February 14, 2005

EA 05-021

Mr. Craig Lambert
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 State Highway 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT
NRC INTEGRATED INSPECTION REPORT 05000305/2004009

Dear Mr. Lambert:

On December 31, 2004, 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 December 17, 2004, with 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.

This report documents one finding concerning an unknown obstruction of the containment equipment hatch that could not be rapidly removed to ensure expeditious hatch closure would it have been necessary to do so during the recently completed refueling outage. This finding has potential safety significance greater than very low significance. This finding did not present an immediate safety concern at the time it was discovered due to the availability of core cooling. The hatch obstruction was removed within 8 hours of discovery. The finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for the inspection finding at this time. Please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review. We will provide you with the results of our preliminary significance determination for this finding under separate correspondence.

In addition, this report documents six NRC-identified findings and one self-revealed finding, all of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the violations were entered in your corrective program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. In addition, two licensee identified violations are listed in Section 4OA7 of this report.

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, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector Office at the Kewaunee facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (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/

Thomas Kozak, Team Leader
Technical Support Section
Division of Reactor Projects

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

Enclosure: Inspection Report 05000305/2004009
w/Attachment: Supplemental Information

cc w/encl: J. Cowan, Executive Vice President,
Chief Nuclear Officer
K. Davison, Plant Manager
Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel & Secretary
D. Molzahn, Nuclear Asset Manager,
Wisconsin Public Service Corporation
L. Weyers, Chairman, President and CEO,
Wisconsin Public Service Corporation
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin

DOCUMENT NAME: E:\Filenet\ML050460220.wpd

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII	RIII		
NAME	RNg/trn*TKozak for	TKozak		
DATE	02/14/05	02/14/05		

OFFICIAL RECORD COPY

ADAMS Distribution:
WDR
CFL
RidsNrrDipmIipb
GEG
HBC
KGO
JFL
CAA1
C. Pederson, DRS (hard copy - IR's only)
DRPIII
DRSIII
PLB1
JRK1
ROPreports@nrc.gov (inspection reports, final SDP letters, any letter with an IR number)

U.S. NUCLEAR REGULATORY COMMISSION
REGION III

Docket No.:	50-305
License No.:	DPR-43
Report No.:	05000305/2004009
Licensee:	Nuclear Management Company, LLC
Facility:	Kewaunee Nuclear Power Plant
Location:	N 490 Highway 42 Kewaunee, WI 54216
Dates:	October 1 through December 31, 2004
Inspectors:	R. Krsek, Senior Resident Inspector D. Jackson, Senior Resident Inspector (Acting) P. Higgins, Resident Inspector R. Morris, Resident Inspector, Point Beach Nuclear Power Plant R. Alexander, Radiation Specialist J. Cameron, Project Engineer M. Holmberg, Senior Reactor Engineer J. Jandovitz, Reactor Engineer R. Langstaff, Senior Engineering Inspector M. Mitchell, Radiation Specialist J. Neurauter, Reactor Engineer R. Ng, Reactor Engineer C. Roque-Cruz, Reactor Engineer W. Slawinski, Senior Radiation Specialist R. Walton, Operations Engineer
Observer:	S. Bakhsh, Reactor Engineer
Approved By:	T. Kozak, Team Leader Technical Support Section Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000305/2004009; 10/01/2004 - 12/31/2004; Kewaunee Nuclear Power Plant; Fire Protection, Refueling and Outage Activities, Identification and Resolution of Problems, Event Follow-up, Other Activities and Cross-Cutting Areas.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections of licensed operator requalification, inservice inspection, reactor pressure vessel head replacement, emergency preparedness and the radiation protection program. The inspections were conducted by the resident inspectors and Region III inspectors. Seven Green Non-Cited Violations (NCVs) were identified. In addition, one apparent violation with potential safety significance greater than Green, and two unresolved items were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was identified by the inspectors for a violation of a fire protection License Condition. The inspectors identified multiple examples of combustible materials either stored or in use without specific authorization. Specifically, the licensee stored and used lubricating oil in an emergency diesel generator room beyond that authorized by the Fire Protection Program Analysis, the licensee stored unauthorized combustible materials above the shelves in the working materials storage area and on top of cabinets nearby, and the licensee stored compressed flammable gas cylinders in the auxiliary building without authorization. Once these issues were identified, the licensee removed the unauthorized materials. This finding was related to the cross-cutting area of problem identification and resolution in that the NRC had previously identified issues relating to control of transient combustible materials above and near the working materials storage area but adequate corrective actions were not put in place to prevent recurrence of this issue.

  The finding was more than minor because the failure to adequately control combustible materials, if left uncorrected, could become a more safety significant concern. The finding was of very low safety significance because the issue was a low degradation of fire prevention and administrative controls. The finding was a Non-Cited Violation of License Condition 2.C(3) which required specific authorization for the storage and use of combustibles in safety-related areas. (Section 1R05.1.b.1)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of a fire protection License Condition. The inspectors identified the storage of compressed oxygen cylinders near compressed flammable gas cylinders. Once this issue was identified, the licensee removed the stored compressed oxygen cylinders from the area.

  The finding was more than minor because the inappropriate storage of compressed oxygen cylinders could result in greater severity of a fire affecting equipment important to safety. The finding was of very low safety significance because the issue was a low degradation of fire prevention and administrative controls. The finding was a Non-Cited Violation of License Condition 2.C(3) which required the bulk storage of compressed oxygen cylinders to be separated from compressed flammable gas cylinders and corrective action of conditions significantly adverse to quality to preclude recurrence. (Section 1R05.1.b.2)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for a violation of a fire protection License Condition. The inspectors identified that the licensee failed to identify pertinent information, such as the presence of compressed flammable gas cylinders, on a fire area strategy for fire brigade personnel. Once this issue was identified, the licensee revised the fire area strategy for the affected area.

  The finding was more than minor because the failure to provide adequate warnings and guidance relating to hazards associated with compressed flammable gas cylinders in fire strategies could adversely impact fire fighting strategies used by the fire brigade in fighting a fire. The finding was of very low safety significance due to extensive training provided to fire brigade members to deal with unexpected contingencies. The finding was a Non-Cited Violation of License Condition 2.C(3) which required that fire area strategies provide pertinent information to help the fire brigade to be better prepared for fire fighting within that area. (Section 4AO2.3.b)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The finding was associated with the licensee's failure to adequately implement scaffold control requirements contained in Procedure GMP-127, "Requirements and Guidelines for Scaffold Construction and Inspection," which required that scaffolding be no closer than 2 inches from any safety-related equipment unless otherwise evaluated and approved by Engineering. Specifically, scaffolding was erected within 2 inches of safety-related piping for the Service Water outlet from the jacket water heat exchangers for Diesel Generator B, the piping for the Emergency Borate MOV (CVC-440), and Safety Injection Pump A, without engineering evaluation and approval. Upon discovery of this condition, the licensee took immediate action to bring all noted scaffolding problems into compliance with licensee procedures and initiated a CAP document for the issue.

The finding was more than minor because, if left uncorrected, the issue may have resulted in a more significant safety concern. Specifically, the failure of scaffolding having adequate spacing in the vicinity of safety-related equipment during a seismic event could result in damage to mitigating equipment. The finding was of very low safety significance because it did not result in the actual loss of the safety function of the train or system. The finding was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." (Section 1R20.1.b.1)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." The original licensing and design basis of the containment sump screens was to prevent any particles greater than 1/8 inch from entering the sump. The inspectors determined that the screen size was 1/8-inch by 15/32-inch which allowed particles greater than 1/8-inch to enter the sump. The inspectors subsequently determined that this issue had been identified and entered into the licensee's corrective action program in 1997. However, adequate corrective actions were not taken to correct this condition adverse to quality. Once this issue was identified, the licensee conducted an operability determination and concluded that there were no immediate operability issues with the containment sump. The licensee determined that the sump screens were nonconforming in accordance with Generic Letter 91-18, and planned long term corrective actions to be developed in conjunction with the resolution of Generic Safety Issue 191 and NRC Generic Letter 2004-02. The inspectors concluded that the primary cause of this finding was related to the performance characteristic of corrective actions in the cross-cutting area of problem identification and resolution.

  This finding was more than minor because the issue affected the Mitigating System cornerstone attribute of design control for initial design and equipment performance reliability and affected the associated cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because it was not a design or qualification deficiency that has been confirmed to result in a loss of function per Generic Letter 91-18. This finding was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." (Section 4OA5.2.c.1)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, And Drawings," regarding licensee instructions and procedures for containment sump inspections. Specifically, the inspectors identified that current licensee procedures did not require inspection or cleaning when boric acid or small debris may be present in the containment sump. The licensee's procedures for containment coatings did not require inspection of the coating located inside the containment sump which had not been inspected since initial application; and the licensee's procedure for containment sump gap inspections did not specify acceptance criteria to ensure this activity was satisfactorily accomplished. The licensee subsequently initiated several corrective actions to address these issues which included, but are not limited to: immediate inspection and cleaning of the safety-related containment sump; immediate inspection and assessment of the safety-related sump concrete coating; revision of preventive maintenance activities to require inspection and cleaning of the safety-related containment sump every refueling outage; revision of procedures to include inspection of the safety-related containment sump concrete coating every refueling outage; and revision of procedures to include appropriate acceptance criteria for determining that important activities were satisfactorily accomplished.

This finding was more than minor because if left uncorrected the finding could become a more significant safety concern and the issue affected the Mitigating System cornerstone attributes of equipment performance reliability and procedure quality and affected the associated cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance because it was not a design or qualification deficiency that has been confirmed to result in a loss of function per Generic Letter 91-18. This finding was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." (Section 4OA5.2.c.2)

Cornerstone: Barrier Integrity

- TBD. The inspectors identified an apparent violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, And Drawings," having potential safety significance greater than green. The finding was associated with the licensee's inability to close the containment equipment hatch in an expeditious manner while the plant was in the refueling shutdown mode, fuel was in the reactor vessel, the time to boil was estimated to be less than 30 minutes, and the reactor coolant system was open to the containment atmosphere. The inability to close the containment equipment hatch was caused by a design error in a large steel rail system installed inside the containment which was to be used to bring heavy equipment into the containment. This large steel rail system obstructed closure of the containment equipment hatch. The inability to close the hatch in an expeditious manner violated the licensee's procedure requirements to do so.

