January 26, 2004

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 05000305/2003008

Dear Mr. Coutu:

On December 31, 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 January 5, 2004, with Mr. Davison 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 were two NRC-identified and one self-revealed finding of very low safety significance (Green). These findings involve violations of NRC requirements. In addition, one issue was reviewed under the NRC traditional enforcement process and determined to be a Severity Level IV Violation of NRC requirements. However, because these violations were of very low safety significance, non-willful and non-repetitive, and because the violations were entered in your corrective action program, the NRC is treating these issues 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 Office at the Kewaunee facility.

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 L. Louden, Chief
Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000305/2003008
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.:	05000305/2003008
Licensee:	Nuclear Management Company, LLC
Facility:	Kewaunee Nuclear Power Plant
Location:	N 490 Highway 42 Kewaunee, WI 54216
Dates:	October 1 through December 31, 2003
Inspectors:	R. Krsek, Senior Resident Inspector R. Berg, Resident Inspector A. Barker, Reactor Engineer, NRR J. Cameron, Project Engineer M. Maley, Reactor Engineer, NRR D. Nelson, Radiation Specialist B. Jorgensen, NRC Contractor
Approved By:	Patrick L. Louden, Chief Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000305/2003008; 10/01/2003 - 12/31/2003; Kewaunee Nuclear Power Plant; Permanent Plant Modifications, Temporary Plant Modifications, Problem Identification and Resolution and Event Notification.

This report covers a 3-month period of baseline resident and announced radiation protection inspections. The inspections were conducted by the resident inspectors, and Region III and NRC Headquarters inspectors. Three NRC-identified Green findings and one self-revealed Green finding associated with Non-Cited Violations were identified. The significance of most findings is indicated by the color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). 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. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- NCV. The inspectors identified a finding of very low safety significance associated with a Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for changes made to the facility. Specifically, the licensee 'screened out' of the 10 CFR 50.59 process a modification that included the addition of a minimum flow recirculation line to the component cooling water pumps. This modification further cross-connected the suction and discharge piping of both component cooling water pump trains. Subsequently, the inspectors identified and the licensee concurred that a safety evaluation was required for this modification.

  Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impeded the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of this violation were assessed using the Significance Determination Process. In this case, the licensee failed to perform a safety evaluation in accordance with 10 CFR 50.59 and had placed the new system in service for testing prior to the completion of the required safety evaluation.

  The inspectors considered this issue to be of more than minor significance because, if left uncorrected, the issue could become a more significant safety concern. Specifically, the inspectors noted that the licensee's processes for permanent modifications failed to identify this issue at several review levels. The inspectors determined that the issue was of very low significance because the new system was placed in service for a short period of time for testing prior to the completion of the required safety evaluation. In addition, the final safety evaluation completed by the licensee in October 2003 determined that the modification did not require prior NRC approval. The inspectors determined this finding was a Severity Level IV Non-Cited Violation of 10 CFR 50.59. The inspectors also determined that the finding had, as a primary cause, a human performance deficiency which affected the cross-cutting are of Human Performance. (Section 1R17.1)

- Green. The inspectors identified a finding of very low safety significance associated with a Non-Cited Violation of 10 CFR Part 50 Appendix B, Criterion III, “Design Control,” for the licensee’s failure to provide for checking of the adequacy of the design in Temporary Change TCR 03-036, in that, the design review failed to confirm the structural integrity of the new pressure boundary established for the studding outlet. Consequently, the licensee performed non-destructive examinations and additional flaw and engineering analyses to confirm the adequacy of the new design.

  The inspectors considered this issue of more than minor significance, because if left uncorrected, the issue could become a more significant safety concern. In addition, the inspectors concluded that the finding was greater than minor because the finding involved the design control attribute of the mitigating systems cornerstone and affected the mitigating systems objective of ensuring the capability of the component cooling water system in response to initiating events to prevent undesirable consequences. Specifically, the temporary design change relied on unsupported assumptions that could have impacted the structural integrity of the component cooling water suction line. The inspectors evaluated the finding using the Significance Determination Process Phase 1 screening and determined that the finding was a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18; therefore, the finding was determined to be of very low safety significance. (Section 1R23.1)

- Green. The inspectors identified a finding of very low safety significance associated with a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” for the licensee’s failure to take adequate corrective actions in response to the installation of non-conforming cranking cutout relays which prevented energizing of the diesel generator engine start relay. The licensee’s corrective actions for this condition adverse to quality addressed routine surveillance procedures, but did not consider the licensee’s Emergency Operating Procedures to ensure the Emergency Diesel Generators would remain operable following Diesel Generator Shutdowns as directed by those procedures.

  The inspectors considered this issue of more than minor significance, because if left uncorrected, the issue could become a more significant safety concern. In addition, the inspectors concluded that the finding was greater than minor because the finding involved the design control attribute of the mitigating systems cornerstone and affected the mitigating systems objective of ensuring the capability of the diesel generators in response to initiating events to prevent undesirable consequences. Specifically, in part, the licensee’s corrective actions included revisions to normal operating procedures to verify continuity across the relay contacts following shutdown of the emergency diesel generators; however, the licensee did not similarly revise its Emergency Operating Procedures to verify continuity across the cranking cutout relay contacts following shutdown of the emergency diesel generators. The inspectors evaluated the finding using the Significance Determination Process Phase 1 screening and determined that the finding was a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18; therefore, the finding was determined to be of very low safety significance. (Section 4OA2.1)

- Green. A finding of very low safety significance involving a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was self-revealing when the “B” emergency diesel generator failed to start on February 26, 2003, during a daily Technical Specification-required test, in response to the “A” emergency diesel generator being out of service for regularly scheduled 18-month periodic maintenance. The generator failed to start due to a pair of electrically open contacts on a cranking cutout relay which prevented energizing of the engine start relay. The cranking cutout relay had been installed during a design change completed in 1998, and the performance ratings of the new relay did not match original design specifications.

  The inspectors considered this issue of more than minor significance, because if left uncorrected, the issue could become a more significant safety concern. In addition, the inspectors concluded that the finding was greater than minor because the finding involved the design control attribute of the mitigating systems cornerstone and affected the mitigating systems objective of ensuring the capability of the diesel generators in response to initiating events to prevent undesirable consequences. Specifically, the temporary design change failed to consider inductive electrical loads across the relay contacts, for which the relays were not rated. The inspectors evaluated the finding using the Significance Determination Process Phase 1 screening and determined that the finding was a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18; therefore, the finding was determined to be of very low safety significance. (Section 4OA3.1)

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

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

REPORT DETAILS

Summary of Plant Status

The plant operated at or near full power during most of the inspection period, except for one brief power reduction to 70 percent to support routine, planned quarterly turbine valve and auxiliary feedwater pump testing. In addition, operators reduced power to approximately 70 percent on December 6, to support emergent maintenance on a switchyard breaker. Operators reduced power, in accordance with Technical Specifications (TSs), on December 12, when the licensee declared the component cooling water system inoperable due to a leak on a common section of piping for both trains of that system. The licensee subsequently terminated the power reduction at 20 percent power, after declaring the component cooling water system operable. Operators returned the plant to 100 percent, where it remained for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's program and procedures for adverse weather and seasonal weather extremes, completing one inspection procedure sample. The review included an assessment of protection capability to verify that selected systems and components remained functional when challenged by adverse weather. The inspectors selectively verified seasonal cold weather protection features for plant systems, structures and components. This included system and area walkdowns to assess the physical condition of weather protection features. The inspectors focused attention on systems/components required for accident mitigation and safe reactor shutdown. The inspectors also re-examined the history of issues raised in the area of seasonal severe weather and assessed the licensee's corrective actions.

The inspectors also evaluated the licensee's readiness for the rapid onset of severe conditions involving extreme high winds, lightning, and/or flooding. The evaluation included verification of operator actions prescribed in licensee procedures to assess the efficacy of those actions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors partially walked down the Emergency Diesel Generator 1A train while the opposite train of equipment was out-of-service, completing one inspection procedure sample. The inspectors verified that the system was correctly aligned to perform the design safety function. In preparation for the walkdown, the inspectors reviewed the system lineup checklists, normal operating procedures, abnormal and emergency operating procedures, and system drawings to verify the correct system lineup. During the walkdowns, the inspectors examined support system valve positions and electrical power availability to verify that valves and electrical breaker positions were consistent with the licensee's procedures and design documentation. The inspectors also inspected the material condition of the equipment.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems: Auxiliary Feedwater (71111.04S)

a. Inspection Scope

The inspectors completely walked down the auxiliary feedwater system, completing one inspection procedure sample. At the time of the inspection, the auxiliary feedwater system was aligned for emergency standby readiness. The inspection included a review of licensee procedures for normal, abnormal and emergency system operations. Other documents reviewed included design drawings, piping & instrument drawings, the degraded equipment log, and system lineup checklists.

The inspectors reviewed "open" and recently "closed" maintenance work requests for the auxiliary feedwater system, to assess whether the identified work had the potential to adversely affect system operability. In addition, the inspectors reviewed in-process engineering design change requests associated with the auxiliary feedwater system and discussed the current status with licensee personnel.

