July 18, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Plant
Nuclear Management Company, LLC
N490 Hwy 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT
NRC INTEGRATED INSPECTION REPORT 50-305/03-04

Dear Mr. Coutu:

On June 30, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Kewaunee Nuclear Power Plant. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 2, 2003, with you and 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, there was one NRC-identified and three self-revealed findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because these violations were non-willful and non-repetitive and because they were entered into your corrective action program, the NRC is treating this findings as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Kewaunee facility.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calender year 2002 and the remaining inspection activities for Kewaunee are scheduled for completion in November 2003. The NRC will continue to monitor overall safeguards and security controls at the Kewaunee Nuclear Power Plant.

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

Sincerely,

/RA/

Patrick Louden, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 50-305/03-04
w/Attachment: Supplemental Information

cc w/encl: D. Graham, Director, Bureau of Field Operations
Chairman, Wisconsin Public Service Commission
State Liaison Officer

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No.:	50-305
License No.:	DPR-43
Report No.:	50-305/03-04
Licensee:	Nuclear Management Company, LLC
Facility:	Kewaunee Nuclear Power Plant
Location:	N 490 Highway 42 Kewaunee, WI 54216
Dates:	April 1 through June 30, 2003
Inspectors:	J. Lara, Senior Resident Inspector (Outgoing) R. Krsek, Senior Resident Inspector (Incoming) Z. Dunham, Resident Inspector (Outgoing) R. Berg, Resident Inspector (Incoming) D. Jones, Reactor Inspector T. Madeda, Physical Security Inspector D. Nelson, Radiation Specialist M. Garza, Nuclear Safety Intern
Approved By:	Patrick Louden, Chief Reactor Projects Branch 5 Division of Reactor Projects

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ATTACHMENT: Supplemental Information

SUMMARY OF FINDINGS

IR 05000305-03-04; 04/01/2003 - 06/30/2003; Kewaunee Nuclear Power Plant; Refueling and Outage Activities; Fitness-for-Duty Program

This report covers a 3-month period of baseline resident inspection and announced baseline inspections for radiation protection, inservice inspection and Temporary Instruction (TI) 2515/150, and physical protection. The inspections were conducted by the resident inspectors, and Region III inspectors. One NRC identified and three self-revealed Green findings associated with non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process 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 3, dated July 2000.

A. NRC-Identified and Self-revealed Findings

Cornerstone: Initiating Events

- Green. A self-revealed, non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified for the licensee's failure to properly sequence a tagout in accordance with the licensee's tagout procedure. This resulted in an approximate 100-gallon loss of inventory from the reactor coolant system. A contributing cause of this finding was related to the cross-cutting area of Human Performance.

  This finding is greater than minor because it affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding is of very low risk significance because none of the checklist attributes of Inspection Manual Chapter 0609, "Shutdown Operations Significance Determination Process," Appendix G, were affected. (Section 1R20.3)

- Green. A self-revealed, non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified for the licensee's failure to ensure that the procedure governing refueling operations and reactor head disassembly had appropriate instructions or cautions to ensure that the reactor head vent remained vented to containment atmosphere. This resulted in the reactor head vent not being vented and affecting the operation of the refueling cavity water level instrument which operators were using to control reactor vessel water level.

  This finding is greater than minor because it is a configuration control issue which affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding is of very low risk significance because none of the checklist attributes of Inspection Manual Chapter 0609, "Shutdown Operations Significance Determination Process," Appendix G, were affected. (Section 1R20.4)

Cornerstone: Mitigating Systems

- Green. A self-revealed non-cited violation 10 CFR 50, Appendix B, Criterion V, was identified for the licensee's failure to ensure that the residual heat removal pump recirculation piping material was in accordance with a facility drawing and engineering specifications. This resulted in the corrosion of three pipe plugs, one of which was corroded to the point of leaking. The pipe plugs were installed on each residual heat removal's recirculation pipe pressure breakdown orifice. The three pipe plugs were made of carbon steel while the residual heat removal system piping, which contained borated water, was required to be made of stainless steel.

  This finding was greater than minor because it affected the Mitigating System Cornerstone objective of equipment reliability and availability, in that the failure to ensure that the residual heat removal piping materials are in accordance with plant engineering specifications and drawings could result in system leakage significant enough to require taking the system out-of-service. The finding is of very low risk significance because this finding was not a design or qualification deficiency which resulted in a loss of function per Generic Letter 91-18. (Section 1R20.2)

Cornerstone: Physical Protection

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 26 Fitness-for-Duty (FFD) reporting requirements. The licensee failed to notify the NRC Operation Center within 24 hours of discovery of an illegal drug found within the licensee's protected area. The licensee failed to report the event because they did not realize this type of event was required to be reported.

  The finding was determined to be of very low significance because it was a vulnerability in the licensee's Safeguards plan, was not a malevolent act, and similar findings had not occurred in the last four calendar quarters. The finding was determined to be more than minor because illegal drugs located within a licensee's protected area are required to be reported to the NRC in accordance with 10 CFR 26.73(a) requirements. (Section 4OA3)

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

Summary of Plant Status

The plant was operated at approximately 100 percent power until April 5, 2003, when the plant was shutdown to commence the 2003 refueling outage. The plant was started and returned online on May 11, and achieved 100 percent power on May 18. The plant was operated at approximately 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Technical Support Center (TSC) Diesel Generator (DG)

a. Inspection Scope

The licensee has credited the TSC DG as the alternate alternating current DG for station blackout mitigation, the inspectors reviewed the TSC DG for its susceptibility to hot weather conditions and high winds. On May 16, 2003, the inspectors walked down the TSC DG and the associated support equipment to verify that the material condition of the equipment was acceptable. The inspectors reviewed the licensee's operating procedures, including Procedure RT-DGM-10-TSC, "Technical Support Center Diesel Generator," Revision W, to ensure the procedures prescribed appropriate instructions for proper operation of the TSC DG.

The inspectors also reviewed the Updated Safety Analysis Report, Section 8.2.4, "Station Blackout," Revision 17, to verify that the design requirements of the TSC DG would be met under adverse weather conditions. Additionally, the inspectors reviewed the TSC DG vendor manual, "Western Engine Co. Model: TSC Diesel Generator," Revision 0, to determine whether there were any operational or design requirements associated with hot weather that were not already incorporated into the licensee's procedures.

b. Findings

No findings of significance were identified.

.2 Substation Switchyard Walkdown

a. Inspection Scope

On May 30, 2003, the inspectors walked down the site substation switchyard to determine whether the area was maintained and ready for seasonal adverse weather conditions which include potentially high winds and tornados. Initiating events involving a loss of offsite power contributed up to 55 percent of the facility's core damage frequency, and therefore the substation switchyard was considered to be a potential risk significant area. The inspectors evaluated the housekeeping of the area, interviewed electrical maintenance personnel, and a reviewed the Updated Safety Analysis Report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns

The inspectors performed three partial system walkdowns during this inspection period. On April 11, 2003, the inspectors walked down the 'B' train DG, reserve auxiliary transformer, tertiary auxiliary transformer, and substation while the 'A' train DG was out-of-service for the refueling outage overhaul. On May 22, 2003, the inspectors walked down the 'A' component cooling water train and associated support equipment while the 'B' component cooling water train was out-of-service for a scheduled surveillance test. On June 13, 2003, the inspectors walked down the Control Room Post-Accident Recirculation System following realignment of the system after testing which was performed for troubleshooting.

To evaluate the operability of the selected equipment, the inspectors reviewed the system lineup checklists, normal operating procedures, abnormal and emergency operating procedures (EOPs), and current system drawings to verify the correct system lineup. During the walkdowns, the inspectors also examined valve positions and electrical power availability to verify that valve and electrical breaker positions were consistent with, and in accordance with, the licensee's procedures and design documentation. The material condition of the equipment was also inspected.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Zone Inspections

a. Inspection Scope

During the inspection period, the inspectors walked down portions of the nine areas listed below to assess the overall readiness of fire protection equipment and barriers:

- Accessible levels and areas of containment (Fire Zone RC-60) - April 9, 2003;
- Accessible areas of steam generator and pressurizer vaults (Fire Zone RC-60) - April 9, 2003;
- Turbine building basement and mezzanine (Fire Zone TU-22) - April 17, 2003;

- Accessible levels and areas of containment (Fire Zone RC-60) - April 30, 2003;
- Safeguards battery rooms ‘A’ and ‘B’ - (Fire Zones TU-97 and TU-98) May 21, 2003;
- Turbine building screenhouse (Fire Zone SC-70A) - May 21, 2003;
- Relay room (Fire Zone AX-30) - June 24, 2003;
- Control room and ventilation room (Fire Zone AX-35) - June 24, 2003; and
- Control rod drive equipment room (Fire Zone AX-77) - June 27, 2003.

