rule Data Tracking Sheets; Nuclear Instrumentation System; January 2007 – December 2009
- Maintenance Rule Scoping Questions; Nuclear Instrumentation System; January 14, 2010
- Maintenance Rule System Basis; 48 Nuclear Instrumentation System; Revision 7
- Nuclear Instrumentation System Unavailability Tracking Graph; July 2008 – December 2009

1R13 Maintenance Risk

- CR 367958; ASV-51B Open Limit Switch Is Sticking
- eSOMS Station Narrative Log; February 8, 2010
- KW 100651423; ASV-51B Open Limit Switch Is Sticking
- Major Activities Data List for Work Week 1006; Week of February 7, 2010
- Major Activities Data List for Work Week 1008; Week of February 21, 2010
- Major Activities Data List for Work Week 1012; Week of March 21, 2010
- Protected Equipment Log Data; Train B; Week of February 21, 2010
- Protected Equipment Log Data; Train A; February 12-21, 2010

1R15 Operability Evaluations

- CA 156155; Pumps – Packing Adjustments
- CR 025779; Battery Room A/B Exhaust Flow Low
- CR 092094; Service Water Pump "B1" Packing Is Dry
- CR 317521; Battery Room Exhaust Fan Low Flow Alarm Setpoint Is Too Low
- CR 323214; Battery Room "A" Air Flow Switch FS-16941 PM Not Completed

- CR 323700; Battery Room “B” Air Flow Switch FS-16942 PM Was Not Completed
- CR 362643; Service Water Pump “B2” Packing Leakage
- CR 364561; Service Water Pump “B2” Had No Gland Flow After Startup
- CR 368298; Battery Room Exhaust Flow Rates Below Design
- eSOMS Station Narrative Log; February 17, 2010
- OD 000240; BRA-106 Heat Load Not Considered For Battery Rooms During Station Blackout Event
- Calculation C10049; Battery Room Hydrogen Generation Calculation; Revision 2
- OP-KW-NOP-SW-001; Service Water System; Revision 2
- Calculation C10049; Battery Room Hydrogen Generation Calculation; Revisions 1 And 2
- 50.59 Applicability Review Of Calculation C10049; Battery Room Hydrogen Generation Calculation; Revision 2
- Control Room Log Entries Report; January 8, 2010
- Regulatory Guide 1.128; Installation Design And Installation Of Vented Lead-Acid Storage Batteries For Nuclear Power Plants
- Regulatory Guide 1.189; Fire Protection For Nuclear Power Plants

1R18 Plant Modifications

- FP-E-MOD-03(t); Temporary Modifications; Revision 2
- TMod 2009-05; Installation Of Temporary Supports To Facilitate Replacement Of Component Cooling Pump “1B”; Revision 0
- 50.59 Applicability Review of TMod 2009-05; Installation Of Temporary Supports To Facilitate Replacement Of Component Cooling Pump 1B; Revision 0
- Calculation C11887; Evaluation Of Component Cooling Piping In support Of The Replacement Of Component Cooling Pumps “1A” And “1B”; Revision 0
- Calculation C11888; Pipe Hanger Qualifications In Support Of The On-Line Replacement Of Component Cooling Pumps “1A” And “1B”; Revision 0
- 50.59 Applicability Review Of Calculation C11888; Pipe Hanger Qualifications In Support Of The On-Line Replacement Of Component Cooling Pumps “1A” And “1B”; Revision 0
- Kewaunee Power Station USAR; Table B.7-1; Load Combinations For Components; Class Of Components; Revision 21
- CR 372664 NRC Identified An Incorrect Reference in a 50.59 Screening for TMod 2009-005