Finally, the inspectors' walkdown of the auxiliary feedwater system included all accessible system piping and valving associated with all three auxiliary feedwater pumps, electrical power supplies, local and dedicated control panel switches and controls, and monitoring and alarm systems. The inspectors verified that support systems and devices were functional and properly aligned to perform the respective safety functions. During the walkdown, the inspectors reviewed correct valve and switch positions; appropriate equipment labeling; availability of electrical power; availability of support systems; and verification of outstanding corrective work orders to ensure system or component functions were not adversely impacted.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Zone Walkdowns

a. Inspection Scope

The inspectors walked down the following five fire protection zones, completing five inspection procedure samples:

- Fire Zone TU-92, Emergency Diesel Generator 1B Room;
- Fire Zone TU-97, Battery Room 1A;
- Fire Zone TU-98, Battery Room 1B;
- Fire Zone TU-95A, Dedicated Shutdown Panel Area; and
- Fire Zone AX-23B, Residual Heat Removal and Component Cooling Water Heat Exchanger Area, Letdown and Seal Water Filter Area, and Refueling Water Storage Tank and Valve Gallery.

During the walkdowns, the inspectors focused on the availability, accessibility, and condition of fire fighting equipment; the control of transient combustibles and ignition sources; and the operating status of installed fire barriers. The inspectors selected fire areas for inspection based on the overall contribution to internal fire risk, and the potential to impact equipment which could initiate a plant transient. The inspectors verified that fire response equipment was in the designated location and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and that fire doors, dampers, and penetration seals were in satisfactory condition. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation (71111.05A)

a. Inspection Scope

The inspectors observed an unannounced fire drill on October 22, 2003, completing one inspection procedure sample. The inspectors evaluated the fire brigade's readiness to fight fires which included the following attributes:

- Required number of fire brigade members reported to the scene promptly;
- Fire brigade members used protective clothing and self-contained breathing

apparatus properly;

- Amount and type of firefighting equipment brought to the fire scene was appropriate, fire hoses were laid out without flow restrictions and enough fire hose was available to reach the fire hazard;

- Fire brigade leader provided clear, thorough and effective directions to both the on-site fire brigade members and to personnel from the off-site departments;
- Communication between fire brigade members and the control room operators was effective; and
- Fire brigade leader used firefighting pre-plan strategies effectively.

In addition, the inspectors verified that the licensee followed the pre-planned drill scenario and met the drill objectives. The inspectors observed the post-drill critique to assess the licensee evaluators' and fire brigade members' ability to self-identify problems. The inspectors also reviewed condition reports to verify that problems identified during the post-drill critique were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On October 29, 2003, the inspectors observed classroom and simulator training on the recent changes incorporated into the emergency operating procedures, completing one inspection procedure sample. The procedure changes addressed operator response to the cavitation of residual heat removal pumps due to containment sump strainer blockage. The licensee made the changes and conducted the training as part of its commitments in response to NRC Bulletin BL-03-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors."

The inspectors observed portions of the classroom training and crew performance in the simulator for clarity and formality of communications, ability to take timely actions in a safe direction, procedure use, control board manipulations, oversight and direction from supervisors, group dynamics and annunciator response. Additionally, the inspectors evaluated the crew's implementation of the facility's abnormal and emergency operating procedures, oversight and direction provided to the crew by the shift manager and control room supervisor.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the implementation of the maintenance rule for the systems and/or equipment problems listed below, completing two inspection procedure samples:

- Component Cooling Water System; and
- Service Water System.

The inspectors verified that the licensee identified, entered, and scoped component and equipment failures within the maintenance rule requirements. The inspectors also verified that the systems and equipment were properly categorized and classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed a sample of station logs, maintenance work orders, action requests, functional failure evaluations, unavailability records, and a sample of condition reports to verify that the licensee identified 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 (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and assessment of plant risk, scheduling, and configuration control during the following planned and emergent work activities, completing three inspection procedure samples:

- Safety Monitor Risk Assessment for October 6 through 10, 2003;
- Safety Monitor Risk Assessment for October 20 through 24, 2003; and
- Safety Monitor Risk Assessment for November 3 through 7, 2003.

In particular, the inspectors evaluated the licensee's planning and management of maintenance and verified 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. The inspectors also reviewed licensee actions to address increased on-line risk during these periods to verify that the actions were in accordance with approved administrative procedures.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors completed one inspection procedure sample during their review of the circumstances surrounding the licensee's declaration on December 12, 2003, that both trains of the Component Cooling Water System were inoperable. Operators declared both trains of the system inoperable due to the discovery of a component cooling water leak on the component cooling water radiation detector housing located on the common header of the component cooling water system piping. The radiation detector housing was a component cooling water system Class 3 piping pressure boundary. Consequently, the operators initiated a TS-required shutdown of the plant.

The inspectors observed portions of the reduction from full power to approximately 20 percent reactor power, when the licensee declared the component cooling water system operable. During the evolution, the inspectors observed control room communications, shift management oversight of the reactor power maneuvers, and operator alarm response.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following corrective action document operability evaluations, completing two inspection procedure samples:

- OPR000049; Trap 26 (Main Steam to Turbine Driven Auxiliary Feedwater Pump MS-102) Not Working Properly; and
- OBD010908; Possible Frazil Ice Blockage of Auxiliary Intakes Makes Service Water System Operable But Degraded.

The inspectors reviewed design basis information, the Updated Final Safety Analysis Report and TS requirements to verify the technical adequacy of the operability evaluations. The inspectors verified that system operability was properly justified and that the system remained available, such that no unrecognized increase in risk occurred.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed previously identified operator workarounds, equipment deficiency logs, and control room deficiencies to verify that the cumulative effects did not create significant adverse consequences regarding the reliability, availability and operation of accident mitigating systems, completing one inspection procedure sample. The inspectors also assessed these cumulative effects on the ability to implement abnormal and emergency response procedures in a correct and timely manner.

The inspectors reviewed the planned actions to address operator workarounds to verify that the priority to resolve the deficiencies were appropriate when considering the potential impact on plant risk and safety. In addition, the inspectors reviewed emergent risk significant operator workarounds to determine whether the functional capability of a system or human reliability of an initiating event was affected. Finally, the inspectors reviewed condition reports regarding operator workarounds to verify that the corrective actions were prioritized and appropriate, commensurate with the safety significance of the issue.

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 installation of the component cooling water system minimum flow recirculation lines, completing one inspection procedure sample. Additionally, the inspectors observed portions of the installation and testing of the lines, reviewed acceptance testing results, and reviewed condition reports associated with the design change to verify that the licensee identified and documented problems at an appropriate threshold.

b. Findings

Introduction: The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for changes made to the facility. Specifically, the licensee 'screened out' of the 10 CFR 50.59 process a modification that included the addition of a minimum flow recirculation line to the component cooling water pumps. Subsequently, the inspectors identified and the licensee concurred that a safety evaluation was required for this modification.

Description: The licensee developed and installed Design Change Request DCR-3381, "Provide a Continuous Minimum (Recirculation) Flow Path for Component Cooling Water Pumps 1A and 1B," as a corrective action to a design weakness. In January 2002, the licensee identified that when the component cooling water pumps were operated in parallel, one component cooling water pump was inoperable. (This issue is discussed in detail in NRC Inspection Report 50-305/02-03.)

Component Cooling Water Pumps '1A' and '1B' were originally designed by the licensee and licensed by the NRC with a common suction and discharge header. In September 2003, the licensee installed a minimum flow recirculation line for each component cooling water pump with DCR-3381. The modification involved the placement of a new piping system which began at the discharge piping of one component cooling water pump and terminated at the suction piping of the opposite component cooling water pump. The new minimum flow recirculation pipe section (one for each pump) contained a strainer, two flow orifices, and manual valves for isolation of the minimum flow line.

The licensee completed Screening No. 03-089 for DCR-3381 on June 18, 2003, and concluded that the modification 'screened out' of the 10 CFR 50.59 process because the activity did not adversely affect the design function. As a result, the licensee determined a 10 CFR 50.59 safety evaluation was not needed for this modification. On June 30, 2003, the Plant Operations Review Committee reviewed and approved DCR-3381 and Screening No. 03-089, and the licensee subsequently installed the minimum flow recirculation modification. The minimum flow recirculation line was not placed in service, except for a period of approximately 8 hours on September 4, 2003, for system testing conducted in accordance with Procedure SOP-CC-31-30, "Component Cooling Pumps Recirculation Flow Verification."

On September 8, 2003, the inspectors questioned the licensee's failure to perform a safety evaluation, as required by 10 CFR 50.59, since modification could allow the failure of one component cooling water pump to potentially adversely affect the operation of the opposite train component cooling water pump in a new manner.
The inspectors also questioned whether the licensee had appropriately followed the guidance in Nuclear Energy Institute Standard NEI-96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, which the NRC endorsed in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments." Specifically, the inspectors noted that Section 4.2.1 of NEI 96-07, stated, in part that changes that would introduce a new type of accident or malfunction would 'screen in,' and that if a change has both positive and adverse effects, the change should be 'screened in' because the analyses necessary to demonstrate acceptability were beyond the scope/intent of 10 CFR 50.59 screening reviews. In addition, the inspectors consulted NRC Regional and Headquarters experts in this area, who independently reached the same conclusions as the inspectors.