The inspectors focused on the licensee’s control of transient combustibles and ignition sources, the material condition of fire protection equipment, and the material condition and operational status of fire barriers used to mitigate fire damage or propagation. Additionally, fire hoses, sprinklers, and portable fire extinguishers were inspected to verify that they were in satisfactory physical condition and were unobstructed. Passive features such as fire doors, fire dampers, and fire zone penetration seals were also inspected to verify that they were in satisfactory condition and capable of providing an adequate fire barrier.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Review of Internal Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the internal flooding analysis for two areas during this inspection period. During the week of April 14, 2003, the inspectors performed a walkdown of flood zone 156-6, ‘B’ component cooling water pump room. On May 14, 2003, the inspectors performed a walkdown of flood zone 3B, ‘B’ DG room. These areas were walked down to determine whether existing configurations and mitigation plans were consistent with design requirements and risk analysis assumptions. The inspectors focused on the material condition and operational status of flood barriers used to mitigate flood damage and propagation.

Flood protection features such as flood doors and door gaps, drains, and flood zone penetration seals were also inspected to verify that they were in satisfactory physical condition, unobstructed, and capable of providing an adequate flood barrier. For those cases in which floor penetrations were not sealed, the inspectors verified that flooding leakage would not prevent the fulfillment of any safety function or adversely impact the operation of mitigating equipment. The inspectors also reviewed the maximum expected water levels in the pump room; calculated the resultant forces on the flood zone door; and estimated the shear strength of the door latches to ensure the door would remain intact during the postulated flooding scenario.

b. Findings

No findings of significance were identified.

.2 Review of External Flood Protection Measures

On May 14, 2003, the inspectors reviewed the design drawings for the facility "trenwa" to determine the susceptibility of the "trenwa" to flooding and the acceptability of its drainage capability. The "trenwa" was a covered concrete lined trench, which ran from the substation to the protected area, which contained control cables for various components located outside the protected area and in the substation. On May 30, the inspectors selected two locations in the "trenwa" for inspection and conducted a visual examination of the overall material condition of the "trenwa" and the cabling located inside. Additionally, the inspectors reviewed Procedure E-0-05, "Natural Disaster," Revision J, which prescribed licensee actions in the event of flood warnings at the site. The inspectors walked down the prescribed actions to verify that the actions could be performed.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

On May 12, 2003, the licensee conducted heat exchanger performance monitoring on the Containment Fan Cooling Units. The inspectors reviewed the test procedure, and reviewed the test data to verify that the test was performed as written, that the acceptance criteria were adequate to demonstrate acceptable heat transfer capability of the heat exchanger, and that the test data met the acceptance criteria. The inspectors also verified that the test accounted for instrument inaccuracies and that the test frequency was sufficient to provide early detection of heat exchanger degradation prior to any loss of heat removal capabilities below design values. Finally, the inspectors compared the current test results with previous test data to verify the performance of the heat exchangers.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's inservice inspection (ISI) program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries. Specifically, the inspectors observed activities in-process and reviewed records of nondestructive examinations.

The inspectors observed:

- Mechanized ultrasonic testing (UT) examination of steam generator 1B, feedwater pipe to nozzle weld FW-59; and
- Liquid penetrant (PT) examination of safety injection socket weld SI-W29S.

The inspectors also reviewed the following reports:

- liquid penetrant examination of weld CVC-W90S;
- liquid penetrant examination of weld RTD-W61B; and
- Kewaunee R26 Eddy Current Summary Report.

These examinations were evaluated for compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspectors also reviewed ISI procedures, personnel certifications, and NIS-2 forms for Code repairs performed during the last outage to confirm that ASME Code requirements were met.

The inspectors reviewed a sample of ISI related problems documented in the licensee's corrective action program (CAP), to assess conformance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In addition, the inspectors reviewed the licensee's evaluation of operating experience for its applicability to the ISI program.

b. <u>Findings</u>

No findings of significance were identified.

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

a. <u>Inspection Scope</u>

On May 20, 2003, the inspectors observed a simulator dynamic requalification exam to evaluate crew performance, formality of communications, and annunciator response. Additionally, the inspectors evaluated the operation crew's implementation of the facility's abnormal and emergency operating procedures, oversight and direction provided by the shift manager and control room supervisor, and the adequacy of identification and reporting of the event classification in accordance with the facility's emergency plan. The inspectors also compared the simulator board configuration with the actual control room board configuration for consistency between the two to ensure that the simulator environment matched the actual control room environment as closely as possible. The inspectors observed the post-scenario critique to determine whether performance issues were accurately identified and addressed.

b. <u>Findings</u>

No findings of significance were identified.

1R12 <u>Maintenance Effectiveness</u>

a. Inspection Scope

During the week of May 5, 2003, the inspectors reviewed the implementation of the maintenance rule for the residual heat removal (RHR) system to verify that component and equipment failures were identified, entered, and scoped within the maintenance rule and that the system was properly categorized and classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed station logs, maintenance work orders, action requests, functional failure evaluations, unavailability records, and a sample of corrective action reports to verify that the licensee was identifying issues related to the maintenance rule at an appropriate threshold and that corrective actions were appropriate. Additionally, the inspectors reviewed the licensee's performance criteria to verify that the criteria adequately monitored equipment performance.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

During the inspection period, the inspectors reviewed the licensee's evaluation and assessment of plant risk, scheduling, and configuration control for the five work-week schedules listed below. In particular, the licensee's planning and management of maintenance was evaluated to verify that on-line risk was acceptable and in accordance with the requirements of 10 CFR 50.65(a)(4). Additionally, the inspectors compared the assessed risk configuration against the actual plant conditions and any in-progress evolutions or external events to verify that the assessment was accurate, complete, and appropriate. Licensee actions to address increased on-line risk during these periods were also inspected to verify that actions were in accordance with approved administrative procedures.

- Shutdown safety assessment for April 7 through 11, 2003;
- Shutdown safety assessment for April 21 through 25, 2003;
- Safety Monitor risk assessment for May 19 through 23, 2003;
- Safety Monitor risk assessment for June 9 through 13, 2003; and
- Safety Monitor risk assessment for May 16 through 20, 2003.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

.1 Personnel and Equipment Response to Loss of Reactor Coolant System (RCS) Inventory

a. Inspection Scope

On April 9, 2003, the inspectors reviewed the control room operators response to an inadvertent loss of RCS inventory which occurred when a tagout associated with the ‘A’ Reactor Coolant Pump (RCP) was sequenced improperly resulting in a loss of approximately 100 gallons from the RCS. The plant was in refueling shutdown mode at the time of the event. The inspectors interviewed operators, reviewed plant computer data and trend recorders, reviewed operator logs, and evaluated equipment response to determine whether the operator’s response was in accordance with plant procedures. Section 1R20.3 of this report documents additional issues associated with this event.

b. Findings

No findings of significance were identified.

.2 Refueling Outage Startup Observations

a. Inspection Scope

The inspectors observed the operators fill and vent the RCS following refueling of the reactor on May 3, 2003, and observed operators start the main turbine on May 10, 2003, to verify that plant procedures were followed and that control room activities were conducted in accordance with plant procedures.

b. Findings

No findings of significance were identified.

.3 Personnel and Equipment Response to a Rod Stop-Turbine Runback Signal

a. Inspection Scope

On June 10, 2003, the inspectors observed and reviewed the control room operators’ response to a rod stop-turbine runback signal which occurred during testing of Channel 1 (Red) of the Reactor Protection System. The inspectors interviewed operators and instrumentation and control technicians, reviewed plant computer data and trend recorders, reviewed operator logs, and evaluated the equipment response to determine whether the operators’ response was in accordance with plant procedures.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the four licensee corrective action documents listed below describing equipment questions and issues for risk significant systems to determine the acceptability of the associated operability evaluations. The inspectors reviewed applicable design information and Technical Specifications to assess the adequacy of the evaluations and implementation of applicable compensatory measures. The inspectors' review included verification that the operability determinations were made in accordance with Procedure GNP 11.08.03, "Operability Determination," Revision B.