1R19 Post-Maintenance Testing

- CR 372689; Small Leak From A Plug On the North Side Of The Upper Casing Of the “B” Component Cooling Water Pump
- CR 372691; Component Cooling Water “B” Pump Inboard Mechanical Seal Flange Is Leaking Approximately Six Drops Per Minute
- CR 372711; New Component Cooling Water Pump “B” Performance
- CR 373186; Maintenance And Senior Management Performed Critical Observation
- KW 100406144; Emergency Diesel Generator “1A” – Replace The Jumper Line Tubing Between The Air Start Motors (Right Bank)
- KW 100413573; Component Cooling Pump “B” – Replace With Spare Pump Assembly And New Stainless Steel Casing
- KW 100527176; PM47-010: Inspect/Clean/Test Reactor Trip Breaker
- OP-KW-OSP-CC-001B; Component Cooling Pump “B” Pre-Service Test At Power – IST; Revision 3
- OP-KW-OSP-DGE-001A; Diesel Generator “A” Monthly Availability Test; Revision 9
- SP-47-062B; Reactor Protection Logic Train “B” Test; Revision 30

1R22 Surveillance Testing

- CR 365925; Annunciator 47032-M Unexpected During SP-48-003H
- CR 365953; Overpower High Range Reset Point Found OOS High During SP-48-003H
- CR 368946; Could Not Resolve Mechanical Issues With The Keeper Assembly On Security Door 436
- CR 370504; Reactor Coolant System Leak Rate Exceeds Tier 3 Action Level
- CR 372930; Administrative Change to OSP-RCS-001
- eSOMS Station Narrative Logs; March 10-11, 2010
- MA-KW-ESP-EHV-002A; Bus 1-5 Loss Of Voltage Relay Test; Revision 6
- OP-KW-OSP-RCS-001; Reactor Coolant System Leak Rate Check; Revision 3
- OP-KW-OSP-DGE-001B; Diesel Generator “B” Monthly Availability Test; Revision 9
- SP-34-099A; Train “A” Residual Heat Removal Pump And Valve Test – IST; Revision 25
- SP-47-316BA; Channel 2 (White) Instrument Channel Test; Revision 30
- SP-48-003H; Nuclear Power Range Channel 4 (Yellow) N-44 Monthly Test; Revision 22
- 50.59 Applicability Review of SP-34-099A; Train “A” Residual Heat Removal Pump And Valve Test – IST; Revision 25
- SP-34-099B; Train “B” Residual Heat Removal Pump And Valve Test – IST; Revision 23
- Drawing OPERXK-100-35; Flow Diagram; Chemical And Volume Control System; Revision AC
- Kewaunee Nuclear Power Plant; Plant Process Computer System Safety Parameter Display System Replacement For Reactor Coolant System Leakage; Revision 54
- Kewaunee Power Station Inservice Testing Basis Valve Data Sheet; Residual Heat Removal Return To Letdown Check Valve RHR-44; Revision 6
- Kewaunee Power Station Inservice Testing Basis Valve Data Sheet; Regen HX To PRT Relief Valve LD-5; Revision 6
- Kewaunee Power Station Real-Time Reactor Coolant System Leakage Data; March 13, 2010
- NRC Generic Letter 96-06; Assurance Of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions
- NRC Information Notice 94-46; Non-Conservative Reactor Coolant System Leakage Calculation
- NRC Regulatory Issue Summary 2003-13: NRC Review Of Responses To Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation And Reactor Coolant Pressure Boundary Integrity”
- Unidentified Leak Rate (Gallons Per Minute) Versus Time; October 25, 2009 - November 22, 2009 – December 20, 2009 – January 17, 2010 – February 14, 2010
- Westinghouse WCAP-16423-NP; Pressurized Water Reactor Owners Group Standard Process And Methods For Calculating RCS Leak Rate For Pressurized Water Reactors; Revision 0
- Westinghouse WCAP-16465-NP; Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels And Response Guidelines For Pressurized Water Reactors; Revision 0

2RS1 Radiological Hazard Assessment and Exposure Controls

- CR 354524; Level 1 PCE Due to Poor Radiation Worker Practice; December 23, 2009
- CR 360078; Worker Alarmed PM-7 While Exiting Security Building; December 18, 2009
- CR 361126; Signed in on RWP12-01 and Not 12-02; December 11, 2009
- CR 328527; Mechanic Entered Containment Building Without Electronic Dosimeter; July 22, 2009
- HP-01.023; Evaluation or Radiological Risk Significant Tasks and Evolutions; Revision 5