The licensee concurred that a safety evaluation was required for this modification and initiated several condition reports regarding the issues raised by the inspectors. The licensee also administratively controlled the minimum flow recirculation isolation valves in the closed position until a safety evaluation was completed. The licensee completed the safety evaluation on October 3, 2003, and determined that the modification did not require prior NRC approval because of the design features engineered into the modification, including the safety-related strainer in the recirculation line which would prevent a pump failure from damaging the opposite train pump.

The licensee also completed Root Cause Evaluation RCE625, and determined that the root cause of this issue was that personnel involved with the development and review of the screening held longstanding misconceptions regarding the 10 CFR 50.59 process. In addition, the subject training failed to adequately communicate guidance associated with 10 CFR 50.59. The licensee's extent of condition review for this issue identified approximately 12 additional modifications which required further review to determine whether the 10 CFR 50.59 screenings were adequate.

<u>Analysis</u>: The inspectors determined that the licensee's failure to identify this modification required a safety evaluation and subsequent failure to perform a safety evaluation for the modification made to the component cooling water system was a licensee performance deficiency warranting a significance evaluation.

Because violations of 10 CFR 50.59 could impede or impact the regulatory process, these violations are typically dispositioned using the traditional enforcement process instead of the Significance Determination Process. However, the results of this violation were assessed using the Significance Determination Process. In this case, the licensee failed to perform a safety evaluation in accordance with 10 CFR 50.59 and had placed the new system in service for testing prior to the completion of the required safety evaluation.

The inspectors considered this issue to be of more than minor significance, because, if left uncorrected, the issue could become a more significant safety concern. The inspectors noted that the licensee's process failed to identify this issue for this design change at several levels, including: the initial design change development; subsequent design change review; Engineering Quality Review Team; and the final design change approval by the licensee's Plant Operations Review Committee. The inspectors also determined that the issue was of very low significance. The inspectors concluded this because the new system was placed in service for a short period of time for testing prior to the completion of the required safety evaluation, and the final safety evaluation completed by the licensee determined that the modification did not require prior NRC approval because of the design features engineered into the modification.

The inspectors also determined that the finding affected the cross-cutting area of Human Performance, because the failure to recognize a safety evaluation was required for this modification was a result of longstanding misconceptions regarding 10 CFR 50.59 screenings by engineering personnel and the failure to identify this issue during the licensee's design change and management reviews of the modification.

<u>Enforcement</u>: 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides for the

determination that the change, test, or experiment does not require a license amendment. Contrary to these requirements, the licensee failed to perform a written safety evaluation for changes to the facility as described in the Updated Final Safety Analysis Report documented in Design Change Request DCR-3381 completed June 30, 2003, and Screening No. 03-089 completed on June 18, 2003. Specifically, the modification was placed in service for a period of time on September 4, 2003, and the Design Change Request nor the screening provided a written safety evaluation that provided for the determination that the change, test, or experiment did not require a license amendment. The results of this violation were determined to be of very low safety significance; therefore, the inspectors determined this finding was classified as a Severity Level IV Violation of 10 CFR 50.59. Because this violation was of very low significance, non-willful, non-repetitive, and documented in the licensee's corrective action program as Condition Report CAP017969, "50.59 Review/Screening/Evaluation for DCR-3381," this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000305/2003008-01)

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post-maintenance testing activities associated with the following scheduled and emergent work activities, completing three inspection procedure samples:

- Turbine Driven Auxiliary Feedwater System Steam Supply;
- Component Cooling Water Pumps Recirculation Lines; and
- 1A Safety Injection Pump.

The inspectors verified 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 to verify that the impact of the testing was appropriately characterized. The inspectors also observed the performance of testing to verify the procedure was followed, and that all testing prerequisites were satisfied. Following the completion of each test, the inspectors walked down the affected equipment to verify removal of the test equipment and to ensure the equipment could perform the intended safety function following the test. The inspectors also reviewed the completed test data to ensure the test acceptance criteria were met for the post maintenance testing.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and reviewed the surveillance testing results for the following surveillances, completing four inspection procedure samples:

- Diesel Generator B Availability Test;
- Channel 4 (Yellow) Instrument Channel and Nuclear Power Range Tests;
- Train A Service Water Pump and Valve Test - Inservice Test; and
- Train A Residual Heat Removal Pump and Valve Test - Inservice Test.

The inspectors verified that the equipment could perform the intended safety function and that the surveillance tests satisfied the requirements contained in TSs 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.

The inspectors observed portions of the test to verify the following attributes: performance of the test in accordance with prescribed procedures; completion of test procedure prerequisites; and verification that the test data was complete, appropriately verified, and met the acceptance criteria of the test. Following the completion of the tests, when applicable, the inspectors walked down the affected equipment to verify test equipment removal and to confirm the equipment tested was in an operable condition.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

.1 Temporary Change to Component Cooling Water Piping Boundary

a. Inspection Scope

The inspectors reviewed temporary plant modification TCR 03-36, "Temporary Change to Component Cooing Water Pressure Boundary at the 12-inch 150# Studding Outlet Flange for Radiation Monitor R-17," completing one inspection procedure sample. The inspectors reviewed the modification documentation, which included the engineering analysis, engineering calculations and associated evaluations conducted in accordance with 10 CFR 50.59.

The inspectors verified that the temporary modification did not adversely impact other safety-related equipment and that the modification was controlled in accordance with the licensee's administrative procedures. The inspectors also verified that the modification did not affect system operability or availability.

b. Findings

Introduction: The inspectors identified a Green finding associated with a Non-Cited Violation of 10 CFR Appendix B, Criterion III, "Design Control," for the licensee's failure to check the adequacy of the design in Temporary Change TCR 03-036, in that, the design review failed to confirm the structural integrity of the new pressure boundary

established for the studding outlet. Consequently, the licensee performed non-destructive examinations and additional flaw and engineering analyses to confirm the adequacy of the new design.

Description: On December 12, 2003, operators declared both trains of the Component Cooling Water (CCW) System inoperable, due to the discovery of a small CCW leak on the CCW Radiation Detector R-17 housing (welded studding outlet). The housing was integral to the common header of the CCW system piping. The licensee attempted to resolve the issue by designating a new Class 3 pressure boundary, through a temporary design change, to the welded studding outlet, rather than pursuing a non-Code repair due to the unique design of the housing. In addition, the licensee was not able to definitively determine and characterize the underlying flaw which caused the leak, and, therefore, could not perform a Class 3 Code repair.

On December 12, 2003, the licensee approved TCR 03-036 associated with the change in pressure boundary from the CCW piping to the welded studding outlet because of the through-wall leak in this Class 3 CCW common suction line. Specifically, the licensee design change included installation of a threaded plug into an air test port in the studding outlet to stop the through-wall leak in the CCW common suction line. This resulted in changing the Class 3 pressure boundary to the outer fillet weld and plugged air test port of the studding outlet and underlying pipe wall.

On December 15, 2003, the inspectors, in conjunction with NRC Region-based inspectors, concluded that the licensee failed to determine in the analysis, the existing condition of the piping base metal at the periphery of the new pressure boundary. The inspectors were concerned that if the original flaw which had caused the leak existed in or had propagated to the piping base metal under the studding outlet outer fillet weld, the unknown flaw could potentially challenge the structural integrity of this piping.

In addition, the licensee stated in Attachment 1 of TCR 03-036 that the ring reinforcement and the large fillet weld at the studding outlet would arrest any flaw growth. The inspectors noted that the licensee's conclusion on flaw growth was not supported by any detailed stress analysis which evaluated the residual tensile stress profiles that could develop in the piping base metal under the fillet weld. Welding residual tensile stress that existed near the fillet weld could potentially support crack propagation in the pipe wall beneath the fillet weld on the studding outlet. The inspectors also noted that as part of TCR 03-036, the licensee had not performed any analysis of the new pressure boundary, such as non-destructive examinations, to confirm the licensee's presumption that the new pressure boundary established under this design change was free of flaws.

Finally, the inspectors noted that the design description for TCR 03-036 stated that

Calculation C11550, “Weld Check of Studding Outlet Associated with R-17,” qualified the threaded plug for use as a CCW system pressure boundary component; however the calculation did not address the threaded plug. The inspectors questioned whether there was an over-pressurization concern with any fluid trapped between the inner-diameter and outer-diameter welds of the studding outlet with the air test port plugged.

The inspectors determined that, since the licensee had indications of leakage from the original pressure boundary and was unable to characterize the cause or subsurface extent of the degradation, the presumption that the new pressure boundary was unflawed and intact was not justified. Therefore, the inspectors concluded that the licensee’s design reviews for temporary change TCR 03-036 were inadequate, in that, the licensee failed to confirm the structural integrity of the new pressure boundary.

The licensee concurred with the inspectors’ assessment of the design change, and initiated corrective actions to address the issues. These actions included: analysis of the plugging of the air test port; a stress analysis which preliminarily concluded the residual tensile stress profiles were adequate (the final analysis was in progress at the end of the inspection); and non-destructive examinations on the outer fillet weld of the studding outlet and base metal. In addition, the licensee began a root cause evaluation for this issue.

Analysis: The inspectors determined that the licensee’s failure to perform adequate design reviews to confirm the structural integrity of the new pressure boundary for Temporary Change TCR 03-036 was a licensee performance deficiency warranting a significance evaluation.

