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

April 26, 2011

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 05000305/2011002

Dear Mr. Heacock:

On March 31, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Kewaunee Power Station. The enclosed report documents the results of this inspection, which were discussed on April 5, 2011, 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, three NRC-identified findings and one self-revealed finding of very low safety significance were identified. Three of these findings involved violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating the findings as non-cited violations (NCVs), in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Kewaunee Power Station. In addition, if you disagree with the cross-cutting aspect assigned to 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.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000305/2011002
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-305
License No:	DPR-43
Report No:	05000305/2011002
Licensee:	Dominion Energy Kewaunee, Inc.
Facility:	Kewaunee Power Station
Location:	Kewaunee, WI
Dates:	January 1, 2011, through March 31, 2011
Inspectors:	R. Krsek, Senior Resident Inspector K. Barclay, Resident Inspector J. Jandovitz, Project Engineer R. Langstaff, Senior Reactor Inspector M. Holmberg, Reactor Inspector E. Sanchez-Santiago, Reactor Inspector J. Cassidy, Senior Health Physicist R. Winter, Reactor Inspector A. Shaikh, Reactor Inspector R. Jones, Reactor Inspector
Approved by:	Michael A. Kunowski, Chief Branch 5 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000305/2011002, 1/01/2011 – 3/31/2011; Kewaunee Power Station; Inservice Inspection Activities, Maintenance Risk Assessments and Emergent Work Control, Outage Activities, and 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. The inspectors identified three Green findings and one Green finding was self-revealed. Three of the findings were considered non-cited violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. <u>NRC-Identified and Self-Revealed Findings</u>

Cornerstone: Initiating Events

- <u>Green</u>. A finding of very low safety significance and associated non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," was identified by the inspectors on March 3, 2011, for the licensee's failure to establish a procedure that incorporated the American Society of Mechanical Engineers Code acceptance criteria for evaluation of flaws detected during ultrasonic examinations. Consequently, the licensee applied incorrect acceptance criteria to the flaws identified during ultrasonic examination of a weld on the chemical and volume control system seal water injection filter 1A housing. Licensee corrective actions included: evaluation of weld flaws to ensure they met applicable Code criteria and revision of a site procedure to incorporate appropriate Code acceptance criteria.

  The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Absent NRC identification, the failure to provide Code acceptance criteria could have allowed components with unacceptable cracks to be returned to service. Cracks in components returned to service would place safety-related piping systems at increased risk for through-wall leakage and/or failure. The licensee promptly corrected this issue before components with unacceptable flaws were returned to service. The inspectors answered "No" to the Significance Determination Process Phase I screening question, "Assuming worst case degradation, would the finding result in exceeding the Technical Specification (TS) limit for any reactor coolant system leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function assuming the worst case degradation?" Therefore, this finding screened as having very low safety significance (Green). This finding has a cross-cutting aspect in the area of human performance, work practices, because the licensee did not effectively implement human error prevention techniques. Specifically, the lack of procedure acceptance criteria was caused by inadequate peer checking during the licensee's review and approval of the procedure for evaluation of non-destructive examination data (H.4(a)). (Section 1R08.1)

- Green. A finding of very low safety-significance was self-revealed for the failure to adequately control relay testing for switchyard breaker installations under Design Change WO KW100691871. Specifically, on March 10, 2011, Dominion Electrical Transmission technicians deviated from standard work practices to test a relay via an internal corporate server, which caused a partial loss of offsite power to the plant through the loss of the main auxiliary transformer backfeed to safety-related bus 6. Licensee corrective actions included a human performance and safety stand-down for substation personnel on the day of the event, the development of a mitigating strategy that outlined expectations and implemented increased direct supervision on critical tasks, and the development of a formal memo describing expectations related to the restricted use of the server for performing remote testing of control functions.

  The finding was determined to be more than minor because, if left uncorrected, the finding had the potential to lead to a more significant safety concern. Specifically, had a different breaker been inappropriately tripped, the station could have experienced a total loss of offsite power. The inspectors concluded that the finding could be evaluated using Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." Specifically, the inspectors qualitatively evaluated the finding by applying the spent fuel pool questions in the Fuel Barrier column of Table 4a, Attachment 4. The inspectors answered "No" to all three questions and determined that the finding was of very low safety significance (Green). The finding has a cross-cutting aspect in the areas of human performance, work practices, because supervisory and management oversight of work activities, including contractors, was not implemented for this evolution (H.4(c)). (Section 4OA3.1)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety-significance and associated non-cited violation (NCV) of Technical Specification 5.4.1, "Procedures," was identified by the inspectors for the failure to implement procedures for shutdown operations involving shutdown operations safety assessments. Specifically, OU-KW-201, "Shutdown Safety Assessment Checklist," step 3.3.1, stated, in part, that a shutdown safety assessment was required to be completed in accordance with the procedure for core cooling; however, the inspectors noted that the February 28, 2011, 6:00 p.m. analysis credited the safety injection system feed and bleed as an available alternate decay heat removal system when the system was not available as described in Section 5.3.2, "Available/Availability," for work scheduled at that time on the emergency core cooling system (ECCS) sump. The licensee initiated condition report CR415539, and at the end of the inspection period, the licensee was performing a causal evaluation to determine the causes of the event and develop corrective actions. On February 28, as a remedial corrective action prior to the start of work, additional steps to the work instructions were added to ensure the equipment would meet the intended function, operators were designated to perform the local manual operations and a pre-job brief was conducted that provided training for using the equipment in the given situation.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of human error (pre-event) and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the availability of the ECCS sump was integral to ensuring that the plant was not in an orange risk path for the evolutions completed on February 28. The inspectors screened the finding as of very low safety significance (Green) because the finding did not degrade the licensee's ability to establish an alternate core cooling path if decay heat removal could not be re-established and, therefore, did not require a Significance Determination Process phase 2 or phase 3 analysis. The finding has a cross-cutting aspect in the areas of human performance, work control, because the licensee failed to plan the work activities by incorporating the need for planned contingencies and compensatory actions to ensure the ECCS sump was available to ensure an orange risk path for core cooling was not entered (H.3(a)). (Section 1R13.1)

- Green. A finding of very low safety-significance and associated non-cited violation (NCV) of Technical Specification 5.4.1, "Procedures," was identified by the inspectors for the failure to establish, implement, and maintain procedures for shutdown operations involving the draining of reactor coolant system (RCS) inventory. Specifically, on March 21, 2011, during a pressurizer draindown evolution, licensed operators unknowingly created a gas void in the reactor vessel closure head (RVCH) that displaced water to a level near the RVCH flange. Subsequent evaluation determined that the procedure for draining the RCS did not contain adequate guidance to ensure that an unacceptable void in the RVCH was not present prior to or formed during operations draindown activities. The licensee subsequently entered the issue into its corrective action program as CR418537 and performed a remedial corrective action of removing the gas void that accumulated in the RVCH. At the end of the inspection period, the licensee was performing an apparent cause evaluation to determine the causes of the event and develop additional corrective actions.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of operating procedure quality and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the formation of the gas void in the RVCH displaced RCS inventory and could have challenged the ability to remove decay heat in the event of a loss of shutdown cooling. The Region III senior reactor analyst determined that this issue is best characterized as a finding of very low safety significance (Green). The finding has a cross-cutting aspect in the areas of human performance, work practices, because operations personnel did not follow or implement the guidance contained in plant procedures. Specifically, procedure OP-KW-AOP-RC-002 prescribed actions to take if a gas void formed in the RVCH that resulted in RVLIS level readings less than 88 percent, which had occurred several hours prior to the start of a pressurizer draining evolution (H.4(b)). (Section 1R20.1)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Kewaunee operated at full power, except for brief downpowers to conduct planned maintenance and surveillance activities, until February 26, 2011, when operators shut down the reactor for a planned refueling outage. The licensee completed the refueling outage on March 26 and returned the reactor to full power on April 2.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- spent fuel pool (SFP) cooling and cleanup system;
- emergency diesel generator (EDG) B;
- EDG A; and
- component cooling water (CCW) pump train A.

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

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On January 31, 2011, the inspectors performed a complete system alignment inspection of the auxiliary feedwater (AFW) system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings 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:

- AX-30, relay room;
- AX-35, control room and air conditioning equipment room;
- TU-90 and TU-91, EDG 1A room and day tank room;
- TU-98, battery room 1-B;
- SC-70A and SC-70B, screen house;
- TU-95B and TU-95C, 480-V switchgear bus 1-61 and 1-62 room and AFW area;
- RC-60, SB-65, reactor containment 592-foot elevation;
- RC-60, SB-65, reactor containment 606-foot elevation;
- RC-60, SB-65, reactor containment 626-foot elevation;
- RC-60, SB-65, reactor containment 649-foot elevation; and
- reserve auxiliary transformer (RAT) deluge testing.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection (FP) program that: adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive FP features in good material condition; and implemented adequate compensatory measures for out-of-service, degraded, or inoperable FP equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on the overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional risk insights, or the potential to impact equipment which could initiate or mitigate a plant transient. The inspectors verified that: fire hoses and extinguishers were in the 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 11 quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

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

This annual heat sink performance inspection constituted two samples as defined in IP 71111.07-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

From February 26, 2011, through March 11, 2011, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system (RCS), steam generator (SG) tubes, feedwater (FW) systems, risk-significant piping and components, and containment systems.

The reviews described in Sections 1R08.1, 1R08.2,1R08.3, 1R08.4, and 1R08.5 below, count as one inspection sample as described by IP 71111.08.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following non-destructive examinations (NDEs) required by the American Society of Mechanical Engineers (ASME) Section XI Code, and/or 10 CFR 50.55a to evaluate compliance with the ASME Code, Section XI, and Section V requirements, and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement:

- ultrasonic testing (UT) of 16-inch diameter FW system weld No. 24;
- UT of 3-inch diameter AFW system weld No. 231;
- magnetic particle testing (MT) of safety injection (SI) pump support attachment welds APSI-1A and APSI-1B; and
- dye penetrant testing (PT) of the No. 34 control rod drive housing circumferential weld.