- CAP 15459; Unexpected response during SP-42-312A "DG Availability Test";
- CAP 16003; 'B' DG heat exchanger tube bundle nuts not staked;
- CAP 15980; SI-5B-1 relief valve operability; and
- CAP 16788; Effect of quarterly calibration on the red channel nuclear instrumentation system and monthly calibration on remaining channels.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (OWAs)

.1 Control Room Deficiency Tags

a. Inspection Scope

On June 27, 2003, the inspectors reviewed the licensee's control room deficiency log, out-of-service control room indicators and equipment logs to verify the licensee's identified operator workarounds. The inspectors also reviewed posted danger cards located in the control room to assess whether known degraded or out-of-service equipment in the control room would impact operator response to plant transients or emergencies and therefore be considered potential operator workarounds.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the engineering analyses, design information and modification documentation for the replacement of the component cooling water heat exchangers during the facility's refueling outage. Additionally, the inspectors observed portions of the removal and installation of the heat exchangers, reviewed acceptance testing results, and reviewed condition reports associated with the design change to verify that the licensee was identifying and documenting problems at an appropriate threshold.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance testing activities associated with the five scheduled and emergent work activities listed below to verify that the testing was adequate for the scope of the maintenance work performed. The inspectors reviewed the tests' acceptance criteria to ensure that the criteria was clear and that testing demonstrated operational readiness consistent with the design and licensing basis documents. The inspectors attended pre-job briefings, when possible, to verify that the impact of the testing had been properly characterized; and observed or reviewed the test to verify that the test was performed as written and that all testing prerequisites were satisfied. Following the completion of each test, the inspectors completed walkdowns of the affected equipment, when applicable, to verify that the test equipment was removed and that the equipment was returned to a condition in which it could perform its safety function. The inspectors also reviewed the completed test data to ensure the test acceptance criteria were met for the following five activities:

- Replacement of DG speed changer motor - April 16, 2003;
- Valve SI-350B motor-operator preventative maintenance - April 17, 2003;
- Auxiliary feedwater pump discharge check valve replacement - May 9, 2002;
- Main steam isolation valve train 'A' closure spring replacement - May 12, 2003; and
- Channel 3 (Blue) Instrument Channel TM-4030 input output gain adjustments - June 11, 2003.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Routine Refueling and Outage Inspection Activities

a. Inspection Scope

The inspectors reviewed the site outage plan and schedule of maintenance, surveillance, and refueling activities for the refueling outage, conducted from April 5 to May 11, 2003, to ensure that the licensee had adequately considered qualitative plant risk for various plant configurations, and had appropriately incorporated industry refueling outage operating experience. During the refueling outage, the inspectors monitored the activities listed below to ensure that the licensee maintained an adequate defense-in-depth strategy commensurate with the site outage plan, maintained compliance with technical specification requirements, and implemented adequate risk management tools in response to emergent work and maintenance issues.

The activities reviewed were as follows:

- Shutdown activities and cooldown rates;
- Tagout clearances and equipment lineups;
- Reactor coolant system pressure, level, and temperature instrumentation configuration;
- Electrical lineups, to verify that they were in accordance with outage risk plan and technical specifications;
- Decay heat removal parameters to ensure that the system was functioning properly;
- Periodic walkdowns of spent fuel pool cooling system;
- Alignment of systems used for reactor coolant system inventory addition;
- Containment penetration controls, to verify that they were in accordance with technical specifications;
- Refueling activities and fuel movement, to ensure that technical specifications were met and that site refueling procedures were followed;
- Plant startup and heatup, including walkdowns of containment to ensure that debris had not been left which could affect performance of containment sumps; and
- Licensee identification and resolution of outage related problems.

b. Findings

No findings of significance identified.

.2 Failure to Maintain Design Requirements of RHR System

a. Inspection Scope

The inspectors reviewed the licensee's response and followup investigation following the discovery that piping components installed in the RHR system were made of carbon steel and not stainless steel as required by facility engineering specifications.

b. Findings

Introduction

A Green self-revealed finding associated with a NCV was identified for the licensee's failure to ensure that the residual heat removal system piping materials were in accordance with plant design specifications as required by 10 CFR 50, Appendix B, Criterion V.

Description

On April 5, 2003, in preparation for cooling the reactor coolant system below 350 degrees Fahrenheit during the refueling outage, control room operators placed the residual heat removal system in service. Shortly afterwards, the operators were notified that a leak had developed on a plug located in a pressure tap connection associated with the 'B' train RHR pump recirculation line pressure breakdown orifice. Upon receipt of this information, the operators secured both trains of RHR. The licensee determined that the plug material was of carbon steel and that it had corroded due to the presence of boric acid in the RHR system fluid. The RHR system normally contains borated water to ensure that the reactor is subcritical during accident conditions.

The licensee documented the issue in CAP 15530. The licensee determined, as documented in Apparent Cause Evaluation 2221, that two other carbon steel plugs were installed in the ‘A’ RHR pump recirculation piping. The licensee concluded that the additional carbon steel plugs were also corroded, but not to the extent such that near term leakage was likely. The licensee reviewed the maintenance history for the RHR pump recirculation piping. The licensee did not determine when the carbon steel plugs had been installed nor identify any maintenance work packages which affected the pipe plugs.

Additionally, the licensee conducted an extent of condition review of other systems which contain borated water. The review consisted of examining readily accessible piping located in the penetration rooms, chemical and volume control system piping, safety injection system piping, containment spray system piping, and uninsulated portions of the boric acid pumps. No additional carbon steel plugs were identified. Drawing XK-100-18, “Flow Diagram Auxiliary Coolant System,” Revision AK, required that RHR system recirculation piping be made of stainless steel. The licensee subsequently replaced all three carbon steel plugs with stainless steel plugs and returned the RHR system to service.

<u>Analysis</u>

The inspectors determined that the licensee’s failure to maintain configuration control by ensuring that the pipe plugs installed in the RHR pump recirculation piping were stainless steel in accordance with Drawing XK-100-18 was a performance deficiency warranting a significance evaluation. This self-revealed finding was greater than minor because the failure to maintain the design of the RHR system affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of a system designed to respond to an initiating event.

Although the reactor was shutdown at the time of this event, the finding was evaluated using Appendix A of Inspection Manual Chapter (IMC) 0609, “Significance Determination Process (SDP) of Reactor Inspection Findings for At-Power Situations.” The inspectors considered this to be acceptable since Appendix G of IMC 0609, “Shutdown Operations Significance Determination Process,” was applicable only once RHR cooling had been initiated and the RHR system had remained in this configuration during power operation over the previous at-power cycle.

The inspectors determined that the finding was of very low safety significance (Green) because the finding represented a design or qualification deficiency which did not result in a loss of function per Generic Letter 91-18, “Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions,” Revision 1.

Enforcement

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedure, or drawings, and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to this, as of April 5, 2003, the licensee failed to ensure that pipe plugs installed in both trains RHR pump recirculation piping pressure breakdown orifices were stainless steel in accordance with Drawing XK-100-18. Therefore, the inspectors determined this finding was a violation of 10 CFR 50, Appendix B, Criterion V. Because this violation was of very low safety significance and was documented in the licensee's corrective action program as CAP 15530, it is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 50-305/03-04-01)

.3 Improper Tagout Sequence Results in Loss of RCS Inventory

a. Inspection Scope

The inspectors reviewed the licensee's response and investigation into an inadvertent loss of RCS inventory which occurred when a tagout for the 'A' Reactor Coolant Pump (RCP) was not processed in accordance with plant procedures. The inspectors interviewed licensee personnel on actions taken and reviewed plant parameters following the event to ensure that core cooling was not challenged. (See also Section 1R14.1 of this report.)

b. Findings

Introduction

A Green self-revealed finding associated with a NCV was identified for the licensee's failure to properly sequence a tagout associated with isolation of the 'A' RCP in accordance with the licensee's tagout procedure. The improper tagout sequence resulted in a loss of inventory from the RCS.

Description

On April 9, 2003, with the plant in refueling shutdown mode and reactor vessel level at 20.6 percent, the licensee was in the process of hanging Tagout 03-266 when control room operators noted that reactor vessel level was decreasing. The operators took immediate action to stabilize and maintain reactor vessel level. The minimum reactor vessel level noted during the transient was 20.4 percent. Tagout 03-266 was hung to isolate the 'A' RCP in preparation for replacement later in the outage. A licensee investigation revealed that the loss of RCS inventory was due to the opening of the 'A' RCP vent and drain valves per Tagout 03-266 without the RCP shaft lowered and backseated to provide an RCS leakage boundary. This configuration resulted in a loss of approximately 100 gallons from the RCS to the 'A' containment sump.

The licensee's outage schedule had sequenced the tagout to be placed after the 'A' RCP was backseated. However, this information was not communicated to the work control center supervisor who authorized Tagout 03-266 to be performed. Additionally, the licensee's review of the event revealed that the work control supervisor who authorized Tagout 03-266 was uncertain if the plant conditions were acceptable for venting and draining the RCP. The inspectors also noted that Procedure GNP 03.03.01, "Tagout Processing," Revision N, provided conditions to be considered in preparing equipment for isolation including draining and venting. The licensee documented the issues in their corrective action program as CAP 15628. Additionally, the licensee conducted Root Cause Evaluation 611 to determine the underlying causes and issues associated with this event and a similar loss of inventory event which occurred later during the refueling outage (see Section 1R20.4 of this report, Inadvertent Installation of Blank Flange as a Foreign Material Exclusion Cover Results in Conflicting Reactor Vessel Level Indications, for more details concerning this second issue).