The inspectors considered this issue to be of more than minor safety significance, because, if left uncorrected, the issue could become a more significant safety concern. In addition, the inspectors concluded that the finding was greater than minor because the finding involved the design control attribute of the mitigating systems cornerstone and affected the mitigating systems objective of ensuring the capability of the CCW system in response to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the temporary design change relied on unsupported assumptions that could have impacted the structural integrity of the CCW suction line.

The inspectors evaluated the finding using Inspection Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding was a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18. Therefore, the inspectors determined that the finding was of very low safety significance (Green).

Enforcement: 10 CFR Part 50, Appendix B, Criterion III "Design Control" requires, in part, that measures provide for checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable test program. In addition, design changes, including field changes, shall be subject to design control measures commensurate to those applied to the original design. Contrary to these requirements, the licensee failed to provide for checking of the adequacy of the design in Temporary Change TCR 03-036, approved December 12, 2003, in that, the design review failed to confirm the structural integrity of the new pressure boundary established for the studding outlet. Consequently, the licensee performed non-destructive examinations and additional flaw and engineering analyses to confirm the adequacy of the new design. Therefore, the inspectors determined this finding was a violation of 10 CFR Part 50, Appendix B, Criterion III. Because this violation was of very low significance, non-willful, non-repetitive and documented in the licensee's corrective action program as Condition Report CAP019295, "Weaknesses in Technical Evaluation for TCR," this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000305/2003008-02)

.2 Shield Block Temporary Modification

a. Inspection Scope

The inspectors reviewed the modification documentation and associated 10 CFR 50.59 evaluation for temporary plant modification TCR 03-16, "Crack in Shield Block #2," completing one inspection procedure sample.

The inspectors verified that the temporary modification did not adversely impact other safety-related equipment and that the modification was controlled in accordance with the licensee's administrative procedures. The inspectors also verified that the modification did not affect system operability or availability. In addition, the inspectors reviewed condition reports to verify that temporary modifications problems were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Drill Evaluation

a. Inspection Scope

The inspectors observed and reviewed the licensee's drill and exercise performance evaluations for emergency response activities associated with licensee drills which occurred in the control room simulator and observed a drill which included the activation of the Technical Support Center and the Emergency Off-Site Facility, completing two inspection procedure samples. The evaluations and critiques documented drill participant performance with regard to proper classification and notification of scenario events, and the adequacy and timeliness of notifications for protective action recommendations associated with those scenarios. On December 4, 2003, the inspectors observed the drill player and control briefs, and the Technical Support Facility critique.

In both instances, the inspectors verified that missed opportunities regarding timely notifications were identified during the licensee's drill critiques and that identified weaknesses were captured in the corrective action program.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

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

a. Inspection Scope

The inspectors reviewed the licensee's performance indicators for the occupational exposure cornerstone, completing one inspection procedure sample. The inspectors reviewed records to determine whether the licensee identified any occupational exposure control cornerstone performance indicators during the previous five calender quarters. When performance indicators had been identified, the inspectors verified that the licensee evaluated the conditions surrounding the performance indicators and entered identified problems into the corrective action program for resolution.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and RWP Reviews

a. Inspection Scope

The inspectors reviewed the adequacy of the licensee's internal dose assessment process for internal exposures greater than 50 millirem committed effective dose equivalent. This review completed one inspection procedure sample.

The inspectors also reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within spent fuel or other storage pools. This review completed one inspection procedure sample.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed licensee records to determine if any PI events had occurred since the last inspection and to determine if any of these PI events involved dose rates greater than 25 R/hr at 30 centimeters or greater than 500 R/hr at 1 meter. If PI events had occurred, the inspectors determined what barriers had failed and if there were any barriers left to prevent personnel access. Unintended exposures greater than 100 millirem total effective dose equivalent (or >5 rem shallow dose equivalent or greater than 1.5 rem lens dose equivalent), were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure. This review represented one sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors interviewed the Radiation Protection Manager concerning high dose rate/high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to verify that any procedure modifications did not substantially reduce the effectiveness and level of worker protection. This review completed one inspection procedure sample.

b. Findings

No findings of significance were identified.

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

.1 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed the licensee’s corrective action program to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed. This review completed one inspection procedure sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors sampled the licensees submittals for the performance indicators listed below, which completed three inspection procedure samples:

- Safety System Unavailability for High Pressure Injection;
- Safety System Functional Failures; and
- RETS/ODCM Radiological Effluents.

The inspectors used performance indicator definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, to verify the accuracy of the PI data. The inspectors reviewed corrective action documents, monthly operating reports, completed surveillance procedures, control room logs, and licensee event reports to independently verify the data that the licensee had collected from October 2002 through October 2003. The inspectors also independently re-performed calculations for system unavailability when applicable.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Diesel Generator "B" Failure to Start

a. Inspection Scope

During review of the circumstances surrounding Licensee Event Report 50-305/2003-002-00, "Shutdown Initiated - Diesel Generator Failed Start Test - Unusual Event - Caused by Relay Start Failure," the inspectors evaluated the adequacy and effectiveness of the licensee's corrective actions, completing one inspection procedure sample. The inspectors reviewed corrective action program documents generated by the licensee as a result of this event and interviewed licensee personnel responsible for the implementation of the proposed corrective actions. Details regarding the inspectors' followup of the circumstances of the event are described in Section 4OA3.1 of this inspection report.

b. Findings

Introduction The inspectors identified a finding of very low safety significance associated with a Non-Cited Violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Actions," for the licensee's failure to take adequate corrective actions in response to the installation of non-conforming cranking cutout relays which prevented energizing of the diesel generator engine start relay. The licensee's corrective actions for this condition adverse to quality addressed routine surveillance procedures, but did not consider the licensee's Emergency Operating Procedures to ensure the Emergency Diesel Generators would remain operable following Diesel Generator Shutdowns as directed by those procedures.

Description The licensee's corrective actions in response to the Emergency Diesel Generator "B" failure to start on February 26, 2003, included a revision to all diesel operating and surveillance test procedures to perform a continuity check of the cranking cutout relay contacts in each diesel engine start relay circuit each time the diesel is shut down. The inspectors reviewed selected diesel operating and surveillance test procedures to confirm that the revision had been completed. However, review of licensee Emergency Operating Procedures did not identify a similar requirement to perform a continuity check of the relay contacts in each diesel engine start relay circuit each time the diesel is shut down. The continuity checks of the relay contacts was necessary to ensure that the contacts had not been oxidized during the previous start sequence, preventing them from closing in the event of a subsequent engine start signal. Interviews of operations personnel and licensee management representatives confirmed that the licensee's corrective actions were limited to normal operating and surveillance procedures and did not envelope the Emergency Operating Procedures.

Analysis The inspectors determined that the licensee's failure to ensure that corrective actions to address the failure of Emergency Diesel Generator "B" failure to start on February 26, 2003, included abnormal as well as normal operating procedures was a licensee performance deficiency warranting a significance evaluation. This inspector-identified finding was greater than minor because the failure affected the mitigating systems attributes of equipment performance and procedure quality. In addition, the finding affected the cornerstone objective of ensuring the reliability and capability of the engineered safeguards systems that respond to initiating events to prevent undesirable consequences.

The inspectors evaluated the finding using Inspection Manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding:

- was not a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18;
- did not represent an actual loss of safety function of a system;
- did not represent an actual loss of a safety function of a single train for greater than TS outage time;
- did not represent an actual loss of a safety function of one or more Non-TS trains of equipment designated as risk significant;
- did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event;
- did not involve the total loss of any safety function that contributed to core damage accident sequences initiated by seismic events; and
- did not involve the loss or degradation of equipment or function designed to mitigate a seismic initiating event.

Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, as of November 2003, the licensee failed to promptly correct a condition adverse to quality involving the performance of a continuity check of the cranking cutout relay contacts in each diesel engine start relay circuit each time the diesel is shutdown. Although the corrective action was addressed in the licensee's operating and surveillance test procedures, the requirement to perform continuity checks was not included as part of the Emergency Operating Procedures. Such checks were necessary to ensure that the contacts had not become oxidized during the previous engine start sequence, preventing the diesel generator from starting following subsequent engine start signals. The inspectors determined this finding was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because this violation was of very low safety significance (Green) and documented in the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000305/2003008-03)

.2 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that issues were entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors also reviewed all condition reports written by licensee personnel during the inspection quarter. Minor issues entered into the licensee's corrective action system as a result of inspectors' observations are included in the list of documents reviewed which is attached to this report.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 50-305/2003-002-00: Shutdown Initiated - Diesel Generator Failed Start Test - Unusual Event - Caused by Relay Start Failure.

Introduction: A Green finding associated with an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was self-revealed when Emergency Diesel Generator 'B' failed to start on February 26, 2003, during TS required daily test, while Emergency Diesel Generator 'A' was out of service for planned maintenance. Emergency Diesel Generator 'B' failed to start because a pair of electrically open contacts on a cranking cutout relay prevented the engine start relay from energizing. The licensee installed the cranking cutout relay via a design change in 1998, and subsequently determined the performance ratings of the new relay did not match the original design specifications.