The inspectors reviewed the following UT weld examination with a recordable indication identified during the previous refueling outage to determine if the indication was characterized, recorded, and evaluated in accordance with the ASME Code Section XI requirements to accept the weld for continued service:

- report UT-09-025; Chemical and Volume Control (CVC) system Seal Water Injection Filter 1A Head Circumferential Weld AFSI-W2.

The inspectors observed fabrication of the following risk-significant pressure boundary ASME Code Section XI Class 2 welds to determine if the licensee: followed the welding procedure; applied appropriate weld filler material; and implemented the applicable Section XI or construction Code non-destructive examinations and acceptance criteria. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of the construction Code and the ASME Code Section IX:

- Containment spray system welds Nos. 1 and 3 fabricated during replacement of isolation valve ICS-5A.

b. Findings

(1) Misapplication of Code Acceptance Criteria for Weld Flaws

Introduction: A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” was identified by the inspectors for the licensee’s failure to establish a procedure that incorporated ASME Code acceptance criteria for evaluation of flaws detected during UT examinations. Consequently, the licensee applied incorrect acceptance criteria to the flaws identified during UT examination of a weld on the CVC seal water injection filter housing.

Description: On March 3, 2011, the inspectors identified that the licensee had applied incorrect ASME Code Section XI acceptance criteria to flaws detected during UT examination of the CVC system Seal Water Injection Filter 1A Head Circumferential Weld.

The licensee had incorporated ASME Code acceptance criteria as an attachment to the procedures that performed the ASME Code Section XI required surface examinations (e.g., MT and PT). However, the inspectors identified that the licensee had not provided or referenced Code acceptance criteria for flaws detected during UT examinations. Specifically, the inspectors reviewed nine UT examination procedures that applied to piping welds, vessel welds, and bolted components. These procedures required recording flaws, but did not provide acceptance criteria. Instead, these procedures required that NDE data records be reviewed and processed in accordance with ER-AA-NDE-140, "Processing of Dominion NDE Data." In procedure ER-AA-NDE-140, the reviewer of NDE data records was required to evaluate indications in accordance with the applicable acceptance standard, but no guidance was provided on how to select the applicable Code acceptance standard. This lack of guidance resulted in a non-conservative and incorrect application of acceptance criteria on a safety-related weld.

In October 2009, the licensee completed a UT examination of the CVC system seal water injection filter 1A head circumferential weld and identified three subsurface flaw indications as documented in examination report UT-09-025. The licensee staff applied acceptance criteria from the ASME Code Section XI, Table IWB-3514-2 "Allowable Planar Flaws." However, for this austenitic weld material and Code Class 2 weld Examination Category (Item C.1.20), the licensee was required to apply the ASME Code Section XI, Article IWA-3100 requirement to evaluate indications in accordance with the construction Code Section III acceptance standards. Because the incorrect acceptance standard (Table IWB-3514-2) provided less conservative flaw size criteria (e.g., bigger flaws allowed) than the Section III Code, the inspectors were concerned that rejectable weld flaws could have been returned to service.

To correct this issue, the licensee evaluated the flaws and determined that they met the ASME Code Section III weld acceptance criteria, and thus did not represent a challenge to the structural integrity of the 1A CVC seal water injection filter housing. The licensee entered this issue into the CAP (CR415894) and issued Revision 3 to procedure ER-AA-NDE-140 to incorporate the ASME Code acceptance criteria for flaws detected during UT examinations. Additionally, the licensee completed an extent-of-condition review and identified one other examination record with flaws identified during UT examination of a SG weld. The licensee confirmed that the correct ASME Code acceptance criteria had been applied for these weld flaws.

Analysis: The inspectors determined that the licensee's failure to establish a procedure that incorporated ASME Code acceptance criteria for evaluation of flaws detected during UT examinations was a violation of 10 CFR Part 50, Appendix B, Criterion IX, and a performance deficiency.

The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated January 10, 2010, because the finding, if left uncorrected, would become a more significant safety concern. Absent NRC identification, the failure to provide ASME Code acceptance criteria could have allowed components with unacceptable cracks to be returned to service. Cracked components returned to service would place safety-related piping systems at increased risk for through-wall leakage and/or failure.

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 I - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems Cornerstone, dated January 10, 2008. The licensee promptly corrected this issue before unacceptable flaws were returned to service. The inspectors answered “No” to the SDP Phase I screening question “Assuming worst case degradation, would the finding result in exceeding the TS limit for any RCS leakage or could the finding have likely affected other mitigation systems resulting in a total loss of their safety function assuming the worst case degradation?” Therefore, the finding screened as having very low safety significance (Green).

This finding had a cross-cutting aspect in the area of human performance, work practices, because the licensee staff did not effectively implement human error prevention techniques. Specifically, the failure to establish a procedure that incorporated ASME Code acceptance criteria was caused by inadequate peer checking during the licensee’s review and approval of procedure ER-AA-NDE-140 (H.4(a)). The inspectors reached this conclusion based on evaluation of the preliminary results of the licensee’s investigation, which identified inadequate peer checking as the primary cause of this finding.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” required, in part, that measures shall be established to ensure that special processes, including non-destructive testing are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Contrary to this, on March 3, 2011, the inspectors identified that the licensee had not provided a non-destructive testing procedure that controlled the application of acceptance criteria in accordance with the ASME Code Section XI. Specifically, acceptance criteria were not provided for evaluation of flaws detected during inservice UT examination of Code Class 1 and 2 components. Consequently, the licensee applied the incorrect acceptance criteria to the weld flaws identified at the CVC seal water injection filter 1A (reference examination report No. UT-09-025). Because of the very low safety significance of this finding and because the issue was entered into the CAP, as CR415894, it will be treated as an NCV, consistent with consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000305/2011002-01, Misapplication of Code Acceptance Criteria for Weld Flaws).

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the reactor vessel head, a bare metal visual (BMV) examination was required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors observed the BMV examination conducted on the reactor vessel head at each of the penetration nozzles to determine if the activities were conducted in accordance with the requirements of ASME Code Case (CC) N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

- if the required visual examination scope/coverage was achieved and limitations (if applicable were recorded), in accordance with the licensee procedures;
- if the licensee criteria for visual examination quality and instructions for resolving interference and masking issues were adequate; and
- for indications of potential through-wall leakage, that the licensee entered the condition into the CAP and implemented appropriate corrective actions (CAs).

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

On February 26, 2011, the inspectors observed the licensee staff performing visual examinations to detect boric acid deposits and system leaks for portions of the RCS and other safety systems inside containment. The inspector observed these examinations to determine whether the licensee focused on locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following licensee evaluations of components with boric acid deposits to determine if the affected components were documented and properly evaluated in the CAP. Specifically, the inspectors evaluated the licensee's CAs to determine if degraded components met the component Construction Code and/or the ASME Section XI Code.

- CR351056, 1A Reactor Coolant Pump Seal and Flange; and
- CR116439, Letdown Heat Exchanger Flange.

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

- CR353840, Moist Boric Acid Deposit LD-332 Cap; and
- CR353866, Seal Leak on Reactor Coolant Pump 1A.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition of eddy current testing (ET) data, reviewed video recordings of the SG secondary side visual examination and cleaning, interviewed ET data analysts, and reviewed documentation related to the SG ISI program to determine if:

- the numbers and sizes of SG tube flaws/degradation identified were consistent with the licensee's previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to meet the TSs, and the Electric Power Research Institute (EPRI) TR–107569, Pressurized Water Reactor Steam Generator Examination Guidelines;
- the SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- the licensee identified new tube degradation mechanisms and implemented adequate extent-of-condition inspection scope and repairs for the new tube degradation mechanism;
- the licensee implemented repair methods, which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- the licensee implemented an inappropriate "plug on detection" tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons-per-day or the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI TR-107569, Pressurized Water Reactor Steam Generator Examination Guidelines;
- the licensee performed secondary side SG inspections for location and removal of foreign materials;
- the licensee implemented repairs for SG tubes damaged by foreign material; and
- foreign objects were left within the secondary side of the SGs, and if so, that the licensee implemented evaluations, which included the effects of foreign object migration and/or tube fretting damage.

a. <u>Findings</u>

No findings were identified.

.5 <u>Identification and Resolution of Problems</u>

a. <u>Inspection Scope</u>

The inspectors performed a review of ISI/SG related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if:

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

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. Documents reviewed are listed in the Attachment to this report.

a. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On March 18 and March 26, 2011, the inspectors observed a crew of licensed operators in the plant’s simulator during licensed operator training activities to verify that training was being conducted in accordance with licensee procedures, and adequately addressed plant modifications. The inspectors evaluated the following areas during training:

- adequacy of revised operating procedures;
- prioritization, interpretation, and verification of new annunciator alarms;
- correct use and implementation of revised abnormal and emergency procedures;
- control board equipment manipulations; and
- oversight and direction from supervisors.

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 were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues for the following systems:

- CCW system;
- incore instrumentation and inadequate core cooling monitor; and
- SI 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/functions classified as (a)(2) or appropriate and adequate goals and CAs 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 that 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 three quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

.1 Inadequate Work Instructions Results in Potential Orange Path

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 (SR) equipment to verify that the appropriate risk assessments were performed prior to removing equipment for work during the week of February 28, 2011.

These activities were selected based on the potential risk significance relative to the RS 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.

This maintenance risk assessment and emergent work control activity constituted one sample as defined in IP 71111.13-05.

b. Findings

Inadequate Work Instructions Results in Potential Orange Path

Introduction: A finding of very low safety-significance and associated NCV of TS 5.4.1, "Procedures," was identified by the inspectors for the failure to implement procedures for shutdown operations involving shutdown operations safety assessments. Specifically, OU-KW-201, "Shutdown Safety Assessment Checklist," step 3.3.1, stated, in part, that a shutdown safety assessment was required to be completed in accordance with the procedure for core cooling; however, the inspectors noted that the February 28, 2011, 6:00 p.m. analysis credited the SI system feed and bleed as an available alternate decay heat removal system when the system was not available as described in Section 5.3.2, "Available/Availability," for work scheduled at that time on the emergency core cooling system (ECCS) sump.