Analysis

The inspectors determined that the licensee's failure to sequence Tagout 03-266 without appropriate consideration to plant conditions and the impact of opening the 'A' RCP vent and drain valves which resulted in the reactor vessel level transient was a performance deficiency warranting a significance evaluation. This self-revealed finding was greater than minor because the failure to properly hang Tagout 03-266 affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined that a contributing cause of this event was related to the cross-cutting area of human performance.

Using the "PWR Cold Shutdown and Refueling Operation RCS Open and Refueling Cavity Level < 23' OR RCS Closed and No Inventory in Pressurizer Time to Boiling Less Than 2 Hours," checklist in Appendix G of IMC 0609, "Shutdown Operations Significance Determination Process," the inspectors determined that the finding was of very low safety significance (Green) because the finding did not affect the checklist attributes and therefore did not warrant a Phase 2 SDP evaluation.

Enforcement

Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality shall be prescribed by documented instructions, procedure, or drawings, and shall be accomplished in accordance with these instructions, procedures, or drawings. Licensee procedure GNP 03.03.01, "Tagout Processing," Revision N, Step 6.1.6.d, required, in part, that the Work Control Supervisor shall evaluate the conditions to be considered in preparing equipment for isolation including, as a minimum, draining and venting. Contrary to this requirement, on April 9, 2003, the Work Control Supervisor failed to evaluate the conditions to be considered in preparing the RCP for isolation, including draining and venting. Subsequently, licensee personnel hung Tagout 03-266 to isolate, vent and drain the 'A' RCP prior to the RCP being backseated, which resulted in a loss of RCS inventory. The inspectors determined this finding was a violation of 10 CFR 50, Appendix B, Criterion V. Because this violation was of very low safety significance (Green) and was documented in the licensee's corrective action program as CAP 15628, it is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 50-305/03-04-02)

.4 Inadvertent Installation of Blank Flange as a Foreign Material Exclusion Cover Results in Conflicting Reactor Vessel Level Indications

a. Inspection Scope

The inspectors reviewed the licensee's response and investigation into conflicting reactor vessel level indications which occurred when a blank flange was inadvertently installed on the reactor vessel head vent. The inspectors interviewed plant personnel and reviewed plant parameters following the event to verify that core cooling was not challenged.

b. Findings

Introduction

A Green self-revealed finding associated with a NCV was identified for the licensee's failure to ensure that the procedure governing refueling operations and reactor head disassembly had appropriate instructions or cautions to ensure that the reactor head vent remained vented to containment atmosphere thereby ensuring proper operation of the refueling cavity water level instrument.

Description

On April 10, 2003, control room operators noted a decreasing trend on Reactor Vessel Level Indication System (RVLIS) and an increasing trend of refueling cavity water level indication. Both indications were used by the operators to monitor current reactor vessel level indication which was initially at 20.6 percent; however, the inspectors noted that operators typically utilized refueling cavity water level indication to control reactor vessel level. The inspectors noted that there were two trains of RVLIS indication available to the operators, while there was only a single train of refueling cavity water level indication. During subsequent troubleshooting efforts of the differing level indications, the licensee determined that following the removal of the reactor vessel head vent spoolpiece, which had been removed in support of refueling activities per Procedure RF-01.00, "KNPP Refueling Procedure," that a metal blank flange was installed on the vessel head vent as a foreign material exclusion cover consistent with the licensee procedure on general foreign material exclusion practices.

The blank flange prevented the venting of non-condensible gases which had come out of solution from the RCS, and caused a gas pocket to form under the reactor vessel head. With a gas pocket forming, refueling cavity water level indicated an increasing trend since water displaced from the gas pocket caused an increased differential pressure on the instrument which resulted in a higher indicated level than actual reactor vessel level. Concurrently, RVLIS was not affected by the increased differential pressure and accurately indicated a decreasing vessel level due to the gas pocket which was forming. While the vessel level indications were in question, the operators controlled vessel level based on RVLIS which indicated the lower vessel level. At no time did the gas pocket interfere with the capabilities of plant systems to remove decay heat from the reactor.

After the licensee discovered that the blank flange had been installed, the blank flange was removed and the vessel head was vented returning the refueling cavity water level indicator to a normal configuration. During the evolution, reactor vessel level was noted to have lowered to approximately 20.5 percent. The licensee documented the transient in their corrective action program as CAP 15673. Additionally, the licensee conducted Root Cause Evaluation 611 to determine the factors and causes which led to the configuration control problem of the vessel head vent path.

Analysis

The inspectors determined that the licensee’s failure to have appropriate instructions or cautions in Procedure RF-01.00 to ensure that the reactor vessel head vent remained vented to atmosphere was considered a licensee performance deficiency warranting a significance evaluation. This self-revealed finding was greater than minor because the failure to ensure that the vessel head remained vented to atmosphere was considered a configuration control issue of an instrument the licensee relied upon for maintaining proper reactor vessel level. It also affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

Using the “PWR Cold Shutdown and Refueling Operation RCS Open and Refueling Cavity Level < 23' OR RCS Closed and No Inventory in Pressurizer Time to Boiling Less Than 2 Hours,” checklist in Appendix G of IMC 0609, “Shutdown Operations Significance Determination Process,” the inspectors determined that the finding was of very low safety significance (Green) because the finding did not affect the checklist attributes and therefore did not warrant a Phase 2 SDP evaluation.

Enforcement

Title 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” required, in part, that activities affecting quality shall be prescribed by documented instructions, procedure, or drawings of a type appropriate to the circumstances. Contrary to this requirement, Procedure RF-01.00, “KNPP Refueling Procedure,” Step 5.2.8, Revision J, directed removal of the reactor vessel head vent spool piece, but did not provide additional instructions or cautions to ensure that the reactor vessel head vent would remain unblocked and vented to containment atmosphere; therefore, the procedure was not appropriate to the circumstances. The inspectors determined this finding was a violation of 10 CFR 50, Appendix B, Criterion V. Because this violation was of very low safety significance (Green) and was documented in the licensee’s corrective action program as CAP 15673, it is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 50-305/03-04-03)

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and reviewed the surveillance testing results for the four systems listed below to verify that the equipment was capable of performing the intended safety function and that the surveillance tests satisfied the requirements contained in Technical Specifications and the licensee's procedures. The inspectors reviewed the surveillance tests to verify the tests were adequate to demonstrate operational readiness consistent with the design and licensing basis documents, and that the testing acceptance criteria were well documented and appropriate to the circumstances.

Portions of the test were observed to verify the test was performed as written, that all testing prerequisites were satisfied, and that the test data was complete, appropriately verified, and met the requirements of the testing procedure. Following the completion of the tests, when applicable, the inspectors conducted walkdowns of the affected equipment to verify that the test equipment was removed and that the effected equipment was returned to an operable condition.

- Residual Heat Removal Valve Interlock Test - April 6, 2003;
- Diesel Generator Automatic Test - April 8, 2003;
- Safety Injection Flow Test - May 3, 2003; and
- Diesel Generator B Availability Test - June 13, 2003.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

.1 Plant Walkdowns and Radiological Boundary Verification

a. Inspection Scope

The inspectors conducted walkdowns of selected radiologically controlled areas within the plant to verify the adequacy of radiological boundaries and postings. Specifically, the inspectors walked down several radiologically significant work area boundaries (high and locked high radiation areas) in the containment and auxiliary buildings. The inspectors performed confirmatory radiation measurements to verify that these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and the Technical Specifications.

b. Findings

No findings of significance were identified.

.2 High Radiation Area and Very High Radiation Area Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures, practices and associated documentation for the control of access to radiologically significant areas (high, locked high, and very high radiation areas) and assessed compliance with Technical Specifications, procedures, and the requirements of 10 CFR 20.1601 and 20.1602. In particular, the inspectors reviewed the licensee's practices and records for the control of keys to locked high radiation areas (LHRAs) and very high radiation areas (VHRAs), the use of access control guards to control entry into such areas, and the licensee's methods for independently verifying proper closure and latching of LHRA and VHRA doors upon area egress. Additionally, radiological postings were reviewed, and access control boundaries were challenged by the inspectors throughout the plant to verify that high, locked high, and very high radiation areas were properly controlled.

b. Findings

No findings of significance were identified.

.3 Review of Radiologically Significant Work

a. Inspection Scope

The inspectors reviewed the radiation work permits (RWP) and attended the pre-job, as-low-as-is-reasonably achievable (ALARA) briefing for the replacement of the 1A Reactor Coolant Pump during the 2003 refueling outage. The inspectors also reviewed selected 2003 refueling outage RWPs associated with preventive maintenance on the 1A and 1B Reactor Coolant Pump motors, activities associated with the In-Service Inspection Program, and activities associated with normal refueling outage reactor maintenance. These inspection activities were performed to verify the adequacy of surveys, access controls, and postings to assess the exchange of work area radiological information and to evaluate radiation worker and radiation protection technician performance. The inspectors also evaluated the licensee's procedure and practices for dosimetry placement and use of multiple dosimetry in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201 and applicable Regulatory Guides.

b. Findings

No findings of significance were identified.