  This finding was more than minor because it affected the Barrier Integrity Cornerstone objective and was associated with the Barrier Integrity Cornerstone attribute of containment boundary preservation. Since this finding was determined to be potentially greater than Green using the SDP Phase 2 Process, this finding is of a to-be-determined (TBD) safety significance pending review by the NRC Significance Determination Process/Enforcement Review Panel (SERP). (Section 1R20.b.2)

- Green. A finding of very low safety significance associated with Technical Specification 3.8 a.1.b., "Refueling Operations - Containment Closure," was self-revealed during required daily surveillance testing of reactor building ventilation system isolation. During the surveillance test, plant operators discovered that radiation monitors would not cause a Reactor Building Ventilation System Isolation to occur as designed. The cause of this failure was that other engineered safeguards testing was in progress that disabled the Reactor Building Ventilation System Isolation function, which was required to be operable at the time. Once this issue was identified, the licensee promptly restored the automatic containment ventilation isolation capability, initiated procedure changes to prevent this issue from recurring, and entered the issue into the corrective action program .

This finding was more than minor, because it represented a degradation of the Barrier Integrity Cornerstone objective and was associated with Barrier Integrity Cornerstone attribute of safety system and component and barrier performance (containment isolation). The finding was of very low safety significance because it did not result in the actual release of radioactive material. This finding was a Non-Cited Violation of Plant Technical Specification 3.8.a.1.b., "Refueling Operations-Containment Closure." (Section 1R20.b.3)

B. Licensee-Identified Violations

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. The violations and the licensee's corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant operated at or near 100 percent power until operators shut it down for refueling outage R27 on October 9, 2004. The licensee completed the outage and returned the plant to operation on December 4, 2004. The plant remained at or near 100 percent power for the remainder of the assessment period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the facility's design and procedures for cold weather protection, completing one inspection procedure sample. 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. Additionally, the inspectors walked down selected plant areas to ensure that operator actions maintained the readiness of essential systems and that accessibility of controls, indications, and equipment would be maintained during these cold weather conditions. The inspectors also examined the history of issues raised in the area of severe cold weather and assessed the licensee's corrective actions.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

Reactor Vessel Closure Head (RVCH) Replacement (71007)

a. Inspection Scope

From October 18, 2004, through October 22, 2004, and November 30, 2004, through December 3, 2004, the inspectors reviewed the licensee's evaluations of applicability determination and screening questions for the design changes associated with the RVCH replacement to determine, for each change, whether the requirements of 10 CFR 50.59 had been appropriately applied. Specifically, the inspectors reviewed design change request No. 3481, which included a review of the function of each changed component, the change description and scope, and the 10 CFR 50.59 screening evaluation for the following eight samples:

- RVCH replacement;
- RVCH insulation inside cooling shroud replacement;
- control rod drive mechanism (CRDM) pressure housing assembly replacements;
- removal of four unused part length CRDMs;
- modification of four capped latch housing penetrations;
- removal of three Conoseal flanges;
- addition of three core exit thermocouple nozzle assembly flanges; and
- relocation of the RVCH vent and separation from the reactor vessel level indication system.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," to determine acceptability of the completed pre-screenings and screening. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments." The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04Q)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems, completing two inspection procedure samples:

C Service Water (SW) Trains 'A' & 'B' inside containment from containment penetration to all Containment Fan Cooling Units; and
C Auxiliary Feedwater in the Turbine Building

The inspectors conducted partial walkdowns of the systems listed to verify that the systems were correctly aligned to perform their design safety function. In preparation for the walkdowns, 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 also examined valve positions and electrical power availability to verify that valve and electrical breaker positions were consistent with, and in accordance with, the licensee's procedures and design documentation. The inspectors also observed the material condition of the equipment.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (71111.04S)

a. Inspection Scope

The inspectors performed a complete walkdown of the Safety Injection (SI) System to verify that the system was correctly aligned to perform its design safety function, completing one inspection procedure sample. 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 walkdown, the inspectors also examined valve positions, electrical power availability, and Control Room control switch positions to verify that valve and electrical breaker positions were consistent with, and in accordance with, the licensee's procedures and design documentation. The inspectors also observed the material condition of the equipment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection Quarterly Walkdown (71111.05Q)

a. Inspection Scope

The inspectors performed fire protection walkdowns of the following twelve plant areas, completing twelve inspection procedure samples:

- AX-23A, Refueling Water Storage Tank Area;
- AX-23B, Reactor Auxiliaries North Center;
- AX-24, Fuel Handling Rooms;
- AX-32, Service Rooms;
- RC-60, Reactor Containment Vessel;
- SC-70A, Screenhouse North;
- SC-70B, Screenhouse South;
- TU-90, Diesel Generator 1-A;
- TU-92, Diesel Generator 1-B;
- TU-95A, Dedicated Shutdown Panel Room;
- TU-95B, Safeguards Alley; and
- TU-95C, Auxiliary Feedwater Pump 1A Room.

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 materiel condition 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 that could initiate a plant transient. The inspectors verified that fire response equipment was in the designated location and available for immediate use without obstruction; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and that passive features such as 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 (CAP).

b. Findings

b.1 <u>Inadequate Control of Combustible Materials</u>

<u>Introduction:</u>

The inspectors identified a Non-Cited Violation (NCV) of License Condition fire protection requirements having very low safety significance (Green) for several examples of storing and using combustible materials without specific authorization.

<u>Description:</u>

Unauthorized Storage and Use of Lubricating Oil

On October 18, 2004, the inspectors observed maintenance personnel changing out the lubricating oil for the 1A emergency diesel generator (EDG). At the time of the inspectors’ observations, the used lubricating oil had been removed from the diesel generator and transferred out of the room, Fire Zone TU-90. Maintenance personnel had staged nine 55-gallon drums in the room and were in the process of transferring new oil to the diesel generator.

The inspectors noted that the 4-kiloVolt (kV) switchgear for the 1-5 electrical bus for the "A" train of safety-related equipment was located in the same fire zone adjacent to the 1A EDG. At the time of the inspectors’ observations, the reactor was shutdown and in refueling operations. Fuel was in the process of being removed from the reactor and being transferred to the spent fuel pool (SFP). The licensee had identified the ‘B’ train as the "protected" train. However, due to the amount of decay heat required to be removed from the SFP at that time, both the 1A and 1B SFP cooling pumps were required to be in service. The 1A SFP pump was powered from the 1-52 480-Volt electrical bus which, in turn, was powered from of the 1-5 4-kV electrical bus which originated in Fire Zone TU-90.

The inspectors reviewed the Fire Protection Program Analysis fire zone summary for Fire Zone TU-90 and noted that the zone was identified as having 303 gallons of lubricating oil located in the diesel generator as part of the combustibles for the room. In addition, the Fire Protection Program Analysis specified that the lubricating oil was in the EDG. However, the quantity of oil in the fire zone at the time of the inspectors’ observations was approximately 495 gallons of lubricating oil, i.e., nine 55-gallon drums. In addition, the majority of lubricating oil was being stored outside of the EDG.

Based on the discussions with site fire protection personnel, the inspectors determined that maintenance personnel had initially requested a transient combustible permit to bring in new lubricating oil while removing the used oil. The fire protection personnel denied their request and directed the maintenance personnel to first remove all of the used oil before bringing new oil into the fire zone. Maintenance personnel had estimated that nine 55-gallon drums of lubricating oil would be necessary based on their review of a technical manual for the EDG. The technical manual indicated that approximately 489 gallons of lubricating oil would be necessary for a diesel generator having an increased capacity oil pan. However, the maintenance personnel did not recognize that the Kewaunee diesel generator had the basic oil pan, which required less oil, in lieu of the optional increased capacity pan.

Unauthorized Transient Combustibles in Fire Zone AX-32:

The inspectors identified two examples where transient combustibles were not being adequately controlled within Fire Zone AX-32. The examples were:

- On October 19, 2004, the inspectors identified that materials, consisting of two cardboard boxes and a plastic bucket, were stacked on top of shelving in the working materials storage area. The materials were high enough such that a fire in the materials would not be detected by the detectors for the automatic deluge system. In addition, there was a potential that the materials would not be extinguished by deluge system due to their location. The inspectors noted that there were cables important to safety located approximately 7 feet above the materials.

- On October 20, 2004, the inspectors identified that materials were stacked on top of a metal cabinet in a hallway on the north side of the partial height wall for the materials storage area. The materials consisted of two cardboard boxes labeled as containing paper towels and a third cardboard box labeled as containing reinforced wipes. As the materials were located outside of the partial height wall for the materials storage area, a fire in the materials would neither be detected by the detectors for the materials storage area automatic deluge system nor suppressed by the materials storage area automatic deluge system. The inspectors noted that cables important to safety were located approximately 6 feet above the materials.

Unauthorized Storage of Hydrogen in Auxiliary Building

On December 1, 2004, the inspectors identified that a compressed gas cylinder containing a flammable mixture of hydrogen (8.92 percent concentration) and nitrogen was stored on the 586 foot elevation of the auxiliary building near door 196 in Fire Zone 23B. The inspectors noted that procedure FPP-08-08, "FP - Control of Transient Combustible Materials," specified designated areas for the bulk storage of large (i.e., greater than one pound in size) compressed flammable gas cylinders and prohibited storage in other locations. The inspectors noted that the 586 foot elevation of the auxiliary building was not among the designated areas and, as such, concluded that the compressed gas cylinder was not authorized to be located there. The inspectors noted that there were a number of overhead cable trays near the compressed gas cylinder. One of the cable trays was designated as a safety related cable tray.

Analysis:

The inspectors determined that the specific example of bringing more lubricating oil into Fire Zone TU-95 than what was permitted by the Fire Protection Program Analysis was a performance deficiency. This specific performance deficiency was determined to be more than minor because it affected the initiating events cornerstone attribute of protection against external factors (fire) in that the amount of lubricating oil exceeded the Fire Protection Program Analysis limit.

The inspectors determined that the specific examples of storing materials above the top shelves in and on top of cabinets near the working materials storage working area of Fire Zone AX-32 was a performance deficiency. The inspectors concluded that the specific examples identified would not affect a initiating event cornerstone because there was not enough material to develop a sufficiently large fire which would affect the cables important to safety located directly above. However, due to the multiple examples identified, the failure to adequately control storage and use of combustibles could become a more significant safety concern if left uncorrected and, as such, this performance deficiency is considered more than minor.