Description: On February 25, 2003, at 2:39 a.m. with the reactor operating at 100 percent power, the licensee removed Emergency Diesel Generator 'A' from service to perform 18-month periodic maintenance. Technical Specification 3.7.b.2 requires that during power operation, one diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure operability. At 1:59 a.m. local time, on February 25, 2003, the licensee successfully tested Emergency Diesel Generator ‘B’ to ensure operability.

On February 26, 2003, at 12:17 a.m. the licensee attempted to start the Emergency Diesel Generator ‘B’ for the required daily testing. The generator failed to start and with both diesel generators out of service, the licensee declared an Unusual Event at 12:22 a.m. The licensee initiated shutdown of the reactor, as required by TSs, at 1:07 a.m.

The licensee immediately began troubleshooting and established the cause of the diesel generator failing to start was an electrically open pair of contacts on the cranking cutout relay, which prevented energizing the engine start relay. The licensee replaced the cranking cutout relay, successfully tested Emergency Diesel Generator ‘B’, and returned the generator to service at 6:24 a.m. Licensee operators halted the power reduction and terminated the Unusual Event at 6:56 a.m. on February 26, 2003.

The licensee sent the failed relay to a vendor laboratory for full analysis of the failed cranking cutout relay contacts. The vendor determined that the electrically open failure was due to buildup of an insulating silver/copper oxide on the contact surfaces, as a result of material transfer between the contacts. The material transfer occurred due to arcing across the relay contacts, as the contacts broke the electrical circuit for the diesel engine start relay. The vendor further determined that the engine start relay dissipated the energy through arcing across the opening relay contacts. This phenomenon occurred normally as the diesel generator accelerated from 200 revolutions per minute (rpm) to reduced or normal operating speeds. The relay subsequently de-energized upon diesel shutdown (engine speed below 200 rpm) and the contacts of the relay re-closed after a 10 second delay. When the cranking cutout relay was closed, the engine start relay circuit was ready for the next diesel demand.

The licensee then monitored the voltages and currents on each cranking cutout relay, to quantify the normal voltage and current transients experienced by the relay contacts. The licensee compared this information to the rated inrush currents for the cranking cutout relay. The licensee determined that the cranking cutout relay was rated for 1.25 amperes at 138 Volts-Direct Current (VDC) and 2.5 amperes at 125 VDC; however, the actual transients experienced by the relay during normal operation was 260 VDC and 4 to greater than 5 amperes, which was significantly in excess of the cranking cutout relay rating. The normal transient conditions acting on the relay contacts caused damage to the cranking cutout relay that resulted in a buildup of an insulating oxide layer and failure of the contacts in the open position.

In 1998, the licensee implemented Design Change DCR 2965, “Replace Diesel Generator Time Delay Relays,” and replaced the original design cranking cutout relay (Agastat Model 2400) with a new cranking cutout relay (Agastat E7000 series time delay relay). The licensee’s design change documentation, engineering analysis and safety review did not take into consideration the transient nature of the engine start relay circuit upon de-energization, nor analyze/document inductive or inrush currents of the circuits where the new E7000 series Agastat contacts were installed. Additionally, the second level review did not consider this potential failure or analyze for transient conditions. The inspectors determined that engineering transient analysis knowledge and application, and the design consideration process had not adequately considered the transient conditions that existed in the original and modified circuits.

Analysis: The inspectors determined that the licensee's failure to ensure the performance ratings of the new cranking cutout relay installed in 1998 matched the original design specifications for the relay 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 relays replaced in the engine start relay circuit in 1998 were qualified for all normal and transient loads that existed in the circuit affected the mitigating systems attributes of design control. In addition, the finding affected the cornerstone objective of ensuring the reliability and capability of the engineered safeguards systems that respond to initiating events to prevent undesirable consequences.

The inspectors evaluated the finding using Inspection Manual Chapter 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding:

- was not a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18;
- did not represent an actual loss of safety function of a system;
- did not represent an actual loss of a safety function of a single train for greater than TS outage time;
- did not represent an actual loss of a safety function of one or more Non-TS trains of equipment designated as risk significant;
- did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event;
- did not involve the total loss of any safety function that contributed to core damage accident sequences initiated by seismic events; and
- did not involve the loss or degradation of equipment or function designed to mitigate a seismic initiating event.

Therefore, the finding was determined to be of very low safety significance (Green).

The inspectors also determined that the finding affected the cross-cutting area of Human Performance, because the design change did not adequately examine the transient nature of the engine start relay circuit upon de-energization. The design description and safety review did not analyze or document inductive or inrush currents of the circuits where the new E7000 series Agastat contacts were being installed. Additionally, the second level review of the design change did not consider this potential failure or analyze for transient conditions..

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated

into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components. Contrary to this requirement, Design Change DCR 2965, "Replace Diesel Generator Time Delay Relays," did not include provisions to assure that appropriate quality standards were specified and included for the new relays installed in the cranking cutout relay circuits for the emergency diesel generators. As a result, relays which did not meet the circuit performance requirements were installed which prevented the "B" emergency diesel generator from starting when called upon on February 26, 2003. The inspectors determined this finding was a violation of 10 CFR Part 50, Appendix B, Criterion III. Because this violation was of very low safety significance (Green) and documented in the licensee's corrective action program as Condition Report CAP015007, "Diesel Generator B Failed to Start as Required," this finding is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000305/2003008-04)

.2 <u>(Closed) Licensee Event Report (LER) 50-305/2003-004-00:</u> Failure to Log Nuclear Instrumentation Quadrant Power Tilt as Required by Tech Spec 3.10.j Due to Procedural Inadequacy

The inspectors reviewed the licensee's report and support documentation, including the root cause evaluation report, and evaluated the scope and adequacy of the licensee's corrective actions for the LER. The event involved receipt of a control room annunciator, "Lower Quadrant Power Tilt Ratio High," during excore nuclear instrument calibration. The operators referred to the applicable response and action procedures but, believing the alarm did not reflect a real power tilt, and knowing three channels had not been affected by the calibration, the operators considered the tilt monitoring system operable. However, while in the alarm condition, the system was not operable because the system could no longer detect and annunciate an actual power tilt. With the system not operable, the operators were required by TSs to log the individual detector outputs and the tilt. An oncoming operating crew re-assessed the event and recognized the condition. The second crew initiated the required logging and the manual tilt calculation verified that an actual power tilt did not exist.

The licensee determined that inadequate procedural instructions caused the failure to implement this TS requirement. The licensee revised the procedure to clarify the requirement. Additional licensee short-term corrective and preventive actions focused on training and reinforcement of management expectations. The inspectors noted that LER 2003-001-00, documented in NRC Inspection Reports 50-305/03-02 and 50-305/03-04, also involved a failure to perform a TS required action when the flux tilt monitoring system received spurious actuations and was temporarily out-of-service. The inspectors concluded there were insufficient similarities in the event causes to expect corrective actions from LER 2003-001-00 to have prevented LER-2003-004-00.

See Section 4OA7 for details regarding a license-identified violation associated with this LER.

.3 (Closed) LER 50-305/2003-005-00: Technical Specification Table 4.1-1, Item 25, requirement for the quarterly test of radiation protection portable survey instruments was not met.

On October 1, 2003, Nuclear Management Company Nuclear Oversight personnel discovered the TS Table 4.1-1, Item 25, requirement for the quarterly test of radiation protection portable survey instruments was not met for several instruments in July 2003. The test consisted of exposing the survey instrument detector to a source of known strength, and verifying the proper response for each range/scale of the instrument. The source check/tests were coordinated with the calibration of the survey instruments. Since calibrations were typically 6 months, the source checks were done during the quarter following the quarter in which the calibration was performed. When the calibration schedules for some instruments were changed to 1 year, some of those instruments were not rescheduled for source checks the third or fourth quarter following calibration. The licensee identified that the root cause for the missed source checks was the failure of the existing process to properly control required TS requirements and the inadequate degree of instructional details in the governing procedure SP-80-060, "Portable Radiation Survey Instrument Checks and Tests," Revision V. Corrective actions included removing from service effected instruments, listing and tracking all survey instruments as needing a source check regardless of the need for calibration that month, and revising SP-80-60 to provide clearer direction and guidance for documentation of test data. The failure to source check survey instruments per TS constitutes a violation of a minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the failure to source check the survey instruments in CAP 18310.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R17.1 of this report had, as the primary cause, a human performance deficiency, in that, the failure to recognize a safety evaluation was required for this modification resulted from longstanding misconceptions regarding 10 CFR 50.59 Screenings by engineering personnel and the failure to identify the issue during the licensee's design change and management reviews of the modification.

.2 A finding described in Section 4OA3.1 of this report had, as the primary cause, a human performance deficiency, in that, the licensee installed a new cranking cutout relay in Emergency Diesel Generator 'B' via a design change in 1998, which was not engineered to match the original design specifications of the cranking cutout relay.

4OA5 Other Activities

.1 (Closed) Temporary Instruction (TI) 2515/154: Spent Fuel Material Control and Accounting at Nuclear Power Plants. The inspectors completed Phase I of the subject TI with no further inspection needed.

4OA6 Meetings

.1 Quarterly Exit Meeting

On January 5, 2004, the resident inspectors presented the inspection results to Mr. Davison 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

An interim exit meeting was conducted for the Radiation Protection inspection with Mr. Kyle Hoops on December 12, 2003.