- HP-01.024; Source Control Program; Revision 6
- HP-02.008; Evaluation of Airborne Radioactive Areas; Revision 3
- HP-02.009; TEDE ALARA Evaluation for Use of Respiratory Protection Equipment; Revision 3
- HP-06.100; Instrument Operating Procedure – SAM-11 Small Article Monitor; Revision 7
- RP-AA-111; Monitoring and Improving Radiological Performance; Revision 0
- RP-AA-201; Access Controls for High and Very High Radiation Areas; Revision 3
- RP-AA-202; Radiological Posting; Revision 2
- RP-AA-224; Airborne Radioactivity Areas; Revision 0
- RP-KW-HSP-HPE-005; Radioactive Source Inventory and Leak Testing Requirements; Revision 2
- RWP 10-0018; NRC, INPO and/or WANO Evaluations; Task 2 NRC Inspection/Evaluation In Higher Dose/Dose Rate Areas.
- RWP 10-0019; Cleaning of SGBT Monitor/Holdup Tanks, Sump/Sludge Intercept/Waste Hold-Up Tanks, 1A/B Laundry Tanks, 1A/B Waste Condensate Tanks, Waste Area Sump and RCA Trenches and To Include Disposal Of Filters In Drumming Room If Necessary; Task 1 Clean SGBT Monitor/Holdup Tanks and Disposal Of Filters
- RWP 10-0049; Support And Perform UT Gas Void Monitoring To Address NRC Generic Letter 2008-1; Task 2 Operations Support For SI and RHR UT Evolutions. Including Routine Drain Down Evolutions And Required Support To Evaluate/Mitigate Any Gas Voids Found During Testing

4OA1 Performance Indicator Verification

- MRE 010277; Bus 5 Relay 27A/B5 Found Out-Of-Tolerance
- MRE 010535; Door 003 Nonfunctional For Flood Barrier Due To Seal Tear/Degradation
- MRE 011153; Diesel Generator “B” Field Did Not Flash During OSP-DGE-004B
- MRE 010870; Flow Loop 924 Found Out of Calibration
- MRE 011592; TM-627 (489501) Out Of Specification
- CR 364204; PRA and Maintenance Rule No Longer Take Credit For Dedicated Operator
- CR 346722; Possibility For Inaccurate Data To Be Sent NRC On MSPI Components
- Kewaunee Mitigating System Performance Index Basis Document; Revision G
- Performance Indicator Data Sets, Diesel Generators; January, 2009 – December, 2009
- Performance Indicator Data Sets, Safety Injection; January, 2009 – December, 2009
- Performance Indicator Data Sets, Residual Heat Removal; January, 2009 – December, 2009
- 2009 MSPI Derivation Reports, Diesel Generator
- 2009 MSPI Derivation Reports, Safety Injection
- 2009 MSPI Derivation Reports, Residual Heat Removal

4OA2 Identification and Resolution of Problems

- CR 113288; Fire Pump A “Power Available” Light Not Lit At Local Control Station In The Screen House
- CR 365243; NRC Identified Drawing Discrepancy
- CR 373623; NRC Identifies Two Valves With The Same Label During System Walkdown
- CR 374525, Pipe Support Calculation S-061-RHR-34-001 Requires Revision
- CR 372664; NRC Identified An Incorrect Reference in a 50.59 Screening for TMod 2009-005
- CR 372930; Administrative Change to OSP-RCS-001