The inspectors determined that the specific example of storing a flammable gas cylinder in the auxiliary building in a non-authorized location was a performance deficiency. This specific performance deficiency was determined to be greater than minor because it affected the initiating events cornerstone attribute of protection against external factors (fire) in that a fire involving the compressed flammable gas cylinder could affect cables important to safety.

The inspectors determined that, in general, failing to adequately control storage and use of combustibles, as evidenced by multiple examples, was a performance deficiency. The inspectors concluded that this performance deficiency could lead to a more significant safety concern if left uncorrected. As such, the inspectors determined that the finding associated with this performance deficiency was more than minor.

In accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated, September 10, 2004, the inspectors performed a Significance Determination Process (SDP) Phase 1 screening for the specific examples and determined that the finding was a fire initiator contributor, i.e., an external event initiator. The inspectors performed a Phase 1 screening in accordance with IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated May 28, 2004, and determined that the finding affected the fire prevention and administrative controls category. Using Attachment 2, "Degradation Rating Guidance Specific to Various Fire Protection Program Elements," the inspectors determined that the specific examples identified represented low degradations. Specifically, the lubricating oil was not a low flashpoint combustible liquid. A fire involving the observed materials in the materials storage working area would not affect cables important to safety. Although the flammable gas cylinder stored in the auxiliary building was comparable to low flashpoint combustible liquids, the gas cylinder was an approved container. As such, under Task 1.3.1, question 1, of IMC 0609, Appendix F, the inspectors determined that the finding screened to Green and no further analysis was required.

The NRC had identified similar issues on July 12, 2004, and July 28, 2004, (documented in Inspection Report 05000305/2004005). At that time, the licensee initiated CAP 021822 and CAP 022025. However, the licensee’s corrective actions (CAs) were insufficient to preclude recurrence. Based on the identification of multiple examples, the inspectors concluded that the licensee's control of transient combustibles continued to be inadequate and previous CAs were ineffective. This finding was related to the cross-cutting area of problem identification and resolution in that the NRC had previously identified issues relating to control of transient combustible materials above and near the working materials storage area but adequate corrective actions were not put in place to prevent recurrence of this issue.

Enforcement:

Kewaunee License Condition 2.C(3), required, in part, that the Nuclear Management Company (NMC) implement and maintain in effect all provisions of the approved fire protection program as described in the Kewaunee Nuclear Power Plant (KNPP) Fire Plan, and as referenced in the Updated Safety Analysis Report (USAR), and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplemented dated February 13, 1981). Section 8.3 of the KNPP Fire Protection Program Plan specified that specific authorization was required for the storage and use of combustibles in safety-related areas. Fire zones TU-90, AX-32, and AX-23B were safety-related areas. Contrary to the above, the inspectors identified the following three examples of the failure to comply with License Condition 2.C(3):

On October 18, 2004, approximately 495 gallons of lubricating oil was either being stored or in use in the fire area TU-90. The majority of the lubricating oil was either being stored or in use outside of the EDG. The quantity of 495 gallons was in excess of the 303 gallons of lubricating oil specifically authorized by the Fire Protection Program Analysis to be in fire zone TU-90. In addition, the lubricating oil was not specifically authorized by the Fire Protection Program Analysis to be outside of the EDG. The licensee initiated CAP 023388 and CAP 023428 to address the issue, completed changing oil out the lubricating oil for the EDG, and removed the lubricating oil which was outside of the EDG. The licensee initiated CAP 023388 and CAP 023428 to address the issue. The licensee planned to revise the maintenance instructions for the diesel generator to indicate the amount of the lubricating oil required.

On October 19, 2004, the inspectors identified that materials were stacked on top of shelves above the working materials storage area within fire zone AX-32. In addition, on October 20, 2004, the inspectors identified materials stacked on top of a cabinet adjacent to the working materials storage area within fire zone AX-32. None of these identified transient combustible materials were specifically authorized. After these uncontrolled transient combustible materials were identified, the licensee entered the issues into their CAP (under CAP 023418 and CAP 023478) and removed the materials. The licensee also installed signs to inform people to not place materials on top of shelves in the working materials storage area or on top of the cabinets in the hallway outside the materials storage area.

On December 1, 2004, the inspectors identified that a compressed flammable gas cylinder was stored in the auxiliary building, a safety-related area, without specific authorization. The licensee initiated CAP 024553 to address this issue and removed the compressed flammable gas cylinder being stored there.

Because the three examples of this violation were of very low safety significance and were entered into the licensee's CAP, it is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000305/2004009-01)

b.2 Storage of Oxygen Cylinders Next to Flammable Gas Cylinders

Introduction:

The inspectors identified a NCV of License Condition fire protection requirements having very low safety significance (Green) for the storage of compressed oxygen cylinders next to compressed flammable gas cylinders.

Description:

On October 21, 2004, the inspectors identified two compressed oxygen gas cylinders stored along with compressed flammable gas cylinders on the 586 foot elevation of the auxiliary building in fire zone AX-23B near doors 196 and 255. The compressed oxygen cylinders were within several feet of compressed flammable gas cylinders. The compressed flammable gas cylinders consisted of four cylinders with hydrogen and nitrogen mixtures, three cylinders with hydrogen and argon mixtures, and one propane cylinder. The cylinders were unattended and were not separated by a barrier. The inspectors noted that Fire Zone AX-23B was a safety-related area and that there was an abundance of cable trays above where the compressed gas cylinders were stored. At least one of the cable trays contained safety-related cables. Section 5.2.3.2 of procedure FPP-08-08, "FP - Control of Transient Combustible Materials," specified that the bulk storage of compressed oxygen cylinders shall be separated from compressed flammable gas cylinders by a minimum of 20 feet or a noncombustible barrier at least 5 feet high having a fire resistance rating of at least ½ hour.

Analysis:

The inspectors determined that failing to follow procedures for the storage of compressed oxygen cylinders near compressed flammable gas cylinders was a performance deficiency. This performance deficiency was determined to be greater than minor because it affected the mitigating systems cornerstone attribute of protection against external factors (fire). Specifically, the inappropriate storage of compressed oxygen cylinders could result in the greater likelihood or severity of a fire which affects equipment important to safety. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated September 10, 2004, the inspectors performed a SDP Phase 1 screening and determined that the finding was a fire initiator contributor, i.e., an external event initiator. The inspectors performed a Phase 1 screening in accordance with IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated May 28, 2004, and determined that the finding affected the fire prevention and administrative controls category. Using Attachment 2, "Degradation Rating Guidance Specific to Various Fire Protection Program Elements," the inspectors determined that the finding represented a low degradation. Under Task 1.3.1, question 1, of IMC 0609, Appendix F, the inspectors determined that the finding screened to Green and no further analysis was required.

Enforcement:

KNPP License Condition 2.C(3), required, in part, that the NMC implement and maintain in effect all provisions of the approved fire protection program as described in the KNPP Fire Plan, and as referenced in the USAR, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplemented dated February 13, 1981). Section 8.3 of the KNPP Fire Protection Program Plan specified, in part, that administrative procedures be in place to review, and limit if necessary, the storage and use of combustibles during all modes of plant operation. Procedure FPP-08-08 was the administrative procedure in place to review and limit, if necessary, the storage and use of combustibles during all modes of plant operation. Section 5.2.3.2 of Procedure FPP-08-08 specified, in part, that the bulk storage of compressed oxygen cylinders shall be separated from compressed flammable gas cylinders by a minimum of 20 feet or a noncombustible barrier. Contrary to the above, the inspectors identified that two compressed oxygen cylinders were stored within 20 feet of compressed flammable gas cylinders with no intervening barrier. Once this issue was identified during this inspection, the licensee entered the issue into their CAP under CAP 023480 and CAP 023483, removed the oxygen cylinders stored in the area, and added signs stating that oxygen cylinders should not be stored in the area. Because this violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000305/2004009-02)

.2 Fire Protection Annual Fire Drill Observation (71111.05A)

a. Inspection Scope

The inspectors observed and evaluated the effectiveness of the fire brigade response to a simulated fire in the plant. This inspection constituted one inspection procedure sample. The inspectors verified that protective clothing was properly donned and was in good condition, and that Self Contained Breathing Apparatus equipment was properly utilized. In addition, the inspectors verified that the fire pre-plan strategy was utilized and that all fire fighting equipment was in good condition and properly utilized. Radio communications were effective between all stations involved in the drill. The inspectors observed the actions of the fire brigade leader, and the manner in which he implemented the fire pre-plan and directed his fire brigade to extinguish the simulated fire. The fire drill plan was thorough, contained evaluation criteria, and was followed appropriately by fire drill coordinators.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Review of External Flood Protection Measures

a. Inspection Scope

The inspectors performed an external flood protection inspection for the lake screen house. This constituted one inspection procedure sample. The inspectors reviewed USAR and related external flooding analysis to identify external flooding barriers and vulnerabilities. The inspectors reviewed plant procedures and performed plant walkdowns to determine the adequacy and conditions of existing flood protection measures. The inspectors also examined the history of issues raised in the area of flood protection and assessed the licensee's CAs.

b. Findings

No findings of significance were identified

.2 Review of Internal Flood Protection Measures

a. Inspection Scope

The inspectors walked down and reviewed piping configurations in the following internal flood zones, constituting one inspection procedure sample. Comparisons with the assumptions made in the plant internal flood analysis were also made.

- Zone 2B EDG 1A Room;
- Zone 5B 480V Switchgear Buses 1-51 and 1-52;
- Zone 5B-2 1A AFW Pump Room;
- Zone 3B EDG 1B room;
- Zone 5B-1 480V Switchgear Buses 1-61 and 1-62; and
- Zone 5B-3 1B AFW Pump Room.

The inspectors evaluated internal flooding hazards in these areas and evaluated the flood protection features, such as area doors, door gaps, and room drains to determine whether the flood protection features were in satisfactory physical condition, unobstructed, and capable of providing adequate flood protection. The inspectors also reviewed design basis documents and risk analyses to determine plant vulnerabilities and protective features for the areas inspected.

b. Findings

Introduction:

The inspectors identified an Unresolved Item (URI) associated with the potential vulnerability of safety-related equipment to flooding in the Turbine Building Basement. The failure of non safety-related equipment in the Turbine Building could impact the ability of safety-related equipment in the areas to perform their intended safety function. The areas inspected were located immediately adjacent to the Turbine Building Basement.