Description: On February 28, maintenance technicians prepared to perform Work Order KW100550237 for temporary modification (TMOD) 2009-07, which installed a pipe plug upstream of valve SI-350B in ECCS sump B, which is common to both trains of SI. The inspectors observed the pre-job brief and reviewed the work instructions and tagout to complete this evolution. The inspectors noted that the pre-job brief and work instructions covered the removal of the ECCS sump B screen, installation of the plug; and reinstallation of the screen, without any additional contingency actions. During a delay in the actual implementation of the field work, the inspectors reviewed the February 28 6:00 p.m. shutdown safety assessment and noted that core cooling was yellow because credit was taken for the SI system feed and bleed. Without an alternate decay heat removal method of feed and bleed, the core cooling risk color was orange, which required additional compensatory measures and licensee actions.

The inspectors reviewed procedure OU-KW-201, "Shutdown Safety Assessment Checklist," which prescribed the actions for completion of the shutdown safety assessment. Step 3.3.1 stated, in part, that a shutdown safety assessment was required to be completed in accordance with the procedure for core cooling utilizing Attachment 2. Attachment 2, "Decay Heat Removal Key Safety Function Background and Instructions," stated, in part, that for SI feed and bleed one train of containment sump recirculation must be available. Step 5.3.2, "Available/Availability," required, in part, that in order to credit equipment as available the following must be met: the equipment had to have written instructions for using the equipment to meet the intended function; designated operators were trained for using the equipment in the given situation; and the local manual operations were performed by designated operators. The inspectors concluded that for this planned evolution the requirements of procedure OU-KW-201 were not met and immediately informed the outage control center management of their concerns that if the work proceeded as written and briefed, the plant would be in an orange risk condition, without the appropriate contingencies in place.

The outage control center placed a hold on the start of the work, reviewed the issues raised by the inspectors, and validated the inspectors' concerns. The licensee then implemented remedial CAs by: adding work instruction steps that upon notification by the control room, the workers in the field would immediately stop work, exit the ECCS sump, and securely fasten the sump screen cover prior to exiting containment; establishing a communication method between the control room and workers;

conducting a new pre-job brief where workers discussed their roles, responsibilities and equipment needed to restore the ECCS sump to an available status; and assigned workers as designated operators. Therefore, when the work was performed the required contingency actions were met and the ECCS sump was considered available in accordance with procedure OU-KW-201.

Analysis: The inspectors determined that the licensee's failure to implement procedure steps as prescribed, to ensure the availability of the ECCS sump for core cooling to prevent an orange risk path, was a performance deficiency warranting a significance evaluation.

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 the finding was associated with the Mitigating Systems Cornerstone attribute of human error (pre-event) and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the availability of the ECCS sump was integral to ensuring that the plant was not in an orange risk path for the evolutions completed on February 28.

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 1, "PWR Hot Shutdown Operation: Time to Core Boiling <2 Hours," contained in Attachment 1 and determined that the finding affected core heat removal guidelines I.B(1), "Procedures," and I.C(2), "Equipment." The inspectors screened the finding as very low safety significance (Green) because it did not degrade the licensee's ability to establish an alternate core cooling path if decay heat removal could not be re-established and, therefore, did not require a phase 2 or phase 3 analysis.

The finding has a cross-cutting aspect in the areas of human performance, work control, because the licensee failed to plan the work activities by incorporating the need for planned contingencies and compensatory actions to ensure the ECCS sump was available to ensure an orange risk path for core cooling was not entered (H.3(a)).

Enforcement: Technical Specification 5.4.1, "Procedures," requires, in part, that written procedures shall be implemented covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2. Regulatory Guide 1.33, states, in part, that procedures for shutdown operations be prepared, as appropriate.

Contrary to this, on February 28, 2011, the licensee performed procedure OU-KW-201 for a shutdown safety assessment; however, the inspectors identified that the licensee incorrectly credited the SI system feed and bleed as an available alternate decay heat removal system. Specifically, Sections 3.3.1 and 5.3.2 required, in part, that in order to credit SI system feed and bleed, the ECCS sump was required to be available with written instructions for using the equipment to meet the intended function, designated operators trained for using the equipment in the given situation and the local manual operations performed by designated operators. None of the actions prescribed in Sections 3.3.1 and 5.3.2 were in place for the implementation of the TMOD 2009-07 work instructions and the ECCS sump would have been unavailable as planned, resulting in an orange risk path. Because this violation was of a very low safety significance and because it was entered into the licensee's CAP, as CR415539, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000305/2011002-02; Inadequate Work Instructions Results in Potential Orange Path)

At the end of the inspection period, the licensee was performing a causal evaluation to determine the causes of the event and develop CAs. On February 28, as a remedial CA, additional steps to the work instructions were added to ensure the equipment would meet the intended function, operators were designated to perform the local manual operations and a pre-job brief was conducted that provided training for using the equipment in the given situation.

.2 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 to verify that the appropriate risk assessments were performed prior to removing equipment for work during the following weeks:

- February 14, 2011;
- March 14, 2011;
- March 21, 2011; and
- March 28, 2011.

These activities were selected based on their potential risk significance relative to the RS 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 four samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- review of open operability evaluations for improved technical specifications (ITS);
- CR416009, Operations Procedures Limit Powering Both ECCS Busses from RAT; and
- Operability Determination OD407, Resolve Potential Degraded Condition Resulting from Fast Transfer.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and USAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of CA 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 three samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification(s):

- install/remove pipe plug upstream of SI-350A/B, TMOD 2009-07 (temporary); and
- DCR 3609-2, AFW flow control cross-tie (permanent).

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screenings against the design basis, the USAR, and the TSs, as applicable, to verify that the modifications did not affect the operability or availability of the affected systems. The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- EDG B return to service;
- EDG A return to service;
- SI pump A return to service;
- AFW pump A;
- AFW pump B; and
- An EDG governor speed setting motor clutch adjustment.

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 CA 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 six post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Unintended Voiding of the Reactor Vessel Closure Head (RVCH)

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the refueling outage (RFO), conducted February 26 through March 26, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown activities and monitored licensee controls for the outage activities listed below:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service;
- configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- monitoring of decay heat removal processes, systems, and components;
- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity; and
- licensee identification and resolution of problems related to RFO activities.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted a portion of the RFO sample as defined in IP 71111.20-05 and documented in Section 1R20.2 of this report.

b. Findings

Unintended Voiding of the Reactor Vessel Closure Head

Introduction: A finding of very low safety-significance and associated NCV of TS 5.4.1, “Procedures,” was identified by the inspectors for the failure to establish, implement, and maintain procedures for shutdown operations involving the draining of RCS inventory. Specifically, during a pressurizer draindown evolution on March 21, 2011, licensed operators unknowingly created a gas void in the RVCH that displaced water to a level near the RVCH flange. Subsequent evaluation determined that the procedure for draining the RCS did not contain adequate guidance to ensure that an unacceptable void in the RVCH was not present prior to or formed during draindown activities.

Description: On March 21, 2011, licensed operators performed RCS dynamic venting with the plant in Mode 5 and the pressurizer full (solid), in order to remove any gas voids. At that time, the RCS pressure was approximately 100 pounds per square inch gauge (psig) and all pressurizer level indications read 100 percent. To facilitate a retest of the EDG A governor following the RCS venting, the operators were required to draindown to 60 percent pressurizer level. The operators utilized procedure OP-KW-NOP-RCS-005, “Draining the Reactor Coolant System,” which did not direct opening a pressurizer head vent or a RVCH vent until pressurizer level was lowered to 55 percent. Utilizing Attachment C of the procedure, the operators estimated a total volume of 3,000 gallons would have to be drained from the system to decrease from 100 to 60 percent level.

The operators began the RCS draindown at approximately 4:30 a.m. and RCS pressure lowered to 35 psig and stabilized. The operators had drained approximately 2,100 gallons of RCS water, as evidenced by the chemical volume control hold up tank level increase of approximately 7 percent, but all pressurizer level indications remained at 100 percent level. The reactor vessel level indication system (RVLIS) showed approximately 62 percent. The draindown evolution was stopped at approximately 5:00 a.m. for operators to evaluate all available indications in the control room and to understand why RCS pressure had stabilized with no indication of a level change in the pressurizer. After reviewing all available indications and caucusing with outage nightshift personnel, the operators reentered procedure OP-KW-NOP-RCS-004, “Filling and Venting the Reactor Coolant System,” and opened the RVCH vents to vent the void that had formed in the RVCH. The oncoming dayshift reactor operators notified the night shift operations crew that OP-KW-AOP-RC-002, “Abnormal Refueling Water Level,” was the correct procedure for venting the RVCH with abnormal RCS levels.

On March 21, 2011, the inspectors reviewed the control room logs and noted that no abnormalities or abnormal operating procedure entries were discussed. The inspectors noted that CR418537 was written on dayshift. The CR documented that the RCS draindown to 60 percent level was stopped, a void had grown in the RVCH, and procedure OP-KW-AOP-RC-002 was utilized to vent the gas void in the RVCH. The CR was screened as a level 3, defined as a routine condition, with minimal CAs planned. Because the logs and condition report lacked detail, the inspectors reviewed the plant procedures and plant process computer history to ascertain the size of the void. The inspectors noted that OP-KW-AOP-RC-002 had an entry condition of unexpected level deviations in the RCS and step 4 verified that the RVCH was full, with an expected response of RVLIS indicating greater than or equal to 88 percent level, considered to be an acceptable void volume.

During their review of the plant process computer data, the inspectors identified that at approximately 2:00 a.m., well before the start of the draindown at 4:30 a.m., both trains of RVLIS indicated that a void in the RVCH had formed that met the AOP criterion of 88 percent for RVCH venting. At the start of the pressurizer draindown evolution at 4:30 a.m., the void had grown to 78.7 percent RVLIS level. At approximately 5:00 a.m., after operators drained 2,100 gallons of RCS water with no indication of a change in pressurizer level, both trains of RVLIS indicated 62 percent level, which equated to a RCS level at the RVCH flange (53 percent level equated to an RCS level at the reactor vessel flange). The gas void formation was caused by the depressurization of the RCS, which resulted in gases coming out of the RCS and collecting in the RVCH.
The inspectors questioned station management regarding the lack of significance placed on the CR, since the void in the RVCH was significantly greater than what was expected during RCS dynamic venting. Station management did not have all the facts surrounding the incident due to the lack of information documented in the station logs and CR. Subsequent to the inspectors' discussions, the CR was reassigned the appropriate significance level and an apparent cause evaluation was requested. Therefore, this finding will be characterized as NRC-identified because the inspectors added value in the identification of previously unknown weakness in the licensee’s initial classification, evaluation, and CAs associated with this issue.