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

.1 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed the status of the station's source term reduction program in order to verify that the licensee had an effective program in place, was knowledgeable of plant source term reduction opportunities, and to verify that efforts were being taken to address them. The inspectors also reviewed the station's overall source term reduction plan, which included improved tracking/mitigation of hot spots and tracking/trending of online/shutdown chemistry initiatives. The inspectors reviewed the licensee's continuing source term reduction techniques to verify that source term control strategies were ongoing and future initiatives were being explored.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors reviewed the licensee's use of ALARA controls for selected 2003 refueling outage work activities performed in radiation areas, HRAs, and LHRA. Specifically, the inspectors reviewed the adequacy of RWPs, radiological surveys, attended pre-job radiological briefings, and assessed job site ALARA controls for the following work activities:

- Assembling and Building Scaffolding Within Containment;
- Work Activities in Support of In Service Inspections; and
- Disassemble and Removal of the Reactor Head.

For each activity the inspectors examined worker instruction requirements which included protective clothing, engineering controls to minimize dose exposures, the use of predetermined low dose waiting areas, as well as the on-the-job supervision by the work crew leaders to verify that the licensee had maintained the radiological exposure for these work activities ALARA. The inspectors evaluated radiation protection technician (RPT) performance for each of the aforementioned work evolutions, as well as observing and questioning workers at each job location to verify that they had adequate knowledge of radiological work conditions and exposure controls.

b. Findings

No findings of significance were identified.

.3 Radiological Work/ALARA Planning

a. Inspection Scope

The inspectors observed radiation workers and RPTs during high dose rate or high exposure jobs to determine whether workers demonstrated the ALARA philosophy in practice. Specifically, the inspectors observed the work activities to determine whether workers were familiar with the job scope and tools to be used, whether workers were using low dose waiting areas, and whether the workers demonstrated that their training/skill levels was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings of significance were identified.

.4 Radiological Work/ALARA Planning

a. Inspection Scope

The inspectors examined the station's procedures for radiological work/ALARA planning and scheduling, and evaluated the dose projection methodologies and practices implemented for the 2003 refueling outage to verify that sound technical bases for outage dose estimates existed. The inspectors reviewed selected radiologically significant ALARA planning packages for work activities during the 2003 refueling outage work activities. Those ALARA plans addressed refueling activities, steam generator inspection activities, in-service inspections, reactor coolant pump work activities, as well as the installation and removal of shielding and insulation. The inspectors conducted the reviews to verify that adequate person-hour estimates, job history files, lessons learned, and industry experiences were utilized in the ALARA planning process. In addition, the reviews were conducted to verify that ALARA requirements were integrated into work procedure and RWPs, and work activities were scheduled to consider the benefits of dose reduction activities.

b. Findings

No findings of significance were identified.

.5 Verification of Exposure Estimate Goals and Exposure Tracking System

a. Inspection Scope

The inspectors reviewed the methodology and assumptions used by the licensee for its 2003 refueling outage exposure estimates and exposure goals. Actual job exposure data was compared with estimates to verify that the licensee could project and, thus, control radiological exposure. The inspectors also reviewed the licensee's exposure tracking system to verify that the level of exposure tracking detail, exposure report timeliness, and exposure report distribution were sufficient to support control of collective exposures. The inspectors evaluated how the licensee had identified problems with it's exposure estimates for some jobs, the processes being utilized to revise dose estimates, and methods to improve it's dose forecasting procedures to verify that the licensee could adequately track dose.

b. Findings

No findings of significance were identified.

.6 Declared Pregnant Workers

a. Inspection Scope

The inspectors reviewed the station's dose minimization controls used for declared pregnant workers. Specifically, the inspectors reviewed the licensee's adherence to the requirements contained in 10 CFR 20.1208 by examining the licensee's fetal protection program procedure for tracking radiological exposure to the embryo/fetus, and the administrative and ALARA controls that could be used by the licensee to minimize the dose to the embryo/fetus of a declared pregnant worker.

b. Findings

No findings of significance were identified.

.7 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed selected corrective action documents, (i.e., Kewaunee Assessment Program (KAP) documents), and discussed the documents with Radiation Protection Program staff and management to determine if problem characterization was accurate, and to verify that extent of condition reviews were adequately completed or were in the process of being performed.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles)

a. Inspection Scope

The inspectors reviewed the licensee's protected area access control testing and maintenance procedures. The inspectors observed licensee testing of all protected area access control equipment to determine if testing and maintenance practices were performance based. On two occasions, the inspectors observed in-processing search of personnel, packages, and vehicles to determine if search practices were conducted in accordance with regulatory requirements.

The inspectors reviewed security related event reports and safeguard log entries associated with the access control program for the period April 1, 2002 through June 18, 2003.

The inspectors also reviewed the licensee's CAP to determine if security related issues associated with the protected area access control program were appropriately identified, evaluated, and resolved.

b. Findings

No findings of significance were identified.

3PP3 Response to Contingency Events

a. Inspection Scope

The inspectors walked down the licensee's protected area intrusion alarm system to identify potential vulnerabilities. The inspectors, accompanied by licensee security representatives, observed testing of selected protected area intrusion alarm zones identified by the inspectors. Alarm zone detection was evaluated by conducting various testing methods.

The inspectors also reviewed the effectiveness of alarm station personnel to recognize, identify, and evaluate activities in protected area alarm detection zones on the assessment monitors located in alarm stations. The inspectors also reviewed the field of view provided by the assessment aids to ensure compliance with licensee security plan requirements.

The inspectors also reviewed a sample of licensee force-on-force drill records, and interviewed security management personnel to determine if the licensee had appropriately identified and resolved issues associated with the contingency response program.

1. Findings

No findings of significance were identified.

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspectors reviewed Revision 19 to the Kewaunee Nuclear Power Plant Security Manual to verify that the changes did not decrease the effectiveness of the submitted document. The referenced revision was submitted in accordance with

10 CFR 50.54(p)(2) requirements by licensee letter dated May 13, 2003.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors reviewed the licensee's Performance Indicator data collection process and historical data from the April 2002 through March 2003 for the two performance indicators listed below to verify the accuracy of collected and submitted data. Additionally, the inspectors reviewed corrective action records, monthly operating reports, control room logs, and licensee event reports to independently verify the data that the licensee had collected. The inspectors also referenced NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, to verify the basis for each data reporting element.

Reactor Safety Cornerstone

- Unplanned Scrams per 7,000 Critical Hours; and
- Scrams with a Loss of Normal Heat Removal.

b. Findings

No findings of significance were identified.

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors verified the licensee's assessment of its performance indicator for occupational radiation safety. Since no reportable elements were identified by the licensee for the last three quarters of 2002 and the first quarter of 2003, the inspectors selectively reviewed the licensee's data elements to verify that there were no occurrences in the occupational radiation safety cornerstone during those quarters.

b. Findings

No findings of significance were identified.

.3 Safeguards Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from April 2002 to June 18, 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline" were used. The following PI's were reviewed:

- Fitness-for-Duty Personnel Reliability;
- Personnel Screening Program; and
- Protected Area Security Equipment.

A sample of plant reports related to security events, security shift activity logs, and fitness for duty reports were also reviewed.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 50-305/2003-001-00 Technical Specifications Action Requirements Not Followed - Personnel Failed to Follow Procedure

This LER documented the licensee's failure to meet technical specifications when an axial flux differential monitor alarm was inadvertently disabled when operators responded to an associated nuisance alarm. This finding, associated enforcement, and human performance aspect was documented and dispositioned in NRC Inspection Report 50-305/03-02, Sections 1R15 and 4OA4. The inspectors reviewed the LER and no additional findings of significance were identified. The licensee documented the corrective actions in CAP 14321 and 14328. This LER is closed.

.2 (Open/Closed) Failure to Report A Fitness-for-Duty Event

a. Inspection Scope

The inspectors reviewed licensee action regarding the identification of illegal drugs within the licensee's protected area.

b. Findings

Introduction: The inspectors identified that the licensee failed to report a Fitness-for-Duty (FFD) event as required by NRC 10 CFR 26.73 (a) requirements. The issue when evaluated by the NRC was considered to be of very low safety significance and was dispositioned as a Severity Level IV Non-Cited Violation (NCV).

Description: On May 28, 2003, a plant employee, while walking in the facility's protected area found, a small plastic bag on the ground which contained a leafy material. There were no personnel observed in the area were the bag was found. The plant staff member took possession of the plastic bag and immediately contacted security. At approximately 7:30 a.m. onsite field testing of the leafy material proved positive for a controlled substance. The plastic bag and substance was turned over to the local police department for investigation and disposal.