Discussion:

On September 14, 2004, the inspectors reviewed internal flood protection measures for the AFW pump rooms, the 480-V Safeguards bus area, the safe shutdown panel area, and the EDG 1-A and 1-B rooms, which also contained safeguards Buses 1-5 and 1-6, respectively. These areas were located immediately adjacent to the Turbine Building Basement. The inspectors identified a previous entry in the licensee's CAP (CAP 016375, dated May 10, 2003) regarding a flooding event which occurred on May 9, 2003, due to a trench overflowing in the area containing the AFW pumps, the 480-V safeguards bus area, and the safe shutdown panel area, also referred to as the "safeguards alley." This trench received discharge flow from all AFW pump lube oil coolers. At the time of the event, both AFW Pump 'A' and 'B' were running with lube oil coolers discharging directly to this trench. Apparent Cause Evaluation (ACE) 002299, written for CAP 016375, stated that at the time of this flooding event, the flocculator in the basement of the turbine building overflowed and that a significant amount of water was dumped to the turbine building sump" located in the basement of the turbine building and, as a result, water no longer flowed to the sump and backed up in safeguards alley.

A review of design drawings by the inspectors revealed a direct piping connection from the turbine building sump to the trench in safeguards alley. The inspectors determined that there were no check valves located in the piping to prevent water spills in the turbine building basement from backing up into the safeguards alley. The inspectors also noted that no flood barriers specifically designed to protect equipment in the safeguards alley from flooding in the turbine building basement were installed.

The inspectors requested additional information from the licensee regarding potential flooding events occurring in the safeguards alley. The licensee documented its response to the inspectors' information request in Condition Evaluation (CE) 014653. This CE stated that it would take approximately 3 hours for flooding caused by AFW pump discharge to affect safety-related equipment, and such flooding could be mitigated by opening doors between the safeguards alley and the turbine building basement. The CE also stated that other sources of flooding in the turbine building basement need not be considered since such flooding events are outside the design basis of the plant.

During a review of the licensee's design basis documents, the inspectors identified that the equipment in the safeguards alley is clearly designated with a Nuclear Safety Design Classification of Class I, in Appendix B to the licensees' USAR, Table B.2-1. In addition, Section B.5 describes how Class I items are protected against damage, as follows: "The Class I items are protected against damage from: (a) Rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the class one function is impaired." Finally, in a letter dated September 26, 1972, the Atomic Energy Commission requested the licensee to provide information on conditions such as flooding that might potentially adversely affect the performance of safety-related equipment. In a letter dated October 31, 1972, the licensee responded, in part, as follows: "It has been determined that consequences of failure of non-category I (seismic) systems could potentially adversely affect the performance of engineered safety systems. Specifically, the non-category I (seismic) items are the fire protection lines in the turbine building basement and the reactor makeup water and demineralized waterline in the auxiliary building basement. However, because of safety equipment redundancy and design arrangement, the functional purpose of the safety equipment would not be jeopardized in the event of failure of any of these lines." Notwithstanding the licensee's assertions, the inspectors identified additional non-category I systems and components in the turbine building basement, including the condenser and condenser boot seals, which could, should they fail, potentially impact the safety-related equipment in the safeguards alley and adjacent rooms.

During plant startup at the conclusion of the 2004 refueling outage, the inspectors asked the licensee to provide information on the operability of the AFW pumps, considering the additional potential sources of flooding in the turbine building basement and the impact of such flooding on the safety-related equipment in the safeguards alley, including the AFW pumps. The licensee responded with a position paper that essentially stated that the consideration of flooding events in the turbine building basement and their potential impact on safety-related equipment in safeguards alley were not within the plant's licensing basis and were therefore not an operability or reportability concern. The licensee position paper also stated that the condition should be reviewed and compensatory actions and/or modifications should be implemented that address this concern.

Pending additional licensee evaluation and inspector review of the potential impact of flooding in the Turbine Building Basement on safety-related equipment located in adjacent rooms, this issue will be treated as an URI (URI 05000305/2004009-03).

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors performed an inspection of the heat exchanger performance on the 1A EDG cooling water heat exchangers, completing one inspection sample. The heat exchanger utilizes SW to cool the EDG during operation. The inspector observed heat exchanger performance data gathering and software calculation of the heat removal capability of the heat exchanger using obtained performance data, and inspected the disassembled heat exchanger for biofouling. The inspector reviewed test acceptance criteria and compared it against calculated test results. The inspector reviewed heat exchanger performance calculation methodology to ensure that both instrument uncertainty and calculation uncertainty were accounted for in the results to be compared against test acceptance criteria. The inspector also reviewed testing frequency to ensure that it was sufficient consistent with potential for biofouling.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08P)

.1 Reactor Coolant System Pressure Boundary Leakage Inspection

a. Inspection Scope

The inspectors visually inspected the under-vessel areas, following cold shutdown of the reactor at the beginning of the 2004 refueling outage. The reactor vessel insulation had been removed, which allowed inspection of the bare reactor vessel metal. The thimble guide tube penetration welds were examined for signs of boric acid deposition which would be indicative of reactor coolant leakage. The thimble guide tubes were inspected for any signs of boric acid streaking. The reactor vessel side walls were also examined for any signs of boric acid streaking. The floor under the reactor vessel was inspected for any signs of boric acid deposition. During the latter portion of the outage, following return to normal operating pressure and normal operating temperature, the inspectors accompanied the plant's ISI team on its performance of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1 System Pressure Test. This inspection included the reactor under vessel areas, the reactor head area, as well as numerous Class 1 piping systems. During this inspection, several reactor coolant system (RCS) leaks were identified and documented by the inspection team. One of the leaks involved a swagelock fitting in instrument Line 33 located about 4 feet above core exit thermocouple nozzle Assembly 35, which is located on top of the reactor head. This RCS leakage was considered by the licensee to be a high risk leak which could produce significant degradation, if it were to contact the reactor head. After investigating several alternatives to stop this leakage, the licensee decided to return the plant to a cold shutdown condition, in order to repair this and other system leakage identified during plant heat up inspections. During this cold shutdown condition, the licensee corrected and evaluated all system leakage identified. The inspectors reviewed licensee CAs on all leakages.

The inspectors also reviewed the documented licensee inspection results for the Class 2 Main Steam and AFW System Pressure Test. This documentation included identification of discrepant conditions found during inspection and CA taken.

b. Findings

No findings of significance were identified.

.2 Implementation of the Licensee's ISI Program

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries, based on a review of nondestructive examination (NDE) records and observations.

From October 12 through 22, 2004, the inspectors evaluated several activities involving NDE examinations with recordable indications, and welding. Specifically, the inspectors observed the following:

- Ultrasonic (UT) examination of two Safety Injection line welds (SI-W49 and SI-W51) inside containment; and
- Magnetic particle (MT) examination to Safety Injection pumps APSI-A and APSI-B in the auxiliary building.

The inspectors selected these components in order of risk priority as identified in Section 03 of IP 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the on-site inspection period. The inspectors evaluated these examinations for compliance with the ASME Boiler and Pressure Vessel Code Section XI and plant Technical Specification (TS) requirements and to determine whether indications and defects (if present) were dispositioned in accordance with the ASME Code.

The inspectors reviewed the licensee's records related to disposition of recordable indications identified in four examinations. Specifically, the inspectors reviewed the evaluation records with recordable indications accepted for continued service for:

- The reactor vessel closure head flange and control rod drive mechanism RV-W12;
- The steam generators SG-1A and SG-1B;
- The seal water injection filters AF SI-1A and AF SI-1B; and
- The RC-RTD line for reactor coolant loop B.

The inspectors evaluated the disposition of indications identified during these examinations for compliance with the requirements of the ASME Code Section XI.

The inspectors reviewed the licensee's records related to pressure boundary welding performed in the following components:

- 3-inch motor operated valves, PR1A/MV32089 and PR1B/MV32090;
- pressurizer power operated relief valve (PORV) block valve; and
- 3-inch check valve at the Auxiliary Feedwater Pump Discharge at Steam Generator 1B.

The inspectors performed this review to determine whether the welding acceptance and pre-service examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with the requirements of the ASME Code, Sections III, V, IX, and XI.

The above review constituted one inspection procedure sample.

From October 12, 2004, through October 22, 2004, the inspectors reviewed a sample of licensee activities related to the Boric Acid Corrosion Control program. This review included:

- direct observation of licensee staff performing a walkdown of systems inside containment, in part to identify evidence of boric acid leakage;
- review of two engineering evaluations performed for boric acid found on reactor coolant system piping and components;
- interviews with licensee staff involved in boric acid program; and
- review of corrective actions performed for evidence of boric acid leaks.

These observations and reviews were performed to confirm that:

- licensee visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components;
- degraded or non-conforming conditions are properly identified in the licensee's corrective action system;
- ASME Code wall thickness requirements were maintained; and
- corrective actions were consistent with requirements of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI.

The review discussed above constituted one inspection sample.

The activities that were not available for inspector's review for this inspection are identified in the table below.

<table>
<tr><th>Inspection Procedure 7111108 Section Number</th><th>Reason Activity was Unavailable For Inspection</th><th>Reduction in Inspection Procedure Samples</th></tr>
<tr><td>Section 02.02 Vessel Upper Head Penetration Inspection Activities.</td><td>The licensee did not perform vessel upper head inspection activities during this outage (Reactor Vessel Head Replacement).</td><td rowspan="2">The inspectors concluded that these unavailable activities constituted a reduction by two from the total number of procedure samples required by Section 71111.08-5 of Inspection Procedure 71111.08.</td></tr>
<tr><td>Section 02.04. Steam Generator (SG) Tube Inspection Activities.</td><td>The licensee did not perform SG tube inspection activities during this outage</td></tr>
</table>

b. <u>Findings</u>

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

.1 Observation of Licensed Operator Simulator Training

a. Inspection Scope

The inspectors observed licensee training personnel evaluate an operating crew during an accident scenario and subsequently observed the operating crew critique their performance. The inspectors observed the crew and verified the following attributes of crew performance: communications, alarm response, emergency operating procedure usage, component operations and emergency plan classifications. The inspectors reviewed the scenario for operational validity and risk significance. The inspectors discussed scenario observations and crew evaluations with the licensee trainers. In addition, the inspectors reviewed the licensees baseline fidelity study to ensure that differences between the simulator and actual control room board configuration were maintained as close as possible. This constitutes one quarterly inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

The inspectors reviewed the implementation of the Maintenance Rule for the Residual Heat Removal (RHR) system, completing one inspection sample. 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)(2) in accordance with 10 CFR 50.65. The inspectors reviewed a sample of maintenance work orders, action requests, functional failure evaluations, unavailability records, and a sample of condition reports (CRs) to verify that the licensee identified issues related to the Maintenance Rule at an appropriate threshold and that CAs were appropriate. Additionally, the inspectors reviewed the licensee’s performance criteria to verify that the criteria adequately monitored equipment performance. The inspectors discussed identified deficiencies with the licensee. The licensee documented these deficiencies on CRs.