At the end of the inspection period, the licensee continued to evaluate the issue in the CAP. Based on the review of operator written statements, several issues of concern were identified including: procedure OP-KW-NOP-RCS-005 did not adequately address all parameters necessary for a successful draindown evolution; just-in-time training for the draindown may have been warranted; the pre-job brief did not discuss the entry conditions for OP-KW-AOP-RC-002 based on RVLIS indications; and not all operations crews were familiar with OP-KW-AOP-RC-002.

Analysis: The inspectors determined that the failure to have an adequate procedure to conduct RCS draindowns without monitoring the presence of or the formation of an unacceptable gas void volume in the RVCH was a performance deficiency warranting a significance evaluation.

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 the finding was associated with the Mitigating Systems Cornerstone attribute of operating procedure quality and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the formation of the large gas void in the RVCH displaced RCS inventory and could have challenged the ability to remove decay heat in the event of a loss of 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 2 contained in Attachment 1, dated May 24, 2004, and determined that the finding required a Phase 2 analysis since the finding led to the loss of RCS inventory based on inadequate procedure guidance and personnel error (Section II.B.2 of Checklist 2).

The Region III senior reactor analyst (SRA) performed the assessment using Appendix G, Attachment 2, "Phase 2 Significance Determination Process Template for PWR During Shutdown," dated February 28, 2005. The SRA determined this to be a precursor to an initiating event (a loss of reactor inventory precursor - LOI). The plant operating state (POS) was determined to be "POS 1" (vessel head on and RCS closed). The time window was late ("TW-L").

The initiating event likelihood for LOI using Table 3, "Initiating Event Likelihoods (IELs) for LOI Precursors," was listed as "1" since the drain path was known and was successfully isolated (draindown stopped). The SRA considered it likely that operators would diagnose and take action to prevent a loss of residual heat removal (RHR) pump suction before a significant loss of inventory event occurred. Using the SPAR-H Human Reliability Analysis Method (NUREG/CR-6883, September 2004), the IEL was determined to be "2."

The SRA used the Loss of Level Control (LOLC) Worksheet 1, "SDP for a PWR Plant - Loss Level Control in POS 1 (RCS Closed)," based on RCS temperature being less than 200 degrees F and the guidance in Step 4.3.3 of Appendix G, Attachment 2. The SRA evaluated the remaining mitigating capability credit from the worksheet and determined that the combined sequences from Worksheet 1 had a risk significance of about 3.3E-7. The most significant core damage sequence involved loss of SG cooling and failure of RCS injection and bleed before core damage. Therefore, the SRA determined that this issue is best characterized as a finding of very low safety significance (Green).

The finding has a cross-cutting aspect in the areas of human performance, work practices, because operations personnel did not follow or implement the guidance contained in plant procedures. Specifically, procedure OP-KW-AOP-RC-002 prescribed actions to take for a gas void in the RVCH that resulted in RVLIS level readings less than 88 percent, which had occurred several hours prior to the start of a pressurizer draining evolution (H.4(b)).

Enforcement: Technical Specification 5.4.1, "Procedures," requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2. Regulatory Guide 1.33, states, in part, that procedures for filling, venting, and draining be prepared, as appropriate, for the RCS.

Contrary to this, on March 21, 2011, the inspectors identified that the licensee performed procedure OP-KW-NOP-RCS-005 to drain the pressurizer; however, the procedure did not establish or maintain adequate instructions to ensure RCS inventory was not affected. Specifically, the procedure did not prescribe actions for monitoring and venting accumulated gases in the RVCH prior to and during the RCS draindown to detect and monitor for a unacceptable gas void in the RVCH, which displaced RCS inventory in the reactor vessel. Because this violation was of a very low safety significance and because it was entered into the licensee's CAP, as CR418537, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000305/2011002-03; Unintended Voiding of the Reactor Vessel Closure Head).

At the end of the inspection period, the licensee was performing an apparent cause evaluation to determine the causes of the event and develop CAs. As a remedial CA, on March 21, operators removed the gas void that accumulated in the RVCH as a result of the RCS draindown activities.

.2 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the OSP and contingency plans for the RFO, conducted February 26 through March 26, 2011, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the following outage activities:

- licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out-of-service;
- implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes, systems, and components;

- controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity;
- maintenance of secondary containment as required by TSs;
- refueling activities, including fuel handling;
- startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- scheduling of covered workers such that the minimum days off for individuals working on the outage activities are in compliance with 10 CFR 26.205(d)(4) and (5); and
- licensee identification and resolution of problems related to RFO activities.

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings 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:

- auxiliary diesel-driven fire pump testing (routine);
- turbine-driven AFW pump full flow testing (routine);
- nuclear power range channel 2, N-42 monthly test under SP-48-003F (routine);
- N-41/N-42 quarterly calibrations under SP-48-004G and SP-48-004H (routine);
- RHR train B under SP-34-099B (inservice testing (IST));
- CCW pump train A under SP-31-168A (IST);
- quarterly AFW full flow testing (IST); and
- main steam isolation valve (MSIV) timing tests (containment isolation valve (CIV)).

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 was 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, three inservice testing samples, and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted one complete sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators for the occupational exposure 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.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors assessed whether 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 evaluated 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 selected plant areas, and evaluated whether the thoroughness and frequency of the surveys were 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:

- reactor coolant pump work;
- refueling work of reactor head in and around reactor cavity; and
- core offload/reload and associated work.

For these work activities, the inspectors assessed whether 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 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 were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt 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), "Exemptions To Labeling Requirements."

The inspectors reviewed the following radiation work permits used to access high radiation areas and evaluated the specified work control instructions or control barriers:

- reactor coolant pump work;
- refueling work of reactor head in and around reactor cavity; and
- core offload/reload and associated work.

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

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter 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 CAP and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings 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 radiological control 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 and 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 type(s) 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 several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards 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, radiation work permits, 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 electronic personal dosimeters in high noise areas as high radiation area 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 that 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 radiation work permits for work within airborne radioactivity areas with the potential for individual worker internal exposures:

- reactor coolant pump work;
- refueling work of reactor head in and around reactor cavity; and
- core offload/reload and associated work.

For these radiation work permits, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, and entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

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 examined the posting and physical controls for selected high radiation areas and very high radiation areas to verify conformance with the occupational performance indicator.

b. Findings

No findings were identified.

.6 Risk-Significant High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk high radiation areas and very high radiation areas. The inspectors discussed methods employed by the licensee to provide stricter control of very high radiation area access as specified in 10 CFR 20.1602, "Control of Access to Very High Radiation Areas," and Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants." The inspectors assessed whether any changes to licensee procedures substantially reduced the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

The inspectors evaluated licensee controls for very high radiation areas and areas with the potential to become a very high radiation areas to ensure that an individual was not able to gain unauthorized access to the very high radiation area.

b. Findings

No findings 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 radiological conditions in their workplace and the radiation work permit controls/limits in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed 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 CA approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the CAs planned or taken.

b. Findings

No findings were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the radiation work permit 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 radiological problem reports since the last inspection that found the cause of the event to be 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 CA approach taken by the licensee to resolve the reported problems.

b. Findings

No findings 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 being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the CAs 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 their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71124.02)

The inspection activities supplement those documented in Inspection Report 05000305/2010004, and constitute a partial sample as defined in IP 71124.02-05.

.1 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers were familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas) and whether there were any procedure compliance issues (e.g., workers were not complying with work activity controls). The inspectors observed radiation worker performance to assess whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

4OA2 Identification and Resolution of Problems (71152)

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

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

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely CAs, 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 CAs were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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 CR packages.

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

b. Findings

No findings were identified.

.3 Selected Issue Follow-Up Inspection: Root Cause Evaluation (RCE) 224, Scaffolding Affecting SR Equipment

a. Inspection Scope

The inspectors reviewed the CAs from RCE 224, Scaffolding Affecting SR Equipment. Specifically, the inspectors reviewed processes and procedures used to ensure that scaffolding was constructed properly when in the vicinity of SR equipment and the inspectors also physically walked down a variety of scaffolding constructed throughout the plant to ensure that the scaffolding conformed to the requirements established in the licensee's procedures.

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

b. Findings

No findings were identified.

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

.1 Partial Loss of Offsite Power on March 10, 2011

a. Inspection Scope

The inspectors responded to the control room following a partial loss of offsite power on March 10, 2011. The loss occurred after the licensee inadvertently opened a switchyard breaker that was providing power to various non-safeguards busses, as well as bus 6, a 4160-volt safeguards bus. The EDG B automatically started as expected and restored power to bus 6. The technical support center (TSC) diesel automatically started as expected; however, the output breaker failed to close and power bus 46. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

(1) Partial Loss of Offsite Power Caused by Less Than Adequate Interface and Oversight of Switchyard Modification Work

Introduction: A finding of very low safety-significance was self-revealed for the failure to adequately control relay testing for switchyard breaker installations under Design Change WO KW100691871. Specifically, two Dominion Electrical Transmission (DET) technicians deviated from standard work practices to test a relay via an internal corporate server, which caused a partial loss of offsite power to the plant through the loss of the main auxiliary transformer (MAT) backfeed to bus 6.