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

- CR 367826; NRC Questioned Corrective Actions in LER 2009-006
- CR 367932; Supplemental Upgrade Writer Failed to Initiate a CR or Communicate Issues
- CR 351923; Unexpected Residual Heat Removal Sample Results
- RCE 989; TAT Lockout
- CY-KW-020-009; Boron Titration Using The Mettler DL53 Titrator; Revision 13
- CY-KW-040-001; Primary Chemistry Sample Specifications; Revision 3
- CY-KW-060-001; HRSR Collection And Analysis; Revision 0
- eSOMS Station Narrative Log; March 18-19, 2010
- OP-KW-AOP-RC-006; Inadvertent Boron Dilution; Revision 0
- OP-KW-NCL-FH-003; Pre-Refueling Checklist; Revisions 0, 1
- SP-06-034B-1; Steam Generator Flow Mismatch and Steam Pressure Instrument Channel 1 (Red) Calibration; Revision 12
- Calculation 11874; Determination of Ramp Acceptance Curves for Steam Pressure Lead/Lag Dynamic Box Calibrations; Revision 0
- Control Room Log Entries Report; October 15, 2009
- Control Room Log Entries Report; October 10, 2009
- Outage Control Center Log Entries Report; October 10, 2009
- Drawing OPERXK-100-18; Flow Diagram – Residual Heat Removal System; Revision BB
- Mettler Toledo DL53 Titrator Boron Sample Results for October 10, 2009
- Drawing OPERXK-100-44; Flow Diagram – Sampling System; Revision AP
- 4.0 Crew Evaluation for October 10, 2009
- LER 2009-006; Protection Instruments Not Calibrated to Individual Technical Specification Setpoint Limits; Dated April 16, 2009
- LER 2009-003; Containment Spray Pump “A” Inoperable At Degraded Voltage Protection Setpoint; Dated April 1, 2009
- LER 2009-009; Automatic Start of Emergency Diesel Generator Due To Safeguards Bus Power Supply Transformer Trip; Dated December 9, 2009
- LER 2009-008; Inadequately Controlled Reactor Coolant System Dilution Results in Violation of Technical Specification; Dated December 4, 2009

4OA5 Other Activities

- CR 374525; Pipe Support Calculation S-061-RHR-34-001 Requires Revision
- CR 317247; An Operations Employee Failed To Notify Medical Of An FFD Concern; December 17, 2008
- CR 364204 PRA and Maintenance Rule No Longer Take Credit For Dedicated Operator
- 10 CFR 50.9; Completeness And Accuracy Of Information
- 10 CFR 55.21; Medical Examination
- 10 CFR 55.25; Incapacitation Because Of Disability Or Illness
- 10 CFR 50.74(c); Notification Of Change In Operator Or Senior Operator Status”
- 10 CFR 26.719; Reporting Requirements

4OA7 Licensee-Identified Violations

- CR 362739; Screen Capture Program Found Running On Simulator Plant Process Computer Work Station; December 21, 2009
- CR 357397; NRC Exam Security Issue Due To New Simulator Log Files; November 10, 2009
- ACE 017917; NRC Exam Security Issue Due To New Simulator Log Files; December 16, 2009
- CA 155526; Apparent Cause Corrective Action – Revise TR-AA-SIM-300; December 17, 2009

- CR 357354; LOR Annual Simulator Exam Was Compromised; December 8, 2009
- CA 152820; Determine/Resolve Issue Of Jump Drive Being Left In The Simulator Computer; December 8, 2009
- American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.4 - 1996, “Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants”
- Kewaunee Power Station Simulator Security Checklist 04-009; No Revision/Date
- KW-PROC-ADM-TR-AA-710; NRC Exam Security Requirements; Revision 1
- LER 2009-006; Protection Instruments not Calibrated to Individual Technical Specification Setpoint Limits; Dated April 16, 2009

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
CAP	Corrective Action Program
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CR	Condition Report
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EPD	Electronic Personal Dosimeter
HP	Health Physics
HRA	High Radiation Area
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IR	Inspection Report
JPM	Job Performance Measure
LER	Licensee Event Report
LOOP	Loss of Off-site Power
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OT∆T	Overtemperature Delta T
PARS	Publicly Available Records
PI	Performance Indicator
POS	Plant Operating State
ppm	Parts Per Million
RAT	Reserve Auxiliary Transformer
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SDP	Significance Determination Process
SI	Safety Injection
SRA	Senior Reactor Analyst
SSC	Structures, Systems, and Components
TAT	Tertiary Auxiliary Transformer
TS	Technical Specification
USAR	Updated Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Testing
VHRA	Very High Radiation Area

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Sincerely,

/RA/

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

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Letter to D. Heacock from M. Kunowski dated May 12, 2010

SUBJECT: KEWAUNEE POWER STATION INTEGRATED INSPECTION REPORT 05000305/2010002

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