After finding the item and confirming its contents, the need to report the issue to NRC was reviewed. Personnel involved in the process included representatives from the licensee's plant security and the site licensing staff, along with a senior corporate management FFD representative. The licensee concluded that the FFD event did not represent a significant issue/event and that logging the event was the appropriate decision. The decision was based on licensee analysis that the discovery of the substance (contraband) in and of itself was not a significant threat to the site, and that NRC FFD reporting criteria required actual possession by a person to represent a significant FFD event.

Later that same day, as a courtesy, the licensee's security manager contacted a Region III physical security inspector to advise the Region of the event, and the licensee's decision to log the event. The inspector questioned the licensee's conclusion based on information contained in Section 10.7 of NRC NUREG-1385, "Fitness-for-Duty in the Nuclear Power Industry: Responses to Implementation Questions," which identified and documented a question and answer that supported the reporting of such an event to the NRC. The question and answer as stated in NUREG-1385:

Question:

"Should finding alcohol or drugs within the protected area (no person in possession) be reported?"

NRC's response:

"Yes. Possession would be inferred and would be required to be reported as a significant FFD event under the meaning of 10 CR26.73(a)."

After further evaluating their discussion with the inspector and identifying that they were not familiar with NUREG-1385, the licensee reported the event to the NRC Operation Center at approximately 11:40 a.m. on May 29, 2003.

Analysis: This issue was evaluated and dispositioned using the NRC Significant Determination Process (SDP), for the Physical Protection cornerstone. The issue was screened through the Physical Protection SDP. The inspectors determined that the finding was more than minor issue because it involved a failure to meet NRC regulatory requirements noted in 10 CFR 26.73(a) that involved the reporting of significant FFD events to the NRC. The finding further represented a vulnerability in a licensee's plan, was not a malevolent act, and greater than two similar findings had not occurred in four calendar quarters. Therefore, the inspectors concluded the issue was of very low safety significance. In this case the licensee demonstrated less than prudent judgement by not recognizing that this event should have been reported as a significant FFD.

Enforcement: Title 10 CFR 26.73(a) requires, in part, that the licensee inform the Commission of significant Fitness-for-Duty events including possession of illegal drugs within the protected area by notification to the NRC Operations Center by telephone within 24 hours of the discovery of the event by the licensee. Contrary to this, on May 28, 2003, the licensee failed to report a significant Fitness-for-Duty event that involved possession of illegal drugs within the licensee's protected area to the NRC Operations Center within 24 hours of discovery. The results of the violation were determined to be of very low safety significance; therefore, this violation of 10 CFR 26.73 was classified as a Severity Level IV violation. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-305-03-04-04). This violation is documented in the licensee's corrective action program as CAP 033527.

4OA4 Cross-Cutting Findings

The finding described in Section 1R20.3 of this report, Improper Tagout Sequence Results in Loss of RCS Inventory, had as a contributing cause, a human performance deficiency, in that, a work control supervisor authorized the processing of a tagout when the supervisor was uncertain that the plant conditions were appropriate. The processing of the tagout resulted in an inadvertent loss of RCS inventory while the unit was in refueling shutdown mode.

4OA5 Other Activities

.1 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (TI 2515/150)

a. Inspection Scope

The objective of TI 2515/150, Revision 1, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles," was to implement an on-site NRC review of the licensees' activities in response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," to verify compliance with applicable regulatory requirements. In response to NRC Bulletin 2002-02, Kewaunee Nuclear Plant calculated the effective degradation years based on time and head temperature which placed the plant in the "Moderate Susceptibility" ranking for leakage of the penetration nozzles. As a result, the licensee performed a 100 percent bare metal visual inspection of the reactor pressure vessel (RPV) head and penetration nozzles. The inspectors interviewed inspection personnel, reviewed procedures and inspection reports, including photographic documentation, to assess the licensee's efforts in conducting the visual examination of the RPV head.

Summary

The licensee did not identify any leaking vessel head penetration nozzles (VHP).

b. Evaluation of Inspection Requirements

In accordance with requirements of TI 2515/150, the inspectors evaluated and answered the following questions:

a. Was the examination:

1. Performed by qualified and knowledgeable personnel?

   Yes. The inspectors verified that the examination was performed by five examiners certified in both VT-1 and VT-3 Methods to at least Level II (one examiner was a Level III), with additional training specific to Electric Power Research Institute (EPRI) Procedure 1006296 Revision 1, "Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of RPV Head."

2. Performed in accordance with demonstrated procedures?

   No volumetric examinations were conducted during this outage. The inspectors verified that the bare metal visual examinations were conducted in accordance with NEP No. 15.5, Revision 0, "Visual Examination for Inservice Inspection," and that EPRI Procedure 1006296 Revision 1, "Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of RPV Head" was used in the pre-job briefing notes and documented as being used as guidance for the examination.

   Kewaunee has block contoured vessel head insulation, consisting of mirror panels fabricated of 4 inch thick perforated metal block insulation. The inspectors verified that the insulation and shroud were completely removed and the as-found head was clean, with no viewing obstructions to the exam. There were 41 penetrations, including the 3/4" head vent, which were fully examined. Loose surface debris on the uphill side of penetrations and staining of control rod drive mechanism lengths were observed.

3. Able to identify, disposition, and resolve deficiencies?

   Yes. The inspectors concluded from the review of the photographic documentation that the licensee had sufficient access to perform a direct visual examination with 360 degree coverage of each penetration. The procedural resolution requirements (able to discern a 1/32 inch black line on an 18 percent neutral gray card) for the direct visual examination of the vessel head was adequate to detect boric acid deposits.

4. Capable of identifying the primary water stress corrosion cracking phenomenon described in the bulletin?

   Yes. The inspectors determined through interviews with inspection personnel, reviews of the Visual Examination Record, and photographic documentation that the licensee's efforts were capable of detecting and characterizing VHP nozzle leakage.

b. What was the condition of the reactor head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

Kewaunee has block contoured vessel head insulation, consisting of mirror panels fabricated of 4 inch thick perforated metal block insulation. The inspectors verified that the insulation and shroud were completely removed and the as-found head was clean, with no viewing obstructions to the exam. There were 41 penetrations, including the 3/4 inch head vent, which were fully examined. Loose surface debris on the uphill side of penetrations and staining of control rod drive mechanism lengths were observed.

c. Could small boron deposits, as described in Bulletin 2001-01, be identified and characterized?

Yes. The inspectors determined through interviews with inspection personnel, reviews of the inspection procedure and examination reports, that small boron deposits, as described in the Bulletin 2001-01, could be identified and characterized.

d. What material deficiencies (associated with the concerns identified in the bulletin) were identified that required repair?

There were no material deficiencies associated with the 41 VHP nozzles that were considered indicative of leakage.

e. What, if any, significant items that could impede effective examinations?

None. The licensee had sufficient access to perform a direct visual examination with 360 degree coverage of each penetration

f. What was the basis for the temperatures used in the susceptibility ranking calculation?

In Bulletin 2002-02, the Effective Degradation Years (EDY) is used as a basis to establish appropriate inspection programs for VHP nozzles based on increasing susceptibility to nozzle cracking with increasing EDY. Calculation No. C11470, “Reactor Vessel Head Effective Degradation Year (EDY),” uses a reactor pressure vessel head temperature of 583 degrees F., which is the published value in MRP-48, “PWR Materials Reliability Program Response to NRC Bulletin 2001-01,” dated August 2001.

c. Findings

No findings of significance were identified.

4OA6 Meetings

On July 2, 2003, the resident inspectors presented the inspection results to Mr. Coutu and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Interim Exit Meetings

Interim Exits were conducted for:

- Radiation Protection inspection with Mr. K. Hoops on April 11, 2003; and
- Inservice Inspection (IP 71111.08) and Temporary Instruction 2515/150 with Mr. K. Hoops on April 17, 2003.
- Safeguards Inspection with Mr. K. Hoops on June 20, 2003.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Nuclear Management Company, LLC

T. Coutu, Site Vice President, Kewaunee Site
K. Hoops, Plant Manager
G. Arent, Regulatory Affairs
L. Armstrong, Engineering Director
S. Baker, Manager, Radiation Protection
P. Bukes, ISI Coordinator
M. Fencl, Security Manager, Kewaunee/Point Beach
G. Harrington, Licensing Leader
B. Kopetsky, Security Coordinator
J. McCarthy, Assistant Plant Manager, Operations
T. Olson, Inspection Services Engineer
J. Palmer, Plant Mechanical Supervisor
B. Piesl, NMC Security Consultant
S. Putman, Assistant Plant Manager, Maintenance
R. Repshas, Manager, Site Services
J. Rista, Licensing Supervisor
J. Stafford, Superintendent, Operations

NRC Personnel

J. Lamb, Project Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

50-305, 03-04-01	NCV	Failure to Ensure Material of Installed Pipe Plug in RHR System is in Accordance with Design Requirements (Section 1R20.2)
50-305, 03-04-02	NCV	Failure to Ensure Plant Conditions Appropriate for Tagout Results in Loss of Reactor Coolant System Inventory (Section 1R20.3)
50-305, 03-04-03	NCV	Failure to Provide Appropriate Instructions in Refueling Procedure Results in Reactor Vessel Level Indication Perturbation (Section 1R20.4)
50-305/03-04-04	NCV	Failure to report a significant Fitness-for-Duty event in a timely manner (Section 4OA3).