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. Shutdown Safety Assessment Checklists and associated compensatory and component protection measures were inspected during the following 4 weeks of the licensee's refueling outage K27, constituting completion of four inspection samples:

- Week of October 11, 2004;
- Week of October 18, 2004;
- Week of October 25, 2004; and
- Week of November 1, 2004.

In particular, the inspectors evaluated the licensee's planning and management of maintenance and verified that shutdown risk was acceptable and monitored in accordance with the requirements of 10 CFR 50.65(a)(4), except as noted in Section 1R20.b.2 of this report. 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 shutdown 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)

.1 Increased Unidentified Leakage in Containment.

a. Inspection Scope

On September 28, 2004, the control room received a Containment Sump 'A' Level High Alarm. This alarm was not expected and on September 29, 2004, the licensee made a containment entry in an attempt to determine the source of the leakage. The resident inspector accompanied licensee personnel into the containment during this entry. The licensee was not able to identify the source of leakage into the sump. The licensee calculated the approximate leak rate and found it to be within acceptable limits. However, the licensee determined that this leak rate represented a significant increase in leakage into the containment sump over the leakage that had been routinely experienced during this operating cycle. The licensee determined by chemical analysis that the water leakage into the sump was from the service water system. On September 30, 2004, the sump high-level alarm was again received and the sump was pumped out.

On October 2, 2004, the sump high-level alarm was received and two additional containment entries were made by the licensee to inspect Containment Fan Coil Units 'A' and 'B'. The resident inspector accompanied licensee personnel on these two containment entries. No indications of leakage were noted. Additional containment entries were made by the licensee to inspect Containment Fan Coil Units 'C' and 'D'. Again, no indications of leakage were noted.

On October 3, 2004, the sump high-level alarm was again received and additional containment entries by the licensee determined that the leakage originated at the Shroud Cooling Units. Additional chemical analysis performed by the licensee on the water leaking into the sump confirmed that it was from the service water system and not reactor coolant. The Shroud Cooling Units were isolated in an attempt to stop the leakage into the containment sump. Following isolation, the leakage into the sump was reduced but it did not stop, indicating that the Shroud Cooling Units SW isolation valves were leaking. The Shroud Cooling Units are non safety-related components and are isolated on a SI signal. Licensee personnel determined that the leakage past the isolation valves was insignificant and would not affect the operability of the Containment Fan Coil Units. On October 9, 2004, the plant entered a refueling outage. Detailed inspection of the Shroud Cooling Units by the licensee confirmed that the leakage was coming from these units. Repairs to these units were made prior to plant startup at the end of refueling outage.

This review constituted one inspection sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability evaluations, completing two inspection samples:

- Control Room Exclusion Zone As Found Air Flows Not per Design Basis; and
- Turbine-Driven AFW Pump Outboard Bearing Oil Level Above Normal Band.

The inspectors reviewed design basis information, the Updated Final Safety Analysis Report, TS requirements, and licensee procedures to verify the technical adequacy of the operability evaluations. In addition, the inspectors verified that compensatory measures were implemented, as required. 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 workarounds did not create significant adverse consequences regarding the reliability, availability, and operation of accident mitigating systems, completing one inspection procedure sample of individual operator workarounds. The inspectors also assessed the effects of the workarounds on the ability to implement abnormal and emergency response procedures in a correct and timely manner. In addition, the inspectors reviewed any emergent risk significant operator workarounds to determine if the functional capability of a system or human reliability of an initiating event was affected.

Inspectors also assessed the cumulative affects of the current listing of operator workarounds for impacts on equipment reliability, availability, and a potential for equipment mis-operation. Impact of these workarounds were also assessed for negative affects on multiple mitigating systems, for the impact on operator actions required to respond to plant events and transients.   This constituted completion of one inspection procedure sample of the cumulative affects of operator workarounds.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

.1 Control Rod Guide Tube Split Pin Replacement

a. Inspection Scope

The inspectors reviewed the engineering analyses, design information and modification documentation for the replacement of the Control Rod Guide Tube Split Pins and the installation of a Fuel Assembly Sized Debris Cannister which occurred during the October 9, 2004, refueling outage. The Control Rod Guide Tube Split Pins restrained the lower end of the control rod guide tubes in the reactor vessel. The Fuel Assembly Sized Debris Canister was provided for the storage of radioactive split-pin-related debris in the Spent Fuel Pool. This inspection constituted one inspection sample. The inspection activities included, but were not limited to, verification and review of the following parameters associated with this modification: structural integrity, material compatibility, environmental qualification, safety classification, functional properties, seismic qualification, failure mode potentials, and the associated 10 CFR 50.59 screening analysis. Additionally, the inspectors observed portions of the installation and testing of the split pins, reviewed acceptance testing results, and reviewed CRs associated with the design change to verify that the licensee identified and documented problems at an appropriate threshold.

b. Findings

No findings of significance were identified.

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 samples:

- Containment Fan Cooling Units ‘A’, ‘B’, ‘C’ and ‘D’;
- EDG B Inspection and Operational Testing; and
- RHR Pump ‘B’ Overhaul Testing;

The inspectors verified that the testing was adequate for the scope of the maintenance work performed. The inspectors reviewed the acceptance criteria of the tests 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.

1R20 Refueling and Outage (71111.20)

a. Inspection Scope

The inspectors observed the licensee’s performance during refueling outage R27, which commenced on October 9, 2004, completing one inspection sample. The inspectors reviewed the Outage Plan Schedule prior to commencement of the outage to assess licensee response to periods of increased risk. The review included planned mitigating actions for periods of increased risk. The reactor shutdown was monitored including periods from initial power reduction to completion of plant cool down to cold shutdown. Specific attention was paid to reactivity control during plant power changes, the reactor shutdown, and boration in preparation for plant cool down. During plant cool down, rates of cool down were monitored to ensure that maximum rates were not exceeded. During the course of the outage, licensee activities were monitored to ensure that systems relied upon for reactor core cooling were maintained in appropriate configurations based on a valid assessment of risk for that point in the outage. Refueling activities were inspected to ensure that fuel handling was conducted in accordance with plant procedures and TSs. Finally, plant heat up and start up activities were inspected to ensure compliance with station procedures and TSs.

The inspectors performed the following observations on a frequent basis:

- Outage Management Outage Control Center turnover meetings to assess sensitivity of the licensee to periods of increased plant risk;

- Control Room panel walkdowns to inspect current configuration of systems required to remove reactor core decay heat;

- CAP issue screening meetings to observe sensitivity to issues identified that could potentially impact plant risk;

- In plant evolutions and on-going work to ensure that systems needed for reactor core cooling, and other required safety functions were being appropriately considered and required protected equipment was properly designated;

- Shutdown Safety Assessment Checklist Reviews to ensure levels of shutdown risk were as expected and the plant configuration matched periodically updated safety assessments; and

- During the extended outage period when the reactor core was fully offloaded to the SFP, the inspectors performed walkdowns at least weekly of the SW system supply to the SFP Cooling system, the SFP Cooling system, and the Normal and Electrical Power Supplies to SFP Cooling Pumps.

The inspectors performed the following specific inspection activities:

- A tag-out walkdown of the 'A' SW system, to ensure that all boundaries were appropriate for the plant and work conditions, all components were correctly positioned, and all safety tags were correctly placed;

- A tag-out walkdown of the 'B' EDG, to ensure that all boundaries were appropriate for the plant and work conditions, all components were correctly positioned, all safety tags were correctly placed, no tagged components negatively impacted the opposite train, and barriers were in place to protect the 'A' EDG;

- A tag-out walkdown of the SFP Cooling Filter and Pre-Filter, to ensure that all boundaries were appropriate for the plant and work conditions, all components were correctly positioned, and all safety tags were correctly placed;

- An inspection of the SFP Cooling System with the core fully offloaded to the SFP during a period of elevated risk with the 'A' SW System out-of-service. The SW System provided cooling to the SFP Heat Exchangers, and thus was the ultimate heat sink for the fuel offloaded to the SFP. The inspectors verified proper functioning and material condition of the SFP Cooling System, and protection of equipment and areas during the period of elevated risk;

- An inspection of electrical power supplies supporting SFP operation while the core was fully offloaded to the SFP. The electrical alignment was correct for the given plant conditions, and supported operation of both SFP Cooling pumps from independent power supplies. In addition, the SW System electrical alignment was proper such that cooling was being supplied to the SFP Heat Exchanger;

- An inspection of Root Cause Evaluation (RCE) 612 "Temporary Procedure Change Used To Inadvertently Bypass a Hold Card" was completed using Inspection Procedure (IP) 71152 "Identification and Resolution of Problems" to determine the adequacy of the threshold for the initiation of CAs and to determine the completeness of CAs associated with the evaluation. In this case, the licensee adequately addressed the root and contributing causes in the evaluation in their CAP. Resolution actions were completed in sufficient detail and in a timely fashion. Each item was reviewed by the licensee's Corrective Action Review Board (CARB) to determine if proposed and completed CAs would sufficiently resolve the issue. In addition, an effectiveness review was conducted for the completed CAs;

- An inspection of RCE 616 "Damaged Rod Control Cluster Assembly" was completed using IP 71152 "Identification and Resolution of Problems" to determine the adequacy of the threshold for the initiation of CAs and to determine the completeness of CAs associated with the evaluation. In this case, the licensee adequately addressed the root and contributing causes in the evaluation in their CAP. Resolution actions were completed in sufficient detail and in a timely fashion. Each item was reviewed by the licensee's CARB to determine if proposed and completed CAs will sufficiently resolve the issue. In addition, an effectiveness review was conducted for the completed CAs;