Description: On March 10, 2011, DET technicians performed relay testing in the switchyard control house on newly installed breaker RST-199, a supply breaker to the station reserve auxiliary transformer, located in the protected area and and de-energized at the time. The standard for relay testing was to connect a laptop directly to a breaker interface in the switchyard south control house and, as part of the switchyard risk plan, protective signs or barriers had been placed in front of breaker interfaces associated with in-service switchyard breakers that could affect offsite power. During the testing process, while moving from one breaker to the next, the battery to the lead technician's laptop became dislodged, requiring him to reboot his computer. The other technician (in training), whose laptop was used to monitor the system tests, suggested that they utilize his laptop and test the relay remotely using an internal corporate server, versus connecting the laptop directly to the breaker interface. The lead technician agreed to this revised method without consultation of his supervisor, and the relay for an inservice breaker, which provided power backfed to bus 6 through the MAT, was inadvertently selected. When the relay was tested, the breaker opened, as designed, causing a partial loss of offsite power. The EDG B automatically started as designed and restored power to bus 6. At the time of the event, the reactor was defueled with all fuel offloaded into the spent fuel pool (SFP). The SFP cooling requirement, at the time, was one train of cooling, which was maintained throughout the event. Had the adjacent breaker been inadvertently selected, a total loss of offsite power would have occurred.

The licensee's evaluation determined that a restriction on the use of the server to test control functions was sent to DET personnel in an e-mail in February 2009. DET personnel also did not verify that everyone read the e-mail, did not verify or test how much of the information was retained by the employees, and did not incorporate the information into an instruction or procedure. Interviews identified that, while the technicians were unfamiliar with this standard, supervisors knew not to use the server for relay actuation. The evaluation also identified that supervisory oversight of the job was lacking and that the work instruction for the job lacked sufficient detail and required the technicians to perform work from memory for a complex task.

Analysis: The inspectors determined that using an unapproved method of testing, by actuating the protective relaying for breaker RST-199 remotely through the corporate server, was contrary to the DET standard of actuating the protective relaying through a direct connection with the serial port on the breaker interface and was a performance deficiency warranting a significance evaluation.

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, if left uncorrected, the finding had the potential to lead to a more significant safety concern. Specifically, had a different breaker been inappropriately actuated, the station would have experienced a total loss of offsite power. The inspectors concluded this finding was associated with the Initiating Events Cornerstone.

The inspectors determined, with consultation of the Region III SRA, that the available shutdown SDP tools did not address conditions where reactor fuel was completely offloaded into the SFP and could not assess the finding that occurred on March 10. The inspectors concluded that the finding could be evaluated using IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," dated December 22, 2006. Specifically, the inspectors qualitatively evaluated the finding by applying the SFP questions in the Fuel Barrier column of Table 4a located in IMC 0609, Attachment 4, dated January 10, 2008. The inspectors answered "No" to all three questions and determined that the finding was of very low safety significance (Green).

The finding has a cross-cutting aspect in the areas of human performance, work practices, because supervisory and management oversight of work activities, including contractors, was not implemented for this evolution. Specifically, the licensee failed to provide adequate supervisory oversight of testing in the switchyard. This resulted in a partial loss of offsite power and had the potential to cause a complete loss of offsite power for the station (H.4(c)).

Enforcement: No violation of regulatory requirements occurred.
(Finding (FIN) 05000305/2011002-04, Partial Loss of Offsite Power Caused by Less Than Adequate Interface and Oversight of Switchyard Modification Work)

The licensee entered this issue into the CAP as CR417078 and took short-term CAs that included a human performance and safety stand-down for all switchyard personnel on the day of the event, the development of a mitigating strategy that outlined expectations and implemented increased direct supervision on critical tasks, and the development of a formal memo describing expectations related to the restricted use of the server for performing remote testing of control functions. Long-term CAs were being finalized at the conclusion of the inspection period.

(2) Unresolved Item (URI) 2011002-05, Technical Support Center Diesel Fails To Load

Introduction: On March 10, 2011, the licensee inadvertently opened a switchyard breaker that was providing power to various non-safeguards busses, as well as, bus 6, a 4160-volt safeguards bus. The TSC diesel automatically started as expected, however, the output breaker failed to close and power bus 46, as designed.

Description: The licensee's troubleshooting and investigation determined that the TSC output breaker failed to close because of a failed breaker latching relay. The licensee replaced the relay and restored the TSC diesel to functional status. The licensee's apparent cause evaluation was still in progress at the conclusion of the inspection period, and the inspectors did not have enough information to determine if a performance deficiency existed. Pending further review and inspection, this issue was considered a URI (URI 05000305/2011002-05, Technical Support Center Diesel Fails To Load).

.2 (Closed) Licensee Event Report (LER) 05000305/2009-006-01: Protection Instruments Not Calibrated to Individual Technical Specification Setpoint Limits

Licensee Event Report 05000305/2009-006-00 was inspected in the first quarter of 2010 and closed in NRC integrated inspection report 05000305/2010002. This revision to the original LER checked the block for 10 CFR 50.73(a)(2)(vii), which covered any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems, or two independent trains or channels to become inoperable in a single system designed to shutdown the reactor and maintain it in a safe shutdown condition or mitigate the consequences of an accident. The licensee failed to check this block in the original submittal and subsequently corrected the report. No new technical information was submitted. This LER is closed.

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

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 5, 2011, 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 the inservice inspection with Engineering Programs Manager, Mr. D. Asbel, and other members of the licensee staff on March 11, 2011;
- The results of the radiological hazard assessment and exposure controls and ALARA inspection with the Site Vice-President, Mr. S. Scace, on March 4, 2011.

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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Scace, Site Vice-President
R. Simmons, Plant Manager
M. Wilson, Director, Safety and Licensing
S. Yuen, Engineering Director
C. Chovan, Outage and Planning Manager
C. Olson, Radiation Protection Supervisor
D. Asbel, Engineering Programs Manager
D. Emery, Supervisor Nuclear Training
D. Laing, Nuclear Training Manager
D. Lawrence, Operations Manager
J. Gadzala, Licensing Engineer
J. Hale, Radiation Protection Manager
J. Langan, Nuclear Oversight Manager
J. Madden, Engineering Systems Manager
J. Stafford, Organizational Effectiveness Manager
K. Hacker, Dominion NDE Level III
M. Aulik, Engineering Design Manager
P. Bukes, ISI Program Owner
R. Repshas, Licensing
T. Breene, Licensing Manager
T. Evans, Maintenance Manager
T. Olsen, Supply Chain Manager
T. Hanna, SG Program Owner

Nuclear Regulatory Commission

M. Kunowski, Chief, Division of Reactor Projects, Branch 5

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000305/2011002-01	NCV	Misapplication of Code Acceptance Criteria for Weld Flaws (Section 1R08.1)
05000305/2011002-02	NCV	Inadequate Work Instructions Results in Potential Orange Path (Section 1R13.1)
05000305/2011002-03	NCV	Unintended Voiding of the Reactor Vessel Closure Head (Section 1R20.1)
05000305/2011002-04	FIN	Partial Loss of Offsite Power Caused by Less Than Adequate Interface and Oversight of Switchyard Modification Work (Section 4OA3.1)
05000305/2011002-05	URI	Technical Support Center Diesel Fails To Load (Section 4OA3.1)

Closed

05000305/2011002-01	NCV	Misapplication of Code Acceptance Criteria for Weld Flaws (Section 1R08.1)
05000305/2011002-02	NCV	Inadequate Work Instructions Results in Potential Orange Path (Section 1R13.1)
05000305/2011002-03	NCV	Unintended Voiding of the Reactor Vessel Closure Head (Section 1R20.1)
05000305/2011002-04	FIN	Partial Loss of Offsite Power Caused by Less Than Adequate Interface and Oversight of Switchyard Modification Work (Section 4OA3.1)
05000305/2009-006-01	LER	Protection Instruments Not Calibrated To Individual Specification Point (Section 4OA3.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or 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.

1R04 Equipment Alignment

- CR319770; PMP-17-05 Requires Revision Prior To Use
- CR325533; FO-1118 (TSC D/G Filter Housing Crossover Valve) Discovered In Mid Position
- CR334660; Request For An Evaluation Of SD-4B2's Position As An APC
- CR343282; SW Pump A1 And A2 Discharge PI Manifold Isolations Found Closed
- CR363677; Valve From LP Tank To Heaters In The ISFSI Building Was Closed
- CR374361; CW-53C2 Found Closed
- CR382594; Traveling Water Screen 1A2 Maintenance Switch Off
- CR388822; Unintentional Valve Manipulation Of Valve SW-6804 Located In "B" Battery Room
- CR388947; DPI 11033J, (HDP 1B Suct Strainer D/P), Found Isolated And Equalized
- Drawing APM-205; Analytical Part Flow Feedwater System; Revision T
- Drawing APM-213-9; Flow Diagram Diesel Generator Startup Air Compressor A And B And Fish Screen Air; Revision F
- Drawing M-220; Flow Diagram Fuel Oil Systems; Revision AS
- MA-KW-MPM-DGM-004; Changing Oil And Filters On TSC Diesel Generator; Revision 4
- N-CC-31; Component Cooling Water System Operation; Revision 45
- N-CC-31-CL; Component Cooling Water System Prestartup Checklist; Revision 30
- N-SFP-21-CL; Spent Fuel Pool Cooling And Cleanup System Prestartup Checklist; Revision T
- OP No. N-FW-058-CL; Auxiliary Feedwater System Prestartup Checklist; Revision 44
- OPERM-218; Flow Diagram Spent Fuel Pool Cooling And Cleanup System; Revision AF
- OPERXK-100-19, Flow Diagram Component Cooling Water System; Revision AP
- OP-KW-AOP-DGM-002A; Abnormal Diesel Generator A Operation, System DGM-10; Revision 3
- OP-KW-NCL-DGM-001A; Diesel Generator A Prestartup Checklist; System 10 ITS; Revision 4

1R05 Fire Protection

- Calculation No. KPS-70151480-S01; Design Of Breakaway Pin (Cane Bolt) On New Door 49 (DCR-3597); Revision 1
- CR414162; NRC Raises Question Regarding Fire Pump Starting Delays
- CR414345; Door 151 Closer May Need Adjustment
- CR416072; Updates Needed In The Area Summaries Of Pre-Fire Plans (NRC Identified)
- Drawing A-535, PFP-4; Screen House SC-70A, SC-70B/Elevations 569' And 586'; Revision B
- Drawing A-536, PFP-5; 1A Diesel Generator And DG Day Tank Rooms, TU-90, TU-91/Elevation 586; Revision C
- Drawing A-540, PFP-9; 480V Switchgear Bus I-61 And I-62 Room And AFW Pump Area; TU-95B, TU-95C/Elevation 586'; Revision D
- Drawing A-544, PFP-13; Battery Rooms 1A And 1B, TU-97, TU-98/Elevation 606'; Revision D
- Drawing A-552, PFP-21; Relay Room And Loft, AX-30/Elevation 606' And 616'; Revised C
- Drawing A-557, PFP-26; Control Room AX-35, AX-34/Elevation 626'; Revision 6
- FPP-08-15; Appendix R Fire Wrap Inspection; Revision 10
- PFP-13; TU-97, 98/Battery Rooms 1A And 1B; Revised December 19, 2007