4OA7 Licensee-Identified Violations

The following violations of very low significance were identified by the licensee and are violations of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Mitigating Systems

Additional information regarding this licensee-identified violation is contained in Section 4OA3.2, “(Closed) Licensee Event Report (LER) 50-305/2003-004-00,” of this report. Technical Specification 3.10.j requires, in part, that, “If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs to the quadrant power tilt shall be logged once per shift, and after a load change of >10 percent of rated power or after 24 steps of control rod motion.” Contrary to this requirement, when the lower quadrant power tilt monitor became inoperable on September 2, 2003, because of a standing alarm condition, individual upper and lower calibrated outputs were not logged once per shift for a period of two shifts, until September 3, 2003. The licensee documented this issue in the corrective action system as Condition Report CAP017903, “Confusion Encountered During the Implementation of A-NI-48.” The inspectors evaluated this licensee identified violation using Inspection Manual Chapter 0609, “Significance Determination Process,” Appendix A “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding was of very low safety significance (Green).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Nuclear Management Company, LLC

T. Coutu, Site Vice President
K. Hoops, Site Director
K. Davison, Plant Manager
L. Armstrong, Engineering Director
S. Baker, Manager, Radiation Protection
M. Fencl, Security Manager, Kewaunee/Point Beach
L. Gerner, Acting Regulatory Affairs Manager
G. Harrington, Licensing
B. Presl, NMC Security Consultant
S. Putman, Assistant Plant Manager, Maintenance
R. Repshas, Manager, Site Services
J. Riste, Licensing Supervisor
J. Stafford, Superintendent, Operations

NRC Personnel

J. Lamb, Project Manager, NRR
T. McMurtray, Senior Project Manager, NRR
R. Daley, Reactor Engineer
M. Holmberg, Senior Reactor Engineer
P. Lougheed, Senior Reactor Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000305/2003008-01	NCV	Green. Non-cited Violation of 10 CFR 50.59, for the failure to perform a written evaluation, as required, for a modification to the component cooling water system. (Section 1R17.1)
05000305/2003008-02	NCV	Green. Non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the failure to provide for the checking the adequacy of design for the temporary modification which changed the component cooling water system pressure boundary. (Section 1R23.1)
05000305/2003008-03	NCV	Green. Non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to take appropriate immediate corrective actions to address the reliability issues associated with the incorrect cranking cutout relay installed in the Emergency Diesel Generators. (Section 4OA2.1)
05000305/2003008-04	NCV	Green. Non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the failure to install the appropriate cranking cutout relay in the Emergency Diesel Generator System in 1998 which matched the original design specifications. This resulted in the failure of the 'B' Emergency Diesel Generator to start in February 2003. (Section 4OA3.1)

Closed

05000305/2003008-01	NCV	Green. Non-cited Violation of 10 CFR 50.59, for the failure to perform a written evaluation, as required, for a modification to the component cooling water system. (Section 1R17.1)
05000305/2003008-02	NCV	Green. Non-cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the failure to provide for the checking the adequacy of design for the temporary modification which changed the component cooling water system pressure boundary. (Section 1R23.1)
05000305/2003008-03	NCV	Green. Non-cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to take appropriate immediate corrective actions to address the reliability issues associated with the incorrect cranking cutout relay installed in the Emergency Diesel Generators. (Section 4OA2.1)

Closed (Continued)

05000305/2003008-04	NCV	Green. Non-cited Violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the failure to install the appropriate cranking cutout relay in the Emergency Diesel Generator System in 1998 which matched the original design specifications. This resulted in the failure of the ‘B’ Emergency Diesel Generator to start in February 2003. (Section 4OA3.1)
50-305/2003-002-00	LER	Shutdown Initiated - Diesel Generator Failed Start Test - Unusual Event - Caused by Relay Start Failure. (Section 4OA3.1)
50-305/2003-004-00	LER	Failure to Log Nuclear Instrumentation Quadrant Power Tilt as Required by Tech Spec 3.10.j Due to Procedural Inadequacy. (Section 4OA3.2)
50-305/2003-005-00	LER	Technical Specification Table 4.1-1, Item 25 Requirement for the Quarterly Test of Radiation Protection Portable Survey Instruments was Not Met. (Section 4OA3.3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

<u>1R01</u> <u>Adverse Weather Protection</u>

Operations Department Instruction - Cold Weather Operation; Revision 4; November 10, 2003

E-0-05; Operating Procedure - Natural Disaster; Revision J; October 29, 2001

N-CW-04; Operating Procedure - Circulating Water System; Revision AD; October 23, 2003

FFP-08-09; Fire Plan Procedure - Barrier Control; Revision F; October 23, 2003

OTH009698; SOER 2002-01 Severe Weather

CE007660; Condition Evaluation - Procedure E-0-05 Observations by NRC Routine Inspection; May 8, 2001

PCR008827; Procedure Change Request - Procedure FFP-08-09 Should Be Revised to Include Additional Barriers and Require an Evaluation to Determine If Conditions Allow the Barriers to Be Opened

CE001400; Condition Evaluation - Guidelines/Procedures Do Not Exist to Prepare Plant Equipment for Adverse Weather Conditions....

CAP001804; Guidelines/Procedures Do Not Exist to Prepare Plant Equipment for Adverse Weather Conditions....

CAP000234; Cold Weather Ops - 1/3/02 Resident Inspector Exit Feedback

PCR007742; Develop a GNP to Provide a Reference to All of the Plant Procedures Associated with Preparation for Hot/cold Weather Operations

CAP014216; Operations Instructions Do Not Address Previous NRC Concerns for Adverse Weather

CE011469; Operations Instructions do not Address Previous NRC Concerns for Adverse Weather

OTH010295; Address Remaining Concerns in ODI on Cold Weather Operations

PCR011019; Include Frazil Ice Operations in GNP 12.06.01, Cold Weather Operations

CAP018586; Adverse Weather Protection Activities Have Not Been Timely

1R04 Equipment Alignment

N-FW-05B-CL; Operating Procedure - Auxiliary Feedwater System Pre-startup Checklist; July 26, 2001

N-DGM-10-CLA; Diesel Generator A Prestartup Checklist; Revision I

Degraded Equipment Log; December 1, 2003

System 05B; Work Request Report - Auxiliary Feedwater System; October 29, 2003

System 05B; Open Work Order Report Sorted by Equipment Number, Auxiliary Feedwater System; October 29, 2003

1R05 Fire Protection

PMP-08-30; FP-CO2 System Inspection and Dry Test (QA-1); Revision K; December 4, 2001

PFP-22; Fire Plan Drawing - RHR Hx Area, Component Cooling water Pump Area, Letdown and Seal Water Filter Area, and RWST and Valve Gallery

Fire Zone Summaries for Fire Zones TU-92, TU-97, TU-98, TU-95A, and AX-23B

FPP-08-03; Fire Plan Qualifications and Training; Revision E

FPP-08-10; Fire Drills; Revision B

FPP-08-11; Fire Brigade/Team Equipment Inspection; Revision E

E-FP-08; Emergency Operating Procedure - Fire; Revision AH

Fire Drill Pre-Plan No. 2003-23; Scenario No. FBD-012

CAP019300; Propane Leak in TSC Expansion

1R11 Licensed Operator Requalification

Simulator Exercise Guide LRC-03-SE601, “LB LOCA, Containment Sump Recirc”; Revision A; October 8, 2003

Simulator Instructor Evaluation Form; LRC-03-SE601, “LB LOCA, Containment Sump Recirc”; October 22, 2003

1R12 Maintenance Implementation

Maintenance Rule Quarterly Report; January 1, 2003 to March 31, 2003

Maintenance Rule Quarterly Report; April 1, 2003 to June 31, 2003

KNPP Maintenance Rule System Basis; Service Water, Revision 5

KNPP Maintenance Rule Scoping Questions; Service Water, Revision 2

KNPP Maintenance Rule Performance Criteria; Service Water, Revision 3

Mentoring/Position Specific Guide ES.MRP.M.002K, "Maintenance Rule Evaluations - Degraded Equipment"; Revision A

Mentoring/Position Specific Guide ES.MRP.M.003K, "Maintenance Rule Evaluations - System Performance"; Revision A

WO 02-001204-000; Install New Actuator (Handle) for Valve SW301A; January 14, 2002

WO 02-007187-000; On 4/27/02 at 2300 We Had a Lake Weed Intrusion. Service Water Pump B2 Strainer DP was at 8.5 psid and the Strainer Was Not in Backwash; April 28, 2002

WO 02-007188-000; On 4/27/02 at 2300 We Had a Lake Weed Intrusion. Service Water Pump A2 Strainer DP was at 7 psid and the Strainer Was Not in Backwash; April 28,2002

WO 02-012810-000; During Performance of SP 02-138B Valve SW-903D was Cycled Closed for Timing. The Valve Appears Stuck in Mid-position; August 1, 2002

WO 03-004176-000; SW-901C-1 Actuator-Header 1C Shroud Cooling Coil C/D Bypass; April 13, 2003

MRE000215; Unable to Establish Service Water Flow Through the Bechtel Chiller Unit Which Is Connected to the C/D Shroud Cooling Service Water Piping; October 11, 2001