- A tag out walkdown of the Flux Map Electrical System was conducted to ensure proper protection against incore thimble movement was in place for the Under Vessel Penetration Inspection. This walkdown included confirmation that all boundaries were appropriate for the plant and work conditions, all components were correctly positioned, and all safety tags were correctly placed;

- An inspection of offsite and onsite electrical power supplies supporting RHR operation was conducted while the core was fully loaded. The electrical alignment was correct for the given plant conditions, and supported operation of both RHR pumps from independent power supplies. In addition, the Component Cooling Water electrical and mechanical system alignments were proper such that cooling was being supplied to the RHR heat exchangers;

- An inspection of decay heat removal systems was performed while the core was fully loaded. Designated decay heat removal systems, including both trains of the RHR System and both Steam Generators, were reviewed to ensure full availability and that these systems were monitored and protected as required by plant procedures and orders. The identified decay heat removal systems were correct for the given plant conditions and as designated in the Shutdown Safety Assessment. In addition, the control room monitoring of the decay heat removal capability and system performance was properly conducted;

- An inspection of the SFP Cooling System was conducted with the core fully offloaded to the SFP during a period of elevated risk with the 480-V power supplies to the SFP Pumps crosstied. This crosstie was established to allow work on the 4 Kv safeguards buses while still providing power to both SFP Pumps. The inspectors found the SFP Cooling System to be functioning properly, and in adequate physical condition. In addition, appropriate equipment and areas were protected by barriers during the period of elevated risk;

- An inspection of reactivity control practices was conducted with the core fully loaded. Potential boron dilution paths were identified, procedures for ensuring proper boron concentration and mixing were reviewed and control room practices for monitoring these parameters were observed. The inspectors found dilution paths properly identified, boron concentration in accordance with procedures and control room monitoring practices properly conducted;

- An inspection of the SW supply to the vital equipment was conducted with the core fully loaded. The inspectors walked down the valve, pump and heat exchanger line ups including the electrical supplies. The inspectors found the SW Cooling System to be functioning properly, and in adequate physical condition. In addition, appropriate equipment and areas were protected by barriers during the period of elevated risk to core cooling;

- An inspection of the plant's ability to close the containment equipment hatch in preparation for reactor vessel head lift was performed with the vessel fully loaded with spent fuel. A large steel rail system was installed inside the containment which was used to bring heavy equipment into the containment. The inspectors reviewed procedure and plans, including two specifically generated to ensure rapid removal of this rail system and walked down the containment hatch area. The inspectors found that the licensee was unable to close the containment equipment hatch in expeditious manner. (Section 1R20.1.b.2)

- The inspectors conducted a total of four containment walkdown inspections prior to containment closeout at the end of the refueling outage. The inspection included all levels interior to the containment, as well as the containment annulus area. The inspectors looked for any debris, equipment, tools or other items which should be removed prior to containment closeout. The inspectors also looked for any significant system leakage or other system discrepancies which would require correction prior to containment closeout. The emergency core cooling system sump basement level was inspected to ensure that any loose material which could affect sump post-accident effectiveness was identified and removed. The area under the reactor vessel as well as the upper refueling cavity area and the area on top of the head were inspected. The area between the inner containment hatch and the outer containment shield block hatch was inspected to ensure proper closure of both hatches. Any material, tools, or equipment which the licensee intended to leave in the containment during plant operation were inspected to ensure they were properly tied down. All discrepancies identified by the inspectors during these walkdowns were turned over to the licensee for disposition and were properly disposition by the licensee; and

- The inspectors conducted an inspection of plant start up activities to include portions of plant heat up, approach to criticality via dilution, and plant synchronization to the electrical grid. Start up evolutions were conducted in accordance with station approved procedures listed in the reference section.

b. Findings

.1 Scaffolding Erected Too Close to Safety-Related Equipment Required To be Operable

Introduction:

A finding of very low significance (Green) was identified by the inspectors for a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" having a very low safety significance. During walkdowns of areas where scaffolding had been erected to support outage activities, the inspectors identified several examples in which the licensee placed scaffolding closer than 2 inches from safety-related equipment without evaluation and approval by Engineering.

Description:

On October 6, 2004, with the plant at 100 percent power, the inspectors performed walkdowns of plant areas to review outage preparation activities associated with Refueling Outage R27. Scaffolding was being erected throughout the plant. Licensee procedure GMP-27, "Requirements and Guidelines for Scaffold Construction and Inspection," required that scaffolding not be erected within 2 inches of safety-related equipment, unless an engineering evaluation had been completed demonstrating that operability of the equipment had not been adversely impacted. The inspectors identified four areas in which scaffolding had been erected closer than 2 inches from safety-related equipment and an engineering evaluation had not been completed to ensure that equipment operability was not negatively impacted. The four areas included:

- B' EDG - A scaffold pic was in direct contact with the SW cooling outlet line from the cooling water heat exchangers. The cooling water heat exchangers were safety-related equipment. The licensee had not prepared an engineering evaluation to ensure that operability of the 'B' EDG, or its support systems were not negatively impacted;

- 'A' SI Pump - Multiple pieces of scaffold were in direct contact and within 2 inches of components and piping associated with the pump. The licensee had not prepared an engineering evaluation to ensure that operability of the 'A' SI pump, or its support systems were not negatively impacted;

- 'A' Internal Containment Spray (ICS) Pump - A scaffold pic was in direct contact with ICS piping in the north penetration area. The licensee had not prepared an engineering evaluation to ensure that operability of the 'A' ICS pump, or its support systems were not negatively impacted; and

- Emergency Borate MOV (CVC-440) - A scaffold pic was in direct contact with the motor associated with the MOV. The licensee had not prepared an engineering evaluation to ensure that operability of the Emergency Borate MOV was not negatively impacted.

All of the above listed components were required to be operable for the plant operating condition at the time of discovery.

Analysis:

The inspectors determined that the failure to erect scaffolding near safety-related equipment in accordance with licensee procedure GMP-27, "Requirements and Guidelines for Scaffold Construction and Inspection," was a performance deficiency warranting a significance determination. The finding was more than minor since it impacted the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that responded to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening and determined that the finding was of very low safety significance because there was no actual loss of function of any of the systems.

Enforcement:

10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required that activities affecting quality be prescribed by documented instructions, procedures, or drawings, and that activities be accomplished in accordance with these instructions, procedures, or drawings. Licensee procedure GMP-27, "Requirements and Guidelines for Scaffold Construction and Inspection," a procedure affecting quality, required that scaffolding not be erected within 2 inches from safety-related equipment, unless an engineering evaluation had completed demonstrating that operability of the equipment had not been negatively impacted. Contrary to this requirement, the licensee erected scaffolding in direct contact with, or within 2 inches from the SW cooling outlet line from the cooling water heat exchangers of the 'B' EDG; components and piping associated with the 'A' SI pump; piping associated with the 'A' ICS pump; and the motor associated with the Emergency Borage MOV. This safety-related equipment was required to be operable based on the operating condition of the plant, and the licensee had not completed an engineering evaluation to demonstrate that the operability of any of the equipment was not negatively impacted. Therefore, the inspectors determined that this finding was a violation of 10 CFR 50, Appendix B, Criterion V. The licensee took immediate action to bring all noted scaffolding problems into compliance with procedural requirements. The licensee initiated a CAP document for the issue (CAP 023040). 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 a NCV, consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000305/2004009-05)

.2 Inability to Close Containment Equipment Hatch

Introduction:

A finding of safety significance yet to be determined was identified by the inspectors for an apparent violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, And Drawings." The licensee was unable to close the equipment hatch in an expeditious manner while the plant was in the refueling shutdown mode, spent fuel was in the reactor vessel, the time to boil was estimated to be less than 30 minutes, and the RCS was open to the containment atmosphere.

Description:

Kewaunee entered a refueling and reactor vessel head replacement outage on October 9, 2004. On October 10, the licensee removed the containment equipment hatch. On October 11, the licensee installed steel runway tracks inside and outside of containment to facilitate reactor vessel head replacement activities. It was the licensee's intent to be able to quickly close the equipment hatch when needed by simply moving the exterior track. Also on October 11, a pressurizer safety valve was removed which vented the reactor coolant system to the containment atmosphere. On October 12, Diesel Generator 1A was removed from service. On October 13, detensioning of the reactor vessel head began, which further vented the reactor coolant system to the containment atmosphere.

On October 14, with reactor coolant time to boil estimated to be less than 30 minutes, in preparation for lifting the reactor vessel head, the licensee was required by TSs to close the hatch. The licensee removed the exterior track and attempted to close the hatch. However, the design of the interior track did not take into consideration the curvature of the equipment hatch. Due to this design flaw, the interior track interfered with the closure of the hatch. There were no procedures or plans in place to modify or remove the interior steel runway track rapidly, no tools were staged to modify or remove the interior steel runway track, no personnel were trained to rapidly remove the interior steel runway track, and unsecured heavy material rested on the interior steel runway track and erected scaffolds were adjacent to the interior steel runway track. These factors complicated the decision making process on removal of the interior track and would have been unknowingly encountered in case of an emergency, thus complicating any attempt to rapidly remove the track. The licensee decided to cut away a portion of the interior track so that it would no longer interfere with the hatch.

Following removal of the interference, difficulties were encountered by plant maintenance staff in bolting the hatch in place in accordance with plant procedure CMP 89A-2, which called for using four specific bolts, due to the unavailability of correctly sized ladders. Containment equipment hatch closure was achieved approximately 8 hours after the discovery of the interference. Later in the outage on Nov. 13, 2004, plant personnel again had difficulties closing the hatch. In this instance, six bolts were used to secure the hatch instead of four as called for by the procedure.

The inability of the licensee to close the hatch in an expeditious manner with the reactor coolant system vented to containment, time to boil less than 30 minutes, and one diesel generator out-of-service, was considered by the inspectors to be a potentially risk significant condition. Therefore, this issue was evaluated using the significance determination process.