- PFP-21; AX-30/Relay Room And Loft
- PFP-26; AX-35, AX-34 (626')/Control Room/Work Control Center; Revised December 19, 2007
- PFP-5; TU-90, 91/DG 1A And Day Tank Rooms; Revised April 25, 2007
- Preventive WO KW100663438; PM08-808: Inspection Of Doors On Elevations 633, 642, 649, And 657; July 23, 2010

1R07 Annual Heat Sink Performance

- ER-KW-NSP-CC-001A; Component Cooling Heat Exchanger 1A Performance Monitoring, System 31; WOKW100280320; Performed February 26, 2011

1R08 Inservice Inspection Activities

- AREVA 3-1275284; Field Procedure for Remote Tolled Plugging Utilizing The LAN SAP Box; Revision 17
- AREVA 3-914585; Secondary Side Visual Inspection Plan And Procedures For Dominion, Kewaunee Unit 1 1R31; Revision 0
- AREVA 51-9151923-000; Kewaunee 1R31 – EPRI Appendix H Eddy Current Technique Review; Revision 0
- AREVA 54-ISI-400-19; Multi-Frequency Eddy Current Examination Of Tubing; December 1, 2010
- BACC Walkdown Results; Recommended New Repairs; Printed February 28, 2011
- CA073187; Evaluate MIC Nodules in B DG Cooler Supply And Return Pipes; April 17, 2008
- CMTR ARCOS Industries LLC; Dated August 27, 2009
- CMTR ARCOS Industries LLC; Dated February 25, 2008
- CMTR ARCOS Industries LLC; Dated March 4, 2007
- Correspondence From D.C. Hintz, Wisconsin Public Service Corporation, To NRC, Re: Generic Letter 88-05: Boric Acid Corrosion Of Carbon; June 3, 1988
- CR098298; Boric Acid At Flange For 1A Excess Letdown Heat Exchanger
- CR116439; Evaluation Of Letdown Heat Exchanger Flange
- CR350540; Indication On Gas Collection Chamber
- CR351056; Evaluation Of 1A RCP Seal And Flange
- CR352042; Out Of Tolerance EDW-H116
- CR353640; Boric Acid Deposit At Cap For LD-332
- CR353866; Seal Leak On RCP 1A
- CR415893; Procedure Qualification Question
- CR416285; BMV Examination Of Reactor Vessel Closure Head
- CR416305; Unauthorized Glue Stick On Purge Dam (ICS-5A)
- CR416327; Inflatable Purge Dam Contacted With Hot Pipe
- CR416507; SG ET Probe Broke In Tube
- CR416515; PLP B SG (R1C68)
- CR417146; Light White Dry Boric Acid At Swagelok Fitting Near RC-602A (NRC Identified)
- CR417150; Light White Dry Boric Acid Deposit From Packing On RC-302A (NRC Identified)
- Dominion Weld Material Control Form 300134; February 28, 2011
- Drawing 1B79542; ADVB-013-96 Bobbin ASME; Revision 2
- Drawing M-1952; SI Discharge Piping Gas Collection Chamber; Revision 0
- ER-AA-NDE-140; Processing Of Dominion NDE Data; Revisions 2 And 3
- ER-AA-NDE-MT-200; ASME Section XI Magnetic Particle Examination Procedure; Revision 4
- ER-AA-NDE-PT-300; ASME Section XI Liquid Penetrant Examination Procedure; Revision 5

- ER-AA-NDE-UT-801; Ultrasonic Examination Of Ferritic Piping Welds In Accordance With ASME Section XI, Appendix VIII, Revisions 1 and 2
- ER-AA-NDE-UT-802; Ultrasonic Examination Of Austenitic Piping Welds In Accordance With ASME Section XI, Appendix VIII, Revision 1
- ER-AA-NDE-UT-803; Ultrasonic Through Wall Sizing Of Pipe Welds In Accordance With ASME Section XI, Appendix VIII, Revision 1
- ER-AA-NDE-UT-810; Ultrasonic Examination Of Dissimilar Metal Welds In Accordance With ASME Section XI, Appendix VIII, Revision 2
- ER-AA-NDE-VT-604; Visual Examination (VE) For Leakage Of PWR Reactor Head Penetrations; Revision 1
- ER-AP-SGP-101; Steam Generator Program; Revision 4
- ETSS BOB 001 MIZ80 RO; March 4, 2011
- ETSS RCP 001 MIZ80 RO; March 4, 2011
- ETSS RCP 002 MIZ80 RO; March 4, 2011
- ETSS RCP 003 MIZ80 RO; March 4, 2011
- ETSS RESO 001 MIZ80 RO; March 4, 2011
- Kewaunee KR 31 Boric Acid Corrosion Control Data Sheet For Leaks Identified During KR 31 For Cleaning; February 28, 2011
- Kewaunee Power Station Refueling Outage 31, Spring 2011 Steam Generator Degradation Assessment, Revision 0
- Kewaunee R29 Steam Generator Condition Monitoring And Operational Assessment Report, Revision 1
- KPS Engineering Specification ES-2003; Revision 21
- Letter; PDI Position Paper- Qualified Ranges For Small Diameter Ferritic Piping; October 15, 2003
- NDE Inspector Certification T. Thomas; March 3, 2011
- NDE Procedure Qualification Report; ASME Section XI Visible Solvent Removable Liquid Penetrant Examination Procedure; March 7, 2011
- P101; General Piping And Pressure Vessel Welding Procedure; Revision 16
- Qualified Data Analyst Letter; QDA Review Of The Kewaunee (1R31) Appendix H Document; March 5, 2011
- Report MT-11-011; SI Pump APSI-1B Integrally Welded Support; March 3, 2011
- Report MT-11-012; SI Pump APSI-1A Integrally Welded Support; March 3, 2011
- Report PT-09-018; Circ Weld on SI Gas Collection Chamber; October 3, 2009
- Report PT-09-026; Circ Weld on SI Gas Collection Chamber; October 10, 2009
- Report PT-11-010; ICS-5A Welds KW-10-01001 & 3; March 7, 2011
- Report PT-11-019; ICS-5A Repair Weld KW-10-01003R1; March 9, 2011
- Report RT; ICS-5A – Weld KW-10-01001; March 4, 2011
- Report RT; ICS-5A – Weld KW-10-01003; March 5, 2011
- Report RT; ICS-5A – Weld KW-10-01003R1; March 6, 2011
- Report UT-08-265; AFW Pipe To Elbow Weld AFW-W94; April 26, 2008
- Report UT-08-268; AFW Pipe To Elbow Weld AFW-W94; April 26, 2008
- Report UT-09-025; Seal Water Injection Filter 1A Head Circumferential Weld AFSI-W2; October 13, 2009
- Report UT-11-041; 16” Valve To Elbow Weld FW-W24; March 10, 2011
- Report UT-11-042; 3” Pipe To Elbow Weld AFW-W24; March 10, 2011
- Report VT-11-076; Reactor Vessel Closure Head Penetrations; March 6, 2011
- Welder Performance Qualification AS-0380; December 12, 2010
- Welder Performance Qualification GD-9512; February 23, 2011
- Welder Performance Qualification SP-2853; December 6, 2010
- Welding PQR 801; December 20, 2001

- Welding PQR 830; July 20, 2001
- Welding PQR 831; July 20, 2001
- Welding Technique Sheet 801; Revision 8
- WO KW-100369619; Replace 1CS-5A And Install CS-H58; Revision 0
- WO KW-100494435; SI-39A; October 21, 2009

<u>1R11 Licensed Operator Requalification Program</u>

- EOP E-0; Reactor Trip Or Safety Injection, Attachment A; Revision 43
- EOP E-0; Reactor Trip Or Safety Injection; Revision 43
- EOP E-0; Reactor Trip Or Safety Injection; Revision 43
- EOP E-2; Faulted Steam Generator Isolation; Revision 24
- EOP ES-01; Reactor Trip Response; Revision 31
- Figure 3 – AFW Discharge Piping Flow Diagram – DCR 3609-2 Configuration
- OP-DW-NOP-HD-001; Heater And Moisture Separator Drain And Bleed Steam System, System No. 11; Revision 1
- OP-KW-AOP-HD-001; Abnormal Heater Drain Operation, System No. HD-11; Revision 2
- OP-KW-ARP=47062-U; HTR Dain Tank VFD-A/B Trouble, System No. MS-06; Revision 0, Draft D
- OP-KW-ARP-47062-S; HTR Drain Tank Level High/Low, System No. MS-06; Revision 1
- OP-KW-NOP-AFW-001; Auxiliary Feedwater System, System No. 05B; Revision 4
- OP-KW-NOP-SUB-003; RST And TST Load Tap Changer Operation; Revision 0, Draft A
- SEG No. LRC-11-JT201; R-31 Rx S/U By Dilution & Power Ascension JITT; March 11, 2011
- SEG No. LRC-11-JT202; R-31 AFW, HD, And Tap Changer JITT; March 14, 2011