MRE001566; AFW Pump A Declared OOS Based on Radiography Results; August 19, 2002

MRE001745; Crew Questions Operability of Service Water Pump A1 Prior to Testing DG B; February 10, 2003

MRE001763; Air leak on Solenoid for SW-4A; February 27, 2003

MRE002104; Perform MRE on Issue Described in CAP 12618; August 15, 2003

CAP001182; Unable to Establish Service Water Flow Through the Bechtel Chiller Unit Which Is Connected to the C/D Shroud Cooling Service Water Piping; October 11, 2001

CAP012616; AFW Pump A Declared OOS Based on Radiography Results; August 16, 2002

CAP012618; Radiography Determines Pipe Thinning Occurring in the Service Water to AFW Pump A; August 16, 2002

CAP014632; Crew Questions Operability of Service Water Pump A1 Prior to Testing DG B; February 6, 2003

CAP014946; Air leak on Solenoid for SW-4A; February 26, 2003

CAP013104; CC Surge Tank Level XMTR 24041 Drift and Nonlinearity

MRE001603; CC Surge Tank Level XMTR 24041 Drift and Nonlinearity; September 30, 2002

CAP015777; Relief CC-652 Exceeded Acceptance Criteria

MRE001832; Relief CC-652 Exceeded Acceptance Criteria

CAP016090; Relief Valve CC-611B High Set Pressure

MRE001906; Relief Valve CC-611B High Set Pressure

CAP016092; Relief Valve CC-611A High Set Pressure

MRE001907; Relief Valve CC-611A High Set Pressure

CAP017643; Ft-23057 Found out of Tolerance

MRE002099; FT-23057 Found out of Tolerance

CAP017778; CC Surge Tank Level Decrease during SP-31-168B

MRE002118; CC Surge Tank Level Decrease during SP-31-168B

CAP017933; CCW Return Flow Indication Acting Erratic; F0619G

MRE002130; CCW Return Flow Indication Acting Erratic; F0619G

KNPP System Description (System No. 31); Component Cooling Water System (CC); Revision 2

Safety Monitor Risk Look Ahead – Kewaunee Plant Configuration Changes and Relative Core Damage Frequency; Revision 3; October 20, 2003

GNP-08.21.01; Risk Assessment for Plant Configurations; Revision D; August 22, 2002

CAP011828; System 31 Maintenance Rule (A)(1) Evaluation Required

ACE001817; Apparent Cause Evaluation - System 31 Maintenance Rule (a)(1) Evaluation Required

Maintenance Rule Expert Panel Meeting Minutes; August 15, 2002

Maintenance Rule Expert Panel Meeting Minutes; May 30, 2003

CAP011582; CCW Hx A Tube Leaks (5/4/02)

RCE000576; Root Cause Evaluation, Tube Leaks Identified in Component Cooling Water Heat Exchanger; August 10, 2002

Report No. 0205-04070; Stork Materials Technology; Failure Analysis of Heat Exchanger Tubes; May 9, 2002

MRE001496; CCW Hx A Tube Leaks (5/4/02)

OTH008524; CCW Hx A Tube Leaks (5/4/02)

CA008584; CCW Hx A Tube Leaks (5/4/02)

OTH008313; Enhance Eddy Current Inspection Methods

OTH008314; Implement Life Cycle Management Program (RCE 576)

CAP001101; CCW HX 1A Degradation

CAP011530; CC System Leak Developed Following SW Flush of CC System Hx

CAP011556; Evaluate ’B’ CCW Hx Condition after Finding Tube Cracks in ‘A’ CCW Hx

CAP011635; 1A CCW Heat Exchanger (135-081) Return End Erosion

CAP011747; Determine Cost/Risk Factors Between 2003 or 2004 CC Hx Replacement

Maintenance Rule System Basis – Component Cooling; Revision 5

Anatec Report; Kewaunee Nuclear Power Plant U1R25; Final Reports Eddy Current Inspections of the Balance of Plant Heat Exchangers; October 2001

GMP-137; Brush/Tube Scrubber Cleaning Heat Exchanger Tubes and Inspection; Revision G

CMP-31-02; (CC) Component Water Heat Exchanger Tube Cleaning (QA-1); Revision G

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

Safety Monitor Risk Assessment; Control Room Logs and Work Schedule for October 6 through 10, 2003

Safety Monitor Risk Assessment; Control Room Logs and Work Schedule for October 20 through 24, 2003

Safety Monitor Risk Assessment; Control Room Logs and Work Schedule for November 3 through 7, 2003

GNP 08.02.15; General Nuclear Procedure - Maintenance Activity Risk Assessment/ Management Process; Revision A

Licensee Document Entitled - Recommended Compensatory and Mitigation Measures for KNPP TSC Diesel Generator Out-of-Service for TSC Building Construction Work; October 28, 2003

Operations Department Instruction - Substation Work Control Process; October 21, 2003

System Operating Guide; Transformer Saturation Caused by Solar Magnetic Disturbances; May 1998

1R14 Personnel Performance During Non-Routine Plant Evolutions

N-O-03; Plant Operation Greater Than 35% Power; Revision AQ

N-O-04; 35 percent Power to Hot Shutdown Condition; Revision X

A-O-03; Rapid Power Reduction; Revision B

CAP019203; T_{ave}-T_{ref} Deviation Alarm

CAP019232; Questionable PPCS Indication for Containment WR Level (L8001A)

CAP019115; Turbine Noisy and Increased Vibes When No. 2 Control Valve Going Closed

1R15 Operability Evaluations

CAP018429; Trap 26 (Main Steam to MS-102) Not Working Properly

OPR000049; Trap 26 (Main Steam to MS-102) Not Working Properly

CAP018435; Steam Trap Monitoring Process

GNP 11.08.03; General Nuclear Procedure - Operability Determination; Revision B

OBD010908; Possible Frazil Ice Blockage of Aux Intakes Makes SW Syst. Operable but Degraded

EWR010909; Perform Analysis/Modification to Prevent Blockage of Aux Intakes with Frazil Ice

Updated Final Safety Analysis Report; Section 9.6.2

WO014649; Repair of 30-inch CW recirculation Pipe intake grid rake

WO #03-011362-000; Trash Rake of 30" Auxiliary Intake Line Does Not Work. Please Investigate/Repair. Reference CAP018495.

DCR014650; Review 30 inch CW Recirc Pipe Intake Grid Rake Design

OPERM-215; Flow Diagram - Circulating Water System

Design Drawing 237127A-S613-G; Circulating Water Intake and Discharge Plan

Design Drawing 237127A-S629-D; Circulating Water Intake Details - Grid Screen

Design Drawing 237127A-S622-F; Discharge Structure - Plans; Sections and Details

1R16 Operator Workarounds

General Nuclear Procedure GNP-01.31.01 - Plant Cleanliness and Storage; Revision E; October 17, 2002

Operator Work-Around 01-05; SD-3A, S/G A PORV, Rod Sleeve Hole That Aligns with Stem Hole Would Not Allow Valve to Open If Needed, When Pin Is Installed; May 31, 2001

Operator Work-Around 01-16; Generic Letter 96-06

Operator Work-Around 01-22; Power Supply Perturbations May Cause SW-1306A&B to Fail Open; November 28, 2001.

Operator Workaround 02-01; Pump Overheating Concerns Associated with Component Cooling Water Pumps During Two Pump Operation

Tagout 03-000934; Generic Letter 96-06. These Valves Must Remain Open When at or Above Hot Shutdown.

November 20, 1997; Licensee Correspondence to the NRC entitled Generic Letter 96-06 Response

C11483; Basis for KNPPs Response to NRC GL96-06 Containment Piping

Overpressurization Issue; Revision 0

CAP016904; Core Cooling CSF Periodically Changes from Yellow to Orange

CAP003292; Followup to Report on Overpressurization of Piping Systems

Control Room Deficiency Log - Danger Tags; December 17, 2003

Degraded Equipment Log; December 17, 2003

Operator Work-Around Status Sheet; December 15, 2003

1R17 Permanent Plant Modifications

DCR-3381; 50.59 Pre-Screening, Screening, Piping and Instrumentation Diagrams, and Design Description for DCR -3381, "Install Component Cooling Water Recirculation Lines," June 18, 2003

DCR-3381; Design Verification Assignment Form and associated documentation for DCR-3381; July 7, 2003

C11538; Calculation - Structural Qualification of the Component Cooling Pump Recirculation Line Strainer; Revision 0

CC-31-018; Calculation - Pipe Stress Analysis of Component Cooling Piping; Revision 0

CC-31-019; Calculation - Pipe Stress Analysis of Component Cooling Piping; Revision 0

SK-M-3381-1; Install CCW Pump Recirc. Lines Flow Diagram; Revision 0

SK-M-3381-2; Install CCW Pump Recirc. Lines Orifice Plate Details; Revision 0-1

Regulatory Guide 1.187; Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments; November 2000

1R19 Post-Maintenance Testing

PMP-06-01; MS-QA-1 Motor Operated Valve Maintenance; Revision J; September 18, 2001

SP-05B-333; Turbine Driven AFW Pump Recirculation Flow Test - IST; Revision B; October 2, 2003

SOP-CC-31-30; Component Cooling Pumps Recirculation Flow Verification; Revision B; September 15, 2003