Closed

50-305/2003-001-00	LER	Technical Specifications Action Requirements Not Followed - Personnel Failed to Follow Procedure (Section 4OA3.1)

Discussed

None

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

N-CC-31-CL; Component Cooling System Prestartup Checklist; Revision Y

N-DGM-10-CLB; Diesel Generator B Prestartup Checklist; Revision H

TAC NO. MA6544; Site-Specific Worksheets For Use In the Nuclear Regulatory Commission's Significance Determination Process; December 7, 2000

OPERM-603; Flow Diagram - Air Conditioning Administration Building and Control Room; Revision AW

E-2003; Integrated Logic Diagram Control Room Air Conditioning System; Revision 0

E-2004; Integrated Logic Diagram Control Room Air Conditioning System; Revision M

CAP017122; Outdoor Air Intake for Control Room HVAC and Auxiliary Building HVAX Needed

CAP017123; Duct Tape on Ventilation Openings Missing

1R05 Fire Protection

KNPP Fire Protection Program Plan; Revision 4

PFP-11; Fire Plan Drawing - Turbine Building Basement; Revision C

PFP-12; Fire Plan Drawing - Turbine Building Mezzanine; Revision C

PFP-4; Fire Plan Drawing - Screenhouse; Revision B

PFP-31; Fire Plan Drawing - Reactor Containment & Shield Building Areas (592' Elev); Revision A

PFP-32; Fire Plan Drawing - Reactor Containment & Shield Building Areas (606' Elev); Revision B

PFP-33; Fire Plan Drawing - Reactor Containment & Shield Building Areas (626' Elev); Revision B

PFP-34; Fire Plan Drawing - Reactor Containment & Shield Building Areas (649' Elev); Revision A

PMP-08-33; Penetration Fire Barrier Inspection; July 10, 2001; Revision D

1R06 Flood Protection Measures

Sargent and Lundy Report SL-7234, Volume 1; Internal Flood Levels Due To Postulated Moderate Energy Piping Failures, Kewaunee Nuclear Power Plant - Unit 1; October 30, 1989

Sargent and Lundy Report SL-7234, Volume 2; Internal Flood Levels Due To Postulated Moderate Energy Piping Failures, Kewaunee Nuclear Power Plant - Unit 1; November 29, 1989

WPS 77688; Phase II, Safe Shutdown Assessment of Internal Flood Levels Due to Postulated Moderate Energy Piping Failures; May 1, 1990

Kewaunee Request For Information RI-39-008; Internal Flooding Study Discrepancies; January 28, 1991

Drawing E-350; Plan - Plant Site Underground Conduit & Cable Routes; Revision AN

Drawing 237127B-E21; Trenwa General Arrangement & Communication Underground Plan; Revision L

Drawing 237127B-E20F; Trenwa - Assembly & Details; June 12, 1970

Drawing 237127A-E347F; Plans & Sections - Plant Site Underground Conduit; November 2, 1970

1R07 Heat Sink Performance

PMP-18-13; Containment Fan Coil Unit Performance Monitoring (QA-1) performed on October 9, 10, 2002 and May 12, 2003; May 30, 2001, Revision A

PM18-002; Fan Coil Unit - Containment 1A Performance Monitoring

PM18-003; Fan Coil Unit - Containment 1C Performance Monitoring

PM18-004; Fan Coil Unit - Containment 1B Performance Monitoring

PM18-005; Fan Coil Unit - Containment 1D Performance Monitoring

1R08 Inservice Inspection

NEP 15.16; Ultrasonic Examination of Austenitic Piping and Austenitic Vessels for Inservice Inspection; August 24, 1999

UT 83; Operation of the LMT "ADAM System;" January 20, 2000

NEP 15.6; Liquid Penetrant Examination for Inservice Inspection; August 29, 1995

CAP014098; OEA 2002-308 Safety Injection Tank Leak

CAP013908; ASME Section XI Appendix VIII Examination of Dissimilar Metal Welds

1R11 Licensed Operator Requalification

EPIP-AD-02; Emergency Class Determination; Revision AG

E-3; Steam Generator Tube Rupture; Revision T

E-0; Reactor Trip or Safety Injection; Revision U

1R12 Maintenance Effectiveness

MRE 1968; Missed Maintenance Rule Evaluation; April 5, 2003

WO 03-3973; Flow Orifice Downstream of RHR-500B Has a Leak From an Installed Cap; April 5, 2003

Maintenance Rule System Basis - System 34 Residual Heat Removal; Revision 2

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 4, 2003 1800-0600; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 5, 2003 0600-1800; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 5, 2003 1800-0600; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 6, 2003 0600-1800; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 6, 2003 1800-0600; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 7, 2003 0600-1800; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 7, 2003 1800-0600; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 8, 2003 0600-1800; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 8, 2003 1800-0600; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 9, 2003 0600-1800; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 9, 2003 1800-0600; dated April 3, 2003, Revision H

Form GNP-08.04.01-1; Shutdown Safety Assessment Checklist for April 10, 2003 0600-1800; dated April 3, 2003, Revision H

Figure 1 GNP-08.04.01; Shutdown Inventory Flow Paths for April 4 - 10, 2003; dated April 3, 2003, Revision H

Figure 2 GNP-08.04.01; Shutdown Reactivity Flow Paths for April 4 - 10, 2003; dated April 3, 2003, Revision H

Figure 3 GNP-08.04.01; Substation for April 4 - 10, 2003; dated April 3, 2003, Revision H

Chemistry Desktop Chart Data; RCS Hotleg, Letdown, and Pressurizer Boron Samples from April 5, 2003 through April 10, 2003

1R14 Non-Routine Evolutions

N-RC-36D; Filling and Venting the Reactor Coolant System; Revision AE

N-TB-54; Turbine and Generator Operation; Revision AX

GNP 3.17.06; Control Room Conduct; Revision A

GNP 3.17.7; Watchstanding Practices; Revision A

CAP 16848; Turbine Control Problem Resulting in Plant Load Decrease; June 10, 2003

CAP 16858; Operation of DEH System at KNPP not Aligned with Industry Practices; June 10, 2003

CAP 16853; Condenser Dump Valve Trip Open Status Light Coming In and Out; June 10, 2003

1R15 Operability Evaluations

CAP 15459; Unexpected Response During SP-42-312A “DG A Availability Test”; April 1, 2003

Drawing E-1621; Integrated Logic Diagram Diesel Generator Mechanical System; Revision AL

Drawing E-1586; Schematic diagram D/G A Shutdown, Governor Control & Auxiliary Relays; Revision AM

Design Description DC 2933; Diesel Generator High-Speed Stop (GHS) Switch Setting Adjustment; Revision 1

OPR 30; SI-5B-1 Operability; April 22, 2003

CAP 16003; B Diesel Generator JW HX Tube Bundles Tie Rod Hex Nut not Staked; April 22, 2003

CAP 16788; Effect of Quarterly Calibration on Red Channel Nuclear Instrumentation System and Monthly on Remaining Channels; June 4, 2003

1R16 Operator Workarounds

OWA 01-15; Tagout 98-460. SAP refrigeration unit is ineffective.

OWA 01-16; Tagout 99-253. NRC Generic Letter 96-06 Valves Must Remain Open When at or Above Hot Shutdown.

OWA 02-01; Component Cooling Pump Overheating Concern in Two Pump Operation with Normal At-Power Flows.

OWA 03-02; SW-2910-1; Air Side Seal Oil Temperature Control Valve does not control temperature within normal band.

OWA 03-03; Service Water Pump A1 Strainer is continuously in backwash.

1R17 Permanent Plant Modifications

Report M-11166-004-CC.2; Component Cooling Water Heat Exchanger Replacement - Heavy Load Analysis; Revision 0

Design Description and Scope - DCR 3449; Component Cooling Water Heat Exchanger Replacement; Revision 1

TCR 02-20; Temporary Electrical Raceway Changes to Support CCW HX Replacement - DCR 3449

TCR 02-19; HVAC Modifications to Support CCW HX Replacement

TCR 02-22; Design Description - Temporary Removal of SW Train A Supply and Return Piping to Control Room HVAC Equipment to Support CCW HX Replacement DCR 3449

DCR 3398, Phase 1, Revision 2, Field Change No. 9 and associated 10 CFR 50.59 analyses - Reduce the margin to trip between OPDT Turbine Runback Rod Stop Withdrawal and OPDT Trip by 2 percent.