<u>1R12 Maintenance Effectiveness</u>

- ACE18150; ICCM Train A Observed Flashing
- ACE18320; CR392422; ICCMS Train A Power Supply Found Out Of Tolerance
- CA013962; MR (a)(1) Action (CET) Replacement and Associated Connectors Upgrade
- CA072254; MR (a)(1) Action t-track DCR002951 tracking action
- Component Cooling Water System Balancing; June 2009 – November 2010
- Component Cooling Water System Unavailability; June 2009 – November 2010
- Kewaunee Maintenance Rule Monthly Review Report For January 2011
- Licensee Maintenance Rule Data Tracking Sheets; Component Cooling Water System; June 2009 – November 2010
- Licensee Maintenance Rule Data Tracking Sheets; System Basis; Safety Injection System; January 2009 – February 2011
- Maintenance Rule (a)(1) Systems With Open Corrective Actions - 5 Systems; January 2011
- Maintenance Rule Performance Criteria; Component Cooling Water System, Attachment B; Revision 4
- Maintenance Rule Performance Criteria; Incore Instrumentation System; Attachment B, Revision 0
- Maintenance Rule Performance Criteria; Safety Injection System; Attachment B, Revision 2
- Maintenance Rule Scoping Questions; Component Cooling Water System, Attachment A; Revision 2
- Maintenance Rule Scoping Questions; Incore Instrumentation System; Attachment A, Revision 0
- Maintenance Rule Scoping Questions; Safety Injection System; Attachment A, Revision 1
- Maintenance Rule System Basis; Component Cooling Water System; Revision 11
- Maintenance Rule System Basis; Incore Instrumentation System; Revision 5

- Maintenance Rule System Basis; Safety Injection System; Revision 9
- Safety Injection System Balancing; July 2009 – December 2010
- Safety Injection System Unavailability; January 2009 – February 2011

1R13 Maintenance Risk

- CR4155539; NRC Concerns Identified During PJB For SI-350B Pipe Plug Installation
- OU-KW-201; Attachment 1, Shutdown Safety Assessment (SSA) Checklist; February 28, 2011
- KW100550237; TMod 2009-07 (Install Pipe Plug Upstream Of SI-350B), Sump B; February 28, 2011
- OU-KW-201; Shutdown Safety Assessment Checklist; Revision 4
- Kewaunee Unit 1 2011 RFO Shutdown Risk Review Report; February 21, 2011

1R15 Operability Evaluations

- 50.59/72.48 Screen For Compensatory Measure For OD407; March 22, 2011
- CR416009; Ops Procedures Incorrectly Limit Powering Both ESF Buses From The Rat
- OD000407; Resolve Potential Degraded Condition Resulting From Fast Transfer; March 21, 2011

1R18 Plant Modifications

- CR349525; ECN Required For TMods 2009-06 And 2009-07 Sump B Pipe Plug
- CR401917; TMod 2009-06/-07 Sump B Pipe Plug Diameter Change
- CR413365; Revise Torque Values For TMod 2009-06 And 2009-07
- DCR 3609-2, Rev. 0, AFW Flow Control; 10CFR50.59 Evaluation 10-03-00, Rev. 0; Attachments A And B
- DCR 3609-2, Rev. 0, AFW Flow Control; 10CFR50.59 Screening, Attachments A, B, And C
- DNAP-3004 – Attachment 6; Evaluation No. 10-03-00, Revision 0
- ECN 2009-07-01; For TMod 2009-07; Revision 0
- ECN 3609-2-01; Prefabrication Of AFW Piping; Revision 0, December 7, 2010
- ECN 3609-2-02; Prefabrication Of SW Flush Piping 2/1 Hangers; Revision 0, January 4, 2011
- ECN 3609-2-03; Prefabrication Of SW Flush Piping 2/1 Hangers; Revision 0, January 10, 2011
- ECN 3609-2-04; Prefabrication Of SW Flush Piping 2/1 Hangers; Revision 0, January 13, 2011
- ECN 3609-2-05; Prefabrication Of AFW Pipe Hangers; Revision 0, January 25, 2011
- ECN 3609-2-06; Revised Location Of The 8-inch Pipe Cap Based Upon Field Walkdowns And Hanger Proximity; Revision 0, January 26, 2011
- ECN 3609-2-07; Add Vent Tubing And Fitting Details For SW-7520 And SW-7602 For Work Planning And Installation; Revision 0, February 4, 2011
- ECN 3609-2-08; MS-05-319 – Increase Baseplate Size And Shift Rod Attachment Because Of Interferences; Revision 0, February 8, 2011
- ECN 3609-2-09; Corrections Noted During Final Design Review And Preparation Of Electrical Drawings For PE Seal; Revision 0, February 16, 2011
- ECN 3609-2-10; Remove Erroneous Weld Symbol; Revision 0, February 22, 2011
- ECN 3609-2-11; Length Of Elbow Offset/Transition; Revision 0, February 25, 2011
- GNP-04.04.01-1; 50.59 Applicability Review Of DCR 3609-2; February 18, 2011
- Letter from D.C. Hintz To NRC; Re: Follow-up Response To NRC Bulletin No. 88-04: Potential Safety-Related Loss; Dated January 31, 1989

- Letter from D.C. Hintz To NRC; Re: Follow-up Response To NRC Bulletin No. 88-04: Potential Safety-Related Loss; Dated July 8, 1988
- Modification No. DCR 3609; AFW Flow Control; Revision 2
- QF-0506(t); DCR 3609-2; AFW Flow Control; May 29, 2009
- QF-0509(t); DCR 3609-2; AFW Flow Control; February 22, 2011
- QF-0525(t); DCR 3609-2; AFW Flow Control; Revision 0
- TMod 2009-07; SI-350B Upstream Pipe Plug; Revision 1
- WO KW100550237; SI-350B; TMod 2009-007 (Install Pipe Plug Upstream Of SI-350B), Sump B; February 24, 2011

1R19 Post-Maintenance Testing

- GNP-03.01.01-1; Tracking And Processing Record for OP-KW-STP-AFW-002; March 21, 2011
- Implementation Summary Sheet For OSP-SI-005a, Revision 3; SI Pump 1A; March 14, 2011
- MA-KW-ICP-ICE-178; M&TE: 91794; Attachment D, Measuring And Test Equipment Calibration Record For 2000 psig; Dated March 21, 2011
- MA-KW-ICP-ICE-178; M&TE: 91795; Attachment D, Measuring And Test Equipment Calibration Record For 2000 psig (Post-Cal); Dated March 21, 2011
- MA-KW-ICP-ICE-178; M&TE: 91798; Attachment A, Measuring And Test Equipment Calibration Record For 30 psig; Dated March 21, 2011
- MA-KW-ICP-ICE-178; M&TE: 91799; Attachment A, Measuring And Test Equipment Calibration Record For 30 psig; Dated March 21, 2011
- MA-KW-ICP-ICE-178; M&TE: 91816; Attachment A, Measuring And Test Equipment Calibration Record For 30 psig; Dated March 21, 2011
- MA-KW-ICP-ICE-178; M&TE: 91824; Attachment A, Measuring And Test Equipment Calibration Record For 30 psig; Dated March 21, 2011
- MA-KW-MCM-DGM-005A; Governor Replacement For Train A Emergency Diesel Generator, System 10; Revision 4
- MA-KW-MCM-DGM-005A; Governor Replacement For Train A Emergency Diesel Generator, System 10; Revision 4
- MA-KW-MCM-DGM-005A; Governor Replacement For Train A Emergency Diesel Generator, System 10; Revision 2
- MA-KW-MCM-SI-001; Safety Injection Pump Inspection And Rebuild; March 10, 2011
- OP-KW-OSP-DGE-004A; Diesel Generator A Elevated Load And Load Rejection Test, System 42; Revision 15
- OP-KW-OSP-SI-001; Diesel Generator Automatic Test, System 33, Attachment A; Revision 7
- OP-KW-OSP-SI-001; Diesel Generator Automatic Test, System 33; Revision 7
- OP-KW-OSP-SI-006A; Train A Safety Injection Pump And Valve Test – IST, System 33; Revision 0
- OP-KW-STP-AFW-001; AFW Pump A, Pump Curve Development And Cavitating Venturi Validation; Procedure Performed March 21, 2011
- OP-KW-STP-AFW-002; Pump Curve Development And Cavitating Venturi Validation; Revision 0
- OP-KW-STP-AFW-005; MDAFW Pump Oil Cooler Tests And Min Flow Orifice Replacement; Revision 0
- SI Pump 1A Kewaunee PMT Test Data Sheet; March 17, 2011
- SOP-AFW-05B-30; TD AFW Pump Curve Development And Cavitating Venturi Validation; Procedure Performed March 25, 2011
- WO KW100276680; PM33-150: 15 Year Inspection-SI Pump 1A; March 3, 2011

- WO KW100770670; Unit 1; **Contingency** Replace D/G "A" Governor; Printed March 12, 2011
- WO KW100770670; Unit 1; **Contingency** Replace D/G "A" Governor; Printed March 14, 2011
- WO KW100778633; Check/Adjust Speed Setting Motor Clutch On The 1A EDG Governor; March 22, 2011

<u>1R20 Outage Activities</u>

- BKG ES-0.2; Natural Circulation Cooldown; Revision 3
- CM-AA-CRS-100; GSI-191 Program Standards, Requirements, And Guidance For The Containment Recirculation Sump; Revision 1
- CM-AA-CRS-102; Control Of Aluminum And Banned/Restricted Materials Inside Containment; Revision 2
- CR365378; Refueling Head Vent Issue
- CR418105; Procedure MA-KW-MCM-BLD-002 Enhancement
- CR418537; RCS Drain Down To 60% Pressurizer Level Stopped
- ES-0.2; Natural Circulation Cooldown; Revision 23
- Feedback Incorporation Process for MA-KW-MCM-BLD-002, Revision 2; Submitted March 17, 2011
- Fire Brigade Outage Schedule
- FSRC Request for Periodic Review Of Open Degraded And Non-Conforming SSCs; March 18, 2011
- GNP-08.12.02; Controls For Use Of Cranes Within The Protected Area; Revision 27
- Kewaunee Power Station Reactor Core Physical Inventory; March 14, 2011
- Kewaunee Unit 1 2011 RFO Shutdown Risk Review Report; February 21, 2011
- KPS USAR 4.1-1; Reactor Coolant System; Revision 22.04
- MA-AA-101; Fleet Lifting And Material Handling; Revision 5
- MA-AA-101; Fleet Lifting And Material Handling; Revision 5
- MA-AA-102; Foreign Material Exclusion; Revision 9
- MA-AA-OCR-101; Overhead Cranes/Hoists; Revision 2
- Maintenance Outage Schedule
- MA-KW-ICP-RC-012; Pressurizer Level Cold Calibration Transmitter 24029 Calibration, System 36; Revision 0
- MA-KW-ISP-RC-196A; Refueling Water Level Indication System Transmitter Calibration, System 36; Revision 0
- Operations Outage Schedule
- OP-KW-AOP-EHV-008; Loss Of All AC Power During Shutdown Conditions, System EHV-39; Revision 3
- OP-KW-AOP-RC-002; Abnormal Refueling Water Level, System RC-36; Revision 1
- OP-KW-AOP-RHR-001; Abnormal Residual Heat Removal System Operation, System RHR-34; Revision 5
- OP-KW-AOP-RHR-002; Shutdown Loss Of Coolant Accident, System RHR-34; Revision 5
- OP-KW-AOP-RHR-003; Loss Of RHR While Operating At Reduced Inventory Conditions, System RHR-34; Revision 2
- OP-KW-AOP-SFP-001; Abnormal Spent Fuel Pool Cooling And Cleanup System Operation, System SFP-21; Revision 3
- OP-KW-GOP-105; Startup From Mode 2 To 35% Power; Revision 9
- OP-KW-NOP-CCI-001; Containment Access, System 56; Revision 2
- OP-KW-NOP-RCS-005; Attachment C, Reactor Vessel Level Reference Sheet; Revision 10
- OP-KW-NOP-RCS-005; Attachment D, Decreased Inventory; Revision 10