DCR 3381; Install CCW Pump Recirc. Lines; September 24, 2003

SP-33-098A; Train A Safety Injection Pump and Valve Test - IST; Revision ORIG, A, B and C

CAP018383; Near Miss - Not Having an Approved 50.59 Evaluation In-Hand Delays Work

CAP018386; Tags on CCW Recirc. Line Cleared Prior to 50.59 Evaluation Being Signed by PORC

CAP018377; Unscheduled 4-Hour VT-2 Hold Time During SOP-CC-31-30

1R22 Surveillance Testing

SP-42-312B; Diesel Generator B Availability Test; Revision S; July 29, 2003

SP-47-316D; Channel 4 (Yellow) Instrument Channel Test; Revision Q; September 23, 2003

SP-48-003H; Nuclear Power Range Channel 4 (Yellow) N-44 Monthly Test; Revision L

SP-02-138A; Train A Service Water Pump and Valve Test - IST; Revision B (FREQ Q); April 24, 2003

SP-55-177; Inservice Testing of Pumps Vibration Measurements; Revision AA

CAP018453; Additional Guidance Needed Regarding D/G Primary Air Receiver Required Capacity

CAP018462; Performance Difficulties Encountered During Service Water Surveillance

CAP018881; Radiography Determines Pitting Occurring in Service Water Supply to Diesel Generator Heat Exchangers

CE011322; NRC Resident Questions ICS-2B Preconditioning

1R23 Temporary Plant Modifications

TCR 03-16; Crack in Shield Block #2; Revision ORIG

Calculation C11496; Temporary Support for Shield Block No. 2 for the Shield Building Equipment Hatch; Revision 0

Calculation 23452-C-046; Equipment Hatch Shield Blocks Anchorage Evaluation; Revision 0

Drawing 237127A-S-217; Shield Building Equipment Door Panels

WO-03-005270-000; Perform Repair to Shield Building Blocks In Accordance with TCR 03-016

CAP016231; Crack Identified in Containment Shield Block During Installation

TCR 02-01; Mechanical Stop on Valve CC-302

TCR 02-07 and TCR 02-08; Remove Valve Internals in Steam Generator Blowdown Piping

TCR 03-036; Temporary Change to Component Cooing Water Pressure Boundary at the 12-inch 150# Studding Outlet Flange for Radiation Monitor R-17

Design Description for TCR 03-036; December 12, 2003

Design Input Checklist Parts A and B for TCR03-036; December 12, 2003

50.59 Applicability Review, 50.59 Pre-Screening, 50.59 Screening for TCR 03-036; December 12, 2003

C11550; Calculation - Weld Check of Studding Outlet Associated with R-17; Revision 0

December 15, 2003, Licensee Document Addressing Whether There is an Over-Pressurization Concern with Any Fluid Trapped Between the ID Weld and OD Weld Now that the Air Test Hole is Plugged

WO 03-0142777; Work Order - Investigate and Repair a Leak Indication on the Component cooling Water Piping

Inservice Inspection Non-Recordable Indication Forms for Component R-17 dated April 6, 2003, May 11, 1999, May 21, 1996, and April 18, 1990

SP-31-248; Component Cooling Water System Pressure Test; Revision G

NEP-15.40; Ultrasonic Examination of Ferritic Piping for ASME Section XI, Appendix VIII, Inservice Inspection; Revision A

CAP019188; Residue From a Leak at R-17 Sample Chamber in Component Cooling Water Pipe

CAP019297; Lessons Learned from Activities to Address Leakage at R-17

1EP6 Drill Evaluation

Emergency Preparedness Drill Records for 1st, 2nd and 3rd Quarter 2003

December 9, 2003, Emergency Preparedness Drill Scenario and Completed Drill Paperwork

EPIP-EOF-08; Emergency Plan Implementing Procedure - Continuing Emergency Notification; Revision AA

EPIP-AD-07; Emergency Plan Implementing Procedure - Initial Emergency Notification; Revision AV

2OS1 Access Control to Radiologically Significant Areas

HP-03.008; Evaluations of Inhalations or Ingestions; Revision B

HP-03.009; Calculating Internal Dose from Whole Body Counter Results; Revision D

RE-24; Special Nuclear Materials Control; Revision K

Non-fuel Items in the Spent Fuel Pool; December 9, 2003

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

CAP012543; Investigate and Implement Ways to Improve Worker Awareness of Radiation Dose Alarms; August 12, 2002

CAP016767; NPS Contractor Was Observed Violating Radiological Posting; June 3, 2003

CAP017042; Chem Tech Observed Violating Radiological Posting; June 25, 2003

KSA-RP-03-01; KNPP Focused Self-Assessment Report; RP Organizational Effectiveness, Radioactive Materials Controls, Contamination Controls, Control of HRA/LHRA and SOER 200101 Response; October 2, 2003

Kewaunee Nuclear Power Plant; Quarterly Effectiveness Review Report; 1st, 2nd; 3rd Quarters 2003

4OA1 Performance Indicator Verification

CAP015007; Diesel Generator B Failed to Start as Required

RCE000607; Diesel Generator B Failed to Start as Required

GNP-03.18.01; General Nuclear Procedure - NRC Performance Indicators Reporting Instructions; Revision G; April 15, 2003

Second Quarter 2003; Safety Injection Trains A/B Unavailability Records

Second Quarter 2003; Residual Heat Removal Trains A/B Unavailability Records

SP-32-115-1; Liquid Effluents Dose Projection Monthly Data Sheet; 3rd and 4th Quarters 2002 and 1st, 2nd and 3rd Quarters 2003; Revision H

SP-32B-268; Semi-Monthly Boundary Dose Update - Gaseous Effluents; 3rd and 4th Quarters 2002 and 1st, 2nd and 3rd Quarters 2003; Revision H

40A2 Problem Identification and Resolution

Design Change Request (DCR) 2965, “Replace Diesel Generator Time Delay Relays,” February 2, 1998

4OA3 Event Followup

CAP015007; Diesel Generator B Failed to Start as Required

RCE000607; Diesel Generator B Failed to Start as Required

Letter dated October 31, 2003, from NMC to the NRC entitled, “Reportable Occurrence 2003-004-00"

RCE624; Root Cause Evaluation Report, “Confusion Encountered During Implementation of A-NI-48

CAP017903; Confusion Encountered During the Implementation of A-NI-48

A-NI-48; Operating Procedure - Abnormal Nuclear Instrumentation; November 11, 2003

CAP018167 and ACE002419; Excore detector current changes on Nuclear Instruments greater than past cycles

NRC-03-116; Reportable Occurrence 2003-005-00; December 1, 2003

RCE 629; Missed Surveillance for Portable Radiation Survey Instruments; December 4, 2003

CAP 018310; Missed Quarterly Instrument Check; October 1, 2003

SP-80-060; Portable Radiation Survey Instrument Checks and Tests; Revision V

Design Change Request (DCR) 2965, “Replace Diesel Generator Time Delay Relays,” February 2, 1998

NRC-03-116; Reportable Occurrence 2003-005-00;December 1, 2003

RCE 629; Missed Surveillance for Portable Radiation Survey Instruments; December 4, 2003

CAP 018310; Missed Quarterly Instrument Check; October 1, 2003

SP-80-060; Portable Radiation Survey Instrument Checks and Tests; Revision V

Condition Reports Initiated for NRC Identified Issues

CAP017965; UFSAR Table 9.3-5 Failure Analyses Not Specifically Noted in DCR 3381 Associated 50.59 Evaluation

CAP017969; 50.59 Review/Screening/ Evaluation for DCR-3381

CAP017971; DCR-3381 Component Cooling Water Recirculation Line Calculation C11538 Missing Input

CAP018297; Response to NRC Bulletin 2003-001: Impact of Debris Blockage Emergency Sump Recirculation

CAP018390; Hose Station No. 7 Found Unsealed

CAP018391; Revise Pre-Fire Plan No. 13 to Show Carbon Dioxide Portable Fire Extinguisher

CAP018392; Carbon Dioxide Portable Fire Extinguisher Does not have Maintenance Tag Attached

CAP018428; NRC Observed Issues During Inverter Maintenance

CAP018433; No CAPs Initiated on NRC Identified Issues

CAP018442; Loss of the Emergency Notification System Dedicated Line

CAP018570; KNPP Severity Level IV Violations Proposed by the NRC 10CFR 50.9

CAP018623; Control Room Logging of Dilution Actions

CAP018624; Fire Drill Scenario Presentation Prior to Drill

CAP018654; Need to Evaluate the Licensed Operator Medical Process

CAP018760; Requires MREs to be performed for WO 02-007187-000 and WO 02-007188-000.

CAP018873; System Description No. 05B Technically Inadequate

CAP018967; Potential for a Misclassification of an Operability Determination

CAP018984; Adverse Weather Protection - High Wind Conditions

CAP019295; Weaknesses in Technical Evaluation for TCR-03-036

CAP019335; TSC ENS Reports Inconsistent with Event Notices - 12/9 Drill

CAP019381; Emergency Operations Director Guidance When Control Room is Evacuated

CAP019400; Notices of Violation - Fitness for Duty

LIST OF ACRONYMS USED

CAP	Corrective Action Process
CCW	Component Cooling Water
CFR	Code of Federal Regulations
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
SDP	Significance Determination Process
VDC	Volts-Direct Current
WO	Work Order