Analysis:

MC 0308, "Significance Determination Process Basis Document", Attachment 3, Section 5, Performance Deficiency Basis, states that the definition of a performance deficiency requires the staff to make a reasonable determination that the licensee intended to meet some requirement or standard and they did not. Such a requirement need not be directly imposed by the NRC. Licensee good operating practices are expected as a means to ensure safety and minimize risk, and may be implemented as initiatives that go beyond regulatory requirements (e.g. management of shutdown safety by following industry-developed guidelines).

NUMARC 91-06 is an industry standard which provides guidelines for industry actions to assess shutdown risk. NUMARC 91-06, Section 4.1.1, states that containment hatches and other penetrations that communicate with the containment atmosphere should either be closed or capable of being closed prior to core boiling following a loss of decay heat removal and should be addressed in procedures. The licensee implemented initiatives to ensure safety and minimize risk by developing procedures to enable the hatch to be closed if needed. However, due to the poor design of the track, the licensee was unable to meet this procedural guidance during this event. The licensee had numerous opportunities to identify the poor track design. Therefore, based on the statements in MC 0308 and the guidance in MC 0612, this is considered a performance deficiency requiring a significance determination.

The finding was assessed under the IMC 0609, Appendix A, Attachment 1 worksheet for Containment Barriers Cornerstone. The finding was determined to represent an actual open pathway in the physical integrity of reactor containment. As a result, Appendix H of IMC 0609 was used to determine the significance of the finding.

The finding was determined to be a Type B finding (affects only LERF, not CDF) at shutdown. Table 6.3 of IMC 0609, Appendix H is the phase 1 screening for these types of findings. The Kewaunee containment is a PWR, large, dry containment. The containment status was determined to be intact because the licensee planned to maintain an intact containment and the finding involved the failure to maintain the ability to close containment. The SSC specifically affected by the finding is the containment equipment hatch which was determined to fit the category of containment penetration seals, isolation valves, vent and purge systems. The phase 1 assessment resulted in the need to perform a phase 2 assessment.

<u>Phase 2 risk evaluation</u>

<u>Assumptions</u>

- The plant was determined to be in POS 2E which represents cold shutdown with the reactor coolant system (RCS) vented, steam generators not available, and within 8 days of shutdown (decay heat high).

- The finding occurred approximately 64 hours into the shutdown.

- The finding existed for greater than 8 hours. The total time that the containment equipment hatch was open and could not be closed was approximately 80 hours. During this 80 hour period, the RCS was open for approximately 76 hours due to the removal of a pressurizer safety valve and also due to de-tensioning of the reactor vessel head studs. For approximately 67 hours of this period, an emergency diesel generator was unavailable.

- The following equipment was available for the duration of the finding:

- Both SI pumps
- Both RHR pumps
- All charging pumps
- All service water pumps
- Both containment spray pumps
- One EDG
- Two offsite power sources

- The time to boil during this period was estimated by the licensee to be less than 30 minutes.

Given these assumptions, the finding was determined to have potential significance greater than very low significance. Therefore, this finding will be evaluated using the Significance and Enforcement Review Process and a preliminary significance determination for the finding will be provided to the licensee under separate correspondence.

Enforcement:

10 CFR Part 50, Appendix B, Criterion V, (Instructions, Procedures, and Drawings) requires, in part, that activities affecting quality be prescribed by documented instructions, or procedures of the type appropriate to the circumstances and shall be accomplished in accordance with these instructions, or procedures. Plant procedure CMP-89 A-02, "Containment Building Inner Equipment Door Opening and Closing Instructions," a procedure affecting quality, required that any equipment which passes through and could obstruct containment hatch closure be designed to allow rapid removal in order to ensure expeditious containment building equipment hatch closure should it become necessary to do so. Contrary to the above, on October 11, 2004, the licensee installed a interior steel runway track which passed through and obstructed containment hatch closure. The track was not designed to allow rapid removal. This finding did not present an immediate safety concern at the time it was discovered due to the availability of core cooling. The hatch obstruction was removed within hours of discovery and the licensee has initiated a root cause investigation to develop long term corrective actions for this issue. Pending determination of the finding's safety significance, this finding is considered an apparent violation of NRC requirements (AV 05000305/2004009-06).

.3 Reactor Building Ventilation Isolation Function Not Available When Required

Introduction:

A Non-Cited Violation (NCV) of TSs was self-revealed. This NCV was characterized as being of very low safety significance (Green). It became apparent during required daily surveillance testing that radiation monitors would not cause an automatic Reactor Building Ventilation System Isolation to occur as designed.

Description:

On November 18, 2004, at approximately 0803 hours, technicians began surveillance procedure SP-55-155C, “Engineered Safeguards Prestartup Logic Test” with the approval of shift operations. The test placed both trains of Engineered Safeguards in “Test” for the duration of the procedure. By placing Engineered Safeguards in “Test,” valid alarm signals from Radiation Monitors R-12 (Containment Vessel Air Monitor) and R-21 (Containment System Vent Activity Monitor) would not actuate a Reactor Building Ventilation Isolation, such that valves CBV-1, CBV-2, CBV-3, and CBV-4 would not automatically close. These valves would have closed in manual, and would be manually closed per procedure by control room operators in the event of a Radiation Monitor Alarm from either R-12 or R-21. During the period of time when the Reactor Building Ventilation Isolation was defeated, the licensee placed the reactor upper internals into the reactor. At 1630 on the same day, operations personnel conducted a required daily surveillance test that tested the R-12 and R-21 systems ability to generate a Reactor Building Ventilation Isolation. During this test, the R-12 was tested and alarmed properly. However, operators recognized that a Reactor Building Ventilation Isolation failed to occur and that the surveillance test had failed. A short investigation ensued where it was determined that the “Engineered Safeguards Prestartup Logic Test” had defeated the ability for a radiation monitor alarm to generate an automatic Reactor Building Ventilation Isolation.

Analysis:

The inspectors determined that a performance deficiency existed in that the Engineered Safeguards System was allowed to be placed in “Test” thus defeating the Reactor Building Ventilation Isolation system function during a period that it was required to be operable by TS 3.8 a.1.b “Refueling Operations- Containment Closure”. This finding was more than minor because it represented a degradation of the Barrier Integrity Cornerstone objective and was associated with Barrier Integrity Cornerstone attribute of safety system and component (SSC) and barrier performance (containment isolation SSC reliability).

The inspectors completed a significance determination of this issue using IMC 0609, Appendix H, “Containment Integrity Significance Determination Process.” The entry condition identified was that a degraded condition affecting the Containment Barrier Integrity potentially increased Large Early Release Frequency (LERF) without affecting Core Damage Frequency (CDF). For this event, the Plant Operating State (POS) During Shutdown is defined as POS 3 (Reactor Cavity Level is at Refueling Level). The time window that applied for this event was the “Late” time window since there was very low decay heat load. The issue was determined to be a “Type B” finding since the problem was related to containment “integrity” without affecting the likelihood of core damage. Section 6.2 of IMC 0609, Appendix H, “Approach for Addressing Type B Findings At Shutdown,” defined the process for performing a “Phase 2" analysis of this issue. Step 2.1 stated that if the performance deficiency was not related to POS 1 or POS 2 and in the “Early” time window, then the performance deficiency was characterized as a GREEN finding. Therefore, the violation of the Plant TSs was of very low safety significance (Green).

Enforcement:

Plant TSs 3.8 a.1.b required that during refueling operations, each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve. Contrary to this, the licensee allowed a surveillance test to defeat automatic closure features for Reactor Building Isolation such that RBV-1, RBV-2, RBV-3, and RBV-4 containment isolation valves would not automatically close when required by a containment high radiation condition. This violation of Plant TSs was of very low safety significance; therefore, this violation was treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000305/2004009-07). Once this issue was identified, the licensee promptly restored the automatic containment ventilation isolation capability, initiated procedure changes to prevent this issue from recurring and entered the issue into the corrective action program (CAP 024107).

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- Containment Isolation Trip Test;
- EDG Blackout Test;
- AFW Pumps Full Flow Test;
- SI Pumps Full Flow Test;
- Control Rod Drop Time Test - Startup Measurements; and
- Reactor Coolant System Leak Rate Test.

The inspectors verified that the equipment could perform the intended safety function and that the surveillance tests satisfied the requirements contained in plant TSs and licensee procedures. The inspectors reviewed the surveillance tests to verify that the tests adequately demonstrated operational readiness consistent with plant 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)

a. Inspection Scope

The inspectors reviewed the modification documentation and associated 10 CFR 50.59 evaluation for temporary plant modification, completing one inspection sample.

- Temporary Change Request (TCR) 04-13, “Raise the setpoint of SFP (SFP) Temperature Switches 12007 and 12012"

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 CRs to verify that temporary modification problems were entered into the CAP with the appropriate significance characterization

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed an Emergency Preparedness Quarterly Drill on December 14, 2004, completing one emergency planning simulator exercise sample. The inspectors observed activities in the Control Room Simulator, Emergency Operations Facility (EOF), Joint Public Information Center (JPIC) and attended the critique session. The inspectors evaluated the drill performance and determined that the critique activities appropriately captured weaknesses identified by the inspectors and verified that deficiencies were entered into the CAP.

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 licensee event reports, corrective action documents, electronic dosimetry transaction data for radiologically controlled area egress and information reported on the NRC's web site relative to the licensee's occupational exposure control performance indicator (PI) to determine whether or not the conditions surrounding any actual or potential PI occurrences had been evaluated, and identified problems had been entered into the corrective action program for resolution. Performance indicator data collection and analysis methods were evaluated by the inspectors as described in Section 4OA1.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following three radiologically significant work areas within radiation areas, high radiation areas (HRAs) and locked high radiation areas in the plant and reviewed work packages which included associated licensee controls and surveys of these areas to determine if the radiological controls including area postings and barricades were adequate:

- Containment Head Lift Pathway (Containment/Auxiliary Buildings - All Areas);
- Containment "C" Sump Area; and
- Containment Seal Table Area.

The inspectors reviewed the radiation work permit (RWP) and associated work packages which governed work activities and access into these areas and into other selected high radiation areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to verify that they were aware of the actions required when their electronic dosimeters malfunctioned or alarmed.

The inspectors walked down and surveyed (using an NRC survey meter) these areas in addition to other radiologically significant area boundaries to verify that the prescribed RWP, procedure, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers were properly located. During the walkdowns, the inspectors challenged access control boundaries to verify that high and locked high radiation area access was controlled in compliance with the licensee's procedures, plant TSs and the requirements of 10 CFR 20.1601.