- OP-KW-NOP-RCS-005; Draining The Reactor Coolant System, System 36; Revision 10
- OP-KW-NOP-RCS-005; Draining The Reactor Coolant System, System 36; Revision 10
- OP-KW-OSP-ESF-002; ESF Monthly Alignment Verification, System 55; Revision 2
- OP-KW-OSP-FH-001; Refueling – Containment Operability Surveillance, S/G Secondary Side Intact, System No. 53; Revision 0
- OU-AA-200; Shutdown Risk Management; Revision 2
- OU-AA-201; Shutdown Safety Assessment Checklist; Revision 2
- OU-KW-201; Attachment 1, Shutdown Safety Assessment (SSA) Checklist; March 21, 2011
- RF-02.01; Removal Of Head Ventilation, Missile Shield, And Seismic Restraints; Revision 13
- SP-36-267; ASME Boiler And Pressure Vessel Code Class 1 System Pressure Test; Revision 22

<u>1R22 Surveillance Testing</u>

- GNP-03.24.01; Job Briefs Implementation; Revision 16
- Kewaunee Power Station PRA Risk Summary; August 17, 2010
- OPERXK-100-19, Flow Diagram Component Cooling Water System; Revision AP
- OP-KW-NOP-DGM-001A; Diesel Generator A Remote Operations; Revision 4
- SP-31-168A; Train "A" Component Cooling Pump And Valve Test – IST; Performed January 13, 2011
- SP-34-099B; Train B RHR Pump And Valve Test IST; Performed February 15, 2011
- SP-48-003F, Revision 21; Nuclear Power Range Channel 2 (White) N-42 Monthly Test; Performed January 25, 2011
- SP-48-004G, Revision 15; Nuclear Power Range Channel 1 (Red) N41 Ion Current Calibration And [ITS] Channel Operational Test (COT); Performed February 17, 2011
- SP-55-167-8; (CTS) Hot/Intermediate Shutdown (ITS) Mode 3 Valve Tests; March 26, 2011

<u>2RS1 Radiological Hazard Assessment and Exposure Controls</u>

- CR391985; 2009 Refueling Outage Worker Entering High Radiation Area
- CR411448; RCS Co-58 Activity
- CR411978; Electronic Dosimeter Dose Rate Alarm
- CR415633; Dose Rate Alarm Received During ISIS Walkdown – CTMT 592'
- RP-AA-105; External Radiation Exposure Control Program
- RP-AA-201; Access Controls For High And Very High Radiation Areas; Revision 5
- RP-AA-225; Unrestricted Release Of Material; Revision 2
- RP-AA-232; Radioactive Material Control; Revision 1
- RWP And Associated ALARA Documents; RWP 11-0254; Refueling Work Of Reactor Head In And Around Reactor Cavity
- RWP And Associated ALARA Documents; RWP 11-0255; Core Offload/Reload And Associated Work
- RWP And Associated ALARA Documents; RWP 11-0263; Reactor Coolant Pump Work;

<u>4OA2 Identification and Resolution of Problems</u>

- ACE018382; Scaffold Built In Contact With Hanger WD-H191
- CA182544; Update MA-AA-105 To Address The Lack Of A Tie For Definition For Safety Related
- RCE-2008-0224; Scaffolding Affecting Safety-Related Equipment

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

- MA-AA-103; Attachment A; Revision 5
- Tracking And Processing Record For OP-KW-AOP-EHV-006; Loss Of Bus 6; March 10, 2011
- Switchyard High Risk Contingency Plan Actions; February 2, 2011
- CR 417078; Loss of Station Backfeed

NRC-Identified Condition Reports

- CR409336; 3 Cross-Cutting Aspects In The Area Of Human Performance - Documentation
- CR409338; 3 Cross-Cutting Aspects In The Area Of Human Performance – Proc Compliance
- CR409342; 3 Cross-Cutting Aspects In The Area Of Human Performance – Conserv Assump
- CR409739; Block 11 In LER 2009-006 Contained One Applicable Box That Was Not Checked
- CR410315; Conduct Performance Monitoring, PMP-18-13, On "C" CFCU
- CR411049; NRC Provided Feedback Regarding 3 ILT Candidates' Documentation Booklets
- CR411065; Critical Steps Designation Questioned By NRC For Reactor Protection SP's
- CR411162; Door 3 Lower Cane Bolt Was Found Not Engaged
- CR412540; Admin Building HVAC Chill Water Coil Leak
- CR412558; Implemented APC For Admin. Building AC Unit Chill Water Coil Leak
- CR412782; Hose Station #8 Hose Contacting The TD AFW Insulated Steam Supply Line
- CR414162; NRC Raises Question Regarding Fire Pump Starting Delays
- CR414311; Containment Response Analyses EPITOME Code May Be Nonconservative
- CR414345; Door 151 Closer May Need Adjustment
- CR414368; Sewage Treatment Plant Laboratory Certification Evaluation Report
- CR415539; NRC Concerns Identified During PJB For SI-350B Pipe Plug Installation
- CR415568; NRC Resident Identified A Concern With Installation Of Pipe Plug In CTMT Sump B
- CR415716; NRC Resident Questions Fall Protection On A RX Coolant Pump Vault Platform
- CR415801; Near Miss For Use Of Soluble Purge Dam For Replacement Of ICS-5A
- CR415888; Liquid Penetrant Examination Procedure Outside Of Standard Temperature Range
- CR415893; Procedure Qualification Questionable
- CR415894; UT Examination Procedures Appear To Be Lacking Specific Acceptance Standards
- CR416072; Updates Needed In The Area Summaries Of Pre-Fire Plans
- CR416126; Potable Water Backflow Preventer Inspection Beyond Annual Inspection Date
- CR416202; NRC Questioned Cavity Dip Sampling
- CR416325; Use Of Unauthorized Glue Stick For Purge Dam On ICS-5A
- CR416327; Inflatable Purge Dam Balloon Hose Came In Contact With Hot Pipe (ICS-5A)
- CR416329; CR350849 From KR30: BACC Question Not Answered Correctly
- CR417146; Light White Dry Boric Acid At Swagelok Fitting Near RC-602A
- CR417150; Light White Dry Boric Acid Deposit From Packing On RC-302A
- CR417186; INPO Debriefs A Poor Rad Worker Practice
- CR417307; Tape Identified In The Refueling Cavity
- CR417321; Reactor Cavity FME Plan Compliance Issues
- CR418288; Evaluate Hot Shutdown Boric Acid Walkdown Practices
- CR419368; Procedure Improvement Item Identified By NRC Senior Resident Inspector
- CR419921; Light Bulbs Needed Replacing On Bus 6 Bkr 1-608
- CR420044; BKR 16211 Trip Indication Lightbulb Burnt Out
- CR420050; Door One (1) Door Knob Sticking
- CR420067; Door 165 Mag Lock Prevents Full Closure Of Door

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Is-Reasonably-Achievable
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
BMV	Bare Metal Visual
CA	Corrective Action
CAP	Corrective Action Program
CCW	Component Cooling Water
CET	Core Exit Thermal
CR	Condition Report
CVC	Chemical and Volume Control
DET	Dominion Electrical Transmission
DG	Diesel Generator
DRP	Division of Reactor Projects
DW	Drywell
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
FIN	Finding
FP	Fire Protection
FW	Feedwater
IEL	Initiating Event Likelihood
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IR	Inspection Report
IR	Issue Report
ISI	Inservice Inspection
IST	Inservice Testing
ITS	Improved Technical Specifications
LER	Licensee Event Report
LLC	Limited Liability Corporation
LOI	Loss of Reactor Inventory
MAT	Main Auxiliary Transformer
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Testing
NCV	Non-Cited Violation
NDE	Non-Destructive Examinations
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PARS	Publicly Available Records System
PMT	Post-Maintenance Testing
POS	Plant Operating State
psig	Pounds Per Square Inch Gauge
PT	Dye Penetrant Testing

RAT	Reserve Auxiliary Transformer
RCE	Root Cause Evaluation
RCS	Reactor Coolant System
RFO	Refueling Outage
RHR	Residual Heat Removal
RS	Reactor Safety
RWP	Radiation Work Permit
RVCH	Reactor Vessel Closure Head
RVLIS	Reactor Vessel Level Indication System
SBLOCA	Small-Break Loss-Of-Coolant Accident
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SG	Steam Generator
SI	Safety Injection
SR	Safety-Related
SRA	Senior Reactor Analyst
SW	Service Water
TMOD	Temporary Modification
TS	Technical Specification
TSC	Technical Support Center
TSO	Transmission System Operator
URI	Unresolved Item
USAR	Updated Safety Analysis Report
UT	Ultrasonic Testing
WO	Work Order

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Sincerely,

/RA/

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

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