1R19 Post-Maintenance Testing

General Maintenance Procedure (GMP) 239; Limitorque MOV Breaker/Starter, Motor, and Actuator Maintenance; Revision E

GMP 239-A1; Limitorque MOV Maintenance Table of Information; Revision B

PMP 33-06; Safety Injection (SI) Containment Sump QA-1 Motor Operated Valve Maintenance; Revision N

NEP-14.18; Local Leak Rate Testing of Non-Reportable Valves; Revision C

SP-34-099B; Train B RHR Pump and Valve Test - IST, Data Sheet 1; December 5, 2002

SP-10-111-3; Inspection of Diesel Generator A (Component Retest); Revision F

OPR 38; AFW Pump Operability Concern Due to Working on Discharge Check Valves; April 30, 2003

WO 02-7551; Replace Springs in 30 in. Valve-Check-Main Steam Isolation Valve Assembly - Gen 1A

WO 03-6467; Adjust Signal Converter (T-avg) Impulse Lag Unit

SP-47-316C; Channel 3 (Blue) Instrument Channel Test; May 29, 2003; Revision O

1R20 Refueling and Outage Activities

N-O-05; Plant Cooldown From Hot Shutdown to Cold Shutdown Condition; dated March 25,2003, Revision AR

N-RC-36E; Draining the Reactor Coolant System; dated April 03, 2003, Revision AC

Chem-40.007; Hydrogen Peroxide Addition to the Reactor Coolant; March 25, 2003, Revision A

N-O-03; Plant Operation Greater Than 35 percent Power; dated February, 27, 2003, Revision AO

N-O-04; 35 percent Power to Hot Shutdown Condition; dated February 13, 2003, Revision X

CAP015561; RCS Cooldown Performed Out of Sequence in N-O-05; dated April 7, 2003

CAP015562; Determine Acceptability of SP34-203; dated April 7, 2003

ACE002221; RHR System Leak Near RHR-500B; dated April 6, 2003

CAP015530; RHR System Leak Near RHR-500B; dated April 5, 2003

WO 03-3974; Replace Class 2 RHR System ½" Plugs of 2" Flange Downstream of RHR-500A

WO 03-3973; Replace Class 2 RHR System ½" Plugs of 2" Flange Downstream of RHR-500B

Tagout 03-000251; Steam Generator 1A & 1B Secondary Side Isolation, dated April 8, 2003

OPERM-436; Flow Diagram Steam Generator Blowdown System Modification; Revision AG

OPERM-205; Flow Diagram Feedwater System; Revision AW

OPERM-214; Flow Diagram Chemical Injection System; Revision AD

Tagout 03-000432; Component Cooling Heat Exchanger ‘1A’ Removal/Replacement; dated April 11, 2003

OPERXK-100-19; Flow Diagram Auxiliary Coolant System; Revision AE

RF-01.00; KNPP Refueling Procedure; Revision J

ES-2001; Standard Specification for Pipe and Fittings; Revision 3

Drawing XK-100-18; Flow Diagram - Auxiliary Coolant; Revision AK

F Specification F-5; Instructions, Precautions, and Limitations for Handling New and Partially Spent Fuel Assemblies; Revision 17

Calculation C11479; SFP Cooling - One Pump Operation 2003 Outage; Revision 0

RF-03.01; Fuel Movement During a Refueling Outage; Revision G

NRC-89-1; Response to Generic Letter 88-17; January 3, 1989

CAP 15768; Poor Engineering Practice Used to Develop Core Unloading Pattern; April 15, 2003

OPR 42; Reactor Vessel Head Examination; May 2, 2003

Tagout 03-266; Lower Pump Shaft on RXCP ‘A’ & Remove/Replace Pump

RCE 611; Unexpected Loss of Level in RCS and Reactor Head Vent Unavailability Due to Blank Flange Installed on Head Vent Line; May 14, 2003

CAP 15628; Reactor Vessel Level Decrease During TO 03-266; April 9, 2003

CAP 16044; Damaged Rod Control Cluster Assembly (RCCA); April 24, 2003

CAP 16236; Possible Duct Tape on Incore Instrumentation Nozzles; May 2, 2003

RXT-22.00; BOL Physics Test; Revision A

RXT-02.00; Low Power Physics Test; Revision W

RXT-01.00; Initial Criticality by Dilution; Revision U

Tagout 03-769; Maintain SW-1800 Closed to Minimize SW Flow and to Minimize Condensation in the Tube Bundle; April 19, 2003

Calculation C11497; Auxiliary Feedwater Lube Oil Cooler Minimum flow Determination, Revision 0

1R22 Surveillance Testing

SP-34-145F;Residual Heat Removal Valves RHR-1A, 1B, 2A &2B Reactor Coolant System Interlock and Alarm Test; dated October 1, 2001, Revision B

SP-33-191; Safety Injection Flow Test; Revision T

KAP 0622; Pump Operability Determination Summary

KNPP Inservice Testing Program Third Ten-Year Interval; Revision P

SP-42-312B; Diesel Generator B Availability Test; dated November 19, 2002, Revision R

20S1 Access Controls For Radiologically Significant Areas

KAP 015612; High Radiation Area Boundary Found Hanging From Stanchion; April 8, 2003

KAP 015654; Potential Weakness in HP Procedures that Provide Guidance on Entries into a LHRA; April 10, 2003

RWP 14; Remove and Re-install Insulation/Lagging in the Aux and Containment; Revision 0

RWP 42; Transfer Fuel to New Racks in Canal & Inspect Assembly Tops; Revision 0

RWP 53; Rx Head Disassembly and Re-Assembly; Revision 0

RWP 57; Remove and Replace Reactor Vessel Head O-Rings; Revision 0

RWP 80; PM's on 1A and 1B RCP Motors; Revision 0

RWP 82; Swap-out RCP Motor and all Supporting Activities; Revision 0

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

HP-04.001; ALARA Plan; Revision D

Kewaunee Nuclear Power Plant Strategic Plan for Radiation Dose Reduction 2002-2007

Actual vs. Estimated Exposures per RWP Sorted by Task, RWPs 30001 to 31000 From April 5, 2003, to April 7, 2003; April 8, 2003

HPF-146; Pregnancy Declaration Form, Revision A

03-001; Refueling ALARA Plan; March 25, 2003

03-005; "Alpha" Reactor Coolant Pump Replacement; February 18, 2002

03-014; In Service Inspection ALARA Plan; January 29, 2003

03-015; KNPP Health Physics and Chemistry; March 13, 2003

KAP 015506; Higher Dose Rates due to Steam Generator Replacement: February 20, 2003

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles)

Security Implementing Procedure 30.2; Testing, Inspection, and Maintenance of Security Equipment; dated March 14, 2003.

Security Loggable Events; April 2002 to May 2003

Corrective Action Program Reports - Security Related; May 2002 to May 2003

3PP3 Response to Contingency Events

NMC Security Exercise and Drill Procedure; Revision 6

Security Drill and Exercise Reports; June 2002 to May 2003

Security Implementing Procedure 30.02; Testing, Inspection, and Maintenance of Security Equipment, March 14, 2003.

3PP4 Physical Protection - Security Plan Change

Kewaunee Nuclear Power Plant Security Manual; Revision 19, dated May 13, 2003.

4OA1 Performance Indicator Verification

General Nuclear Procedure (GNP-03.18-01; NRC Performance Indicators Reporting Requirements; Revision 15.

Security Department Policy (SDP-45); NRC Security Performance Indicators; August 12, 2002.

Kewaunee Security System Tracking; June 2003 to May 2003.

4OA5 Other Activities

NEP 15.5; Visual Examination for Inservice Inspection; September 19, 1995

CAP015746; NRC Review of Calculation C11470 identified some discrepancy

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Process
CFR	Code of Federal Regulations
DG	Diesel Generator
DRP	Division of Reactor Projects, Region III
DRS	Division of Reactor Safety, Region III
EDY	Effective Degradation Years
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
FFD	Fitness For Duty
HRA	High Radiation Area
ICM	Interim Compensatory Measures
ISI	In-Service Inspection
KAP	Kewaunee Assessment Process
LER	Licensee Event Report
LHRA	Locked High Radiation Areas
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PT	Penetrant Testing
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPT	Radiation Protection Technician
RPV	Reactor Pressure Vessel
RVLIS	Reactor Vessel Level Instrumentation System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SSC	Systems, Structures, and Components
TI	Temporary Instruction
TSC	Technical Support Center
UT	Ultrasonic Testing
VHP	Vessel Head Penetration
VHRA	Very High Radiation Areas