The inspectors reviewed RWPs for the following airborne radioactivity area to verify barrier integrity and engineering controls performance (e.g., filtered ventilation system operation) and to determine if there was a potential for individual worker internal exposures of > 50 millirem committed effective dose equivalent: Containment Seal Table. Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had performed surveys to determine the potential for transuranic isotopes.

The inspectors reviewed the licensee's procedures and its methods for the assessment of internal dose as required by 10 CFR 20.1204, to ensure methodologies were technically accurate and would include the impact of hard to detect radionuclides such as pure beta and alpha emitters, if applicable. No worker intakes that resulted in a committed effective dose equivalent (CEDE) in excess of 50 millirem occurred during the outage. However, internal dose assessments which resulted in exposures less than 50 milllirem CEDE were selected reviewed by the inspectors for adequacy.

The inspectors reviewed the licensee's practices and programmatic controls which prohibited the temporary storage of highly activated and/or contaminated materials (non-fuel) within the spent fuel pool attached to cables/lanyards and consequently easily removable from the pool. Specifically, radiation protection staff were interviewed and a walkdown of the refuel floor was performed to verify the licensee's practices.

These reviews represented six inspection samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed Licensee Event Reports (LERs) and Special Reports, as applicable, related to the access control program to verify that identified problems were entered into the corrective action program for resolution. Review of LER 2004-002 related to leak testing of sealed sources was discussed in Section 4OA3.

The inspectors reviewed the licensee's corrective action program database for 2004, and several corrective action reports related to access and exposure controls and three related to high radiation area radiological incidents (non-Performance Indicator issues identified by the licensee in high radiation areas < 1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions; and
- Implementation/consideration of risk significant operational experience feedback.

The inspectors evaluated the licensee's process for problem identification, characterization, prioritization, and verified that problems were entered into the corrective action program and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies.

The inspectors reviewed licensee documentation packages for all PI or potential PI events occurring since an occurrence was last reported for an April 15, 2003, event. The review was conducted to determine if any events involved dose rates > 25 R/hr at 30 centimeters or > 500 R/hr at 1 meter. Unintended exposures > 100 millirem total effective dose equivalent (or > 5 rem shallow dose equivalent or > 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. No examples of these type of PI events occurred.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed five jobs that were being performed in radiation areas, high radiation areas (HRAs) and/or locked high radiation areas to evaluate work activities that presented the greatest radiological risk to workers. This review was conducted in conjunction with Inspection Procedure 71121.02, and was documented in Section 2OS2.4 of this report.

The inspectors also reviewed the licensee's procedure and generic practices associated with dosimetry placement and the use of multiple whole body dosimetry for work in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201(c) and applicable industry guidelines.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors held discussions with 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.

The inspectors discussed with radiation protection (RP) supervisors the controls that were in place for special areas that had the potential to become very high radiation areas during certain plant operations, to determine if these plant operations required communication beforehand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

The inspectors conducted plant walk downs to verify the posting and locking of entrances to selected locked high radiation areas, high dose rate high radiation areas, and Very High Radiation Areas (VHRAs).

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements and evaluated whether workers were aware of the significant radiological conditions in their workplace, the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present.

The inspectors reviewed three radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. These problems, along with planned and taken corrective actions were discussed with Radiation Protection Management.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation protection technician performance with respect to radiation protection work requirements and evaluated whether they were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed four radiological problem reports which found that the potential cause of the event was radiation protection technician error to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the identified problems.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

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

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective outage exposure history, current exposure trends and ongoing outage activities in order to assess current performance and exposure challenges. This included determining the plant’s current 3-year rolling average for collective exposure in order to help establish resource allocations and to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed the outage work scheduled during the inspection period and associated work activity exposure and time/labor estimates for the following six work activities which resulted in the highest personnel collective exposures or were otherwise activities that were conducted in radiologically significant areas:

- Bottom Mounted Insulation Replacement;
- Outage Containment Scaffolding;
- Reactor Vessel Closure Head (RVCH) Disassembly/Reassembly;
- Reactor Disassembly/Reassembly;

- In-Service Inspection; and
- Reactor Coolant Pump Seals.

The inspectors determined the site specific trends in collective exposures based on plant historical exposure and source term data. The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and assessed those processes used to estimate and track work activity exposures.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were completed during the outage and reviewed the following six work activities of highest exposure significance:

- Bottom Mounted Insulation Removal;
- Outage Containment Scaffolding;
- RVCH Disassembly/Reassembly;
- Reactor Disassembly/Reassembly;
- In-Service Inspection; and
- Reactor Coolant Pump Seals.

For the activities listed above, the inspectors reviewed the ALARA Plan and associated RWP, exposure estimates, and exposure mitigation requirements in order to verify that the licensee had established radiological engineering controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the exposure results achieved for its combined refueling and reactor head replacement outage, including the dose rate reductions and person-rem expended, with the dose projected in the licensee's ALARA planning for these work activities. Reasons for inconsistencies between intended (projected) and actual work activity doses were evaluated to determine if the activities were planned adequately and to ensure the licensee identified any work interface/planning deficiencies. Those jobs that accrued greater than 5 rem and that exceeded their respective initial dose estimates by greater than 50 percent were investigated by the inspectors. The investigations were conducted to determine if deficiencies with radiological planning or with work execution contributed significantly to the dose overages which the licensee should reasonably have identified and prevented.

The interfaces between radiation protection, plant engineering and scheduling groups were reviewed to varying degrees to identify potential interface problems. The integration of ALARA requirements into work procedure and RWP documents was evaluated to verify that the licensee's radiological job planning would reduce dose.

The inspectors compared the person-hour estimates provided by maintenance planning and/or craft groups to the radiation protection ALARA staff with the actual work activity time expenditures in order to evaluate the accuracy of these time estimates.

The inspectors evaluated if work activity planning included consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components/piping, system flushing and hydrolazing and sequencing of scaffold and shielding installation/removal in order to maximize dose reduction.

The licensee's work in progress reports were reviewed for those outage jobs that accrued collective exposures between 50 and 100 percent of that projected to verify that the licensee could identify problems and address them as work progressed. Jobs that accrued greater than one rem and exceed 125 percent of the projected doses were also reviewed to ensure work was suspended, if warranted, and identified problems were entered into the corrective action program consistent with the licensee's procedure. Additionally, post job reviews being developed during the latter stages of the inspection period were discussed with the licensee's ALARA staff to determine the scope and breadth of the deficiencies that were identified and the status of documenting outage lessons learned.

These reviews represented eight inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's assumptions and basis for its collective outage exposure estimate, and evaluated the methodology and practices for projecting work activity specific exposures. This included evaluating both dose rate and time/labor estimates for adequacy compared to historical station specific or industry data.

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which significantly impacted worker exposures. This included determining that adjustments to estimated exposure (intended dose) were based on radiation protection and ALARA principles and not adjusted to account for failures to plan or control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process.

The licensee's exposure tracking system was evaluated to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of collective exposures. RWPs were reviewed to determine if they covered too many work activities to allow work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased significantly beyond exposure estimates.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following five jobs that were being performed in radiation areas, airborne radioactivity areas, or high/locked high radiation areas to evaluate those work activities that presented the greatest radiological risk to workers:

- Bottom Mounted Insulation Replacement;
- Outage Containment Scaffolding;
- RVCH Disassembly;
- In-Service Inspection & Support; and
- Reactor Coolant Pump Seals.

The licensee's use of ALARA controls for these work activities was evaluated using the following:

> The licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

Job sites were observed to determine if workers were utilizing the low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays.

The inspectors attended work briefings and observed ongoing work activities to determine if workers received appropriate on-the-job supervision to ensure the ALARA requirements are met. This included verification that the first-line job supervisor ensured that the work activity was conducted in a dose efficient manner by minimizing work crew size, ensuring that workers were properly trained, and that proper tools and equipment were available when the job started.

The inspectors reviewed the exposures of individuals involved in the Bottom Mounted Insulation Replacement Project. Worker exposures were reviewed to determine whether any significant variations were the result of poor ALARA work practices.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to understand historical trends and current status of plant source terms. The inspectors discussed the plant's source term with ALARA staff to determine if the licensee had developed an adequate understanding of the input mechanisms and the methodologies and practices necessary to achieve reductions in source term. The inspectors discussed the water chemistry control initiatives implemented during the cool-down for the outage and its impact on source term reduction compared to industry practices.

While the licensee did not have a formal source term control strategy in place, source term reduction initiatives typically implemented by the licensee were discussed with ALARA staff as were plans for the development of a long term reduction plan. The inspectors determined if specific sources had been identified by the licensee for exposure reduction initiatives and that priorities were being considered for the implementation of these actions.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was observed during work activities being performed in radiation areas, airborne radioactivity areas, and high radiation areas that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas, and that they had knowledge of the radiological conditions and adhered to the ALARA requirements for the work activity. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.7 Monitoring of Declared Pregnant Women and Dose to Embryo/Fetus

a. Inspection Scope

The inspectors reviewed the licensee's monitoring methods and procedures, exposure controls, and the information provided to declared pregnant women to determine if an adequate program had been implemented to limit embryo/fetal dose. The inspectors also reviewed the pregnancy declaration and radiation exposure results for several individuals that declared their pregnancy to the licensee in 2003 through December 2004, to verify compliance with the requirements of 10 CFR 20.1208 and 20.2106.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.8 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and Special Reports related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational Cornerstone met the requirements of 10 CFR 20.1101(c).

Several corrective action reports related to the ALARA program were reviewed and staff members were interviewed to verify that follow-up activities had been conducted in an effective and timely manner commensurate with their importance to safety and risk using the following criteria:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions; and
- Implementation/consideration of risk significant operational experience feedback.

The inspectors reviewed and/or discussed with ALARA staff its ongoing post-job reviews of outage exposure performance. The inspectors determined whether dose performance issues were being adequately characterized, prioritized and resolution was being sought through the corrective action process.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Waste Characterization and Classification of the Old RVCH

a. Inspection Scope

The inspectors reviewed the licensee's waste stream radiochemical sample analysis results, radiological surveys, and shielding and source term calculations that were used to develop the Class "A" waste characterization of the old RVCH. These reviews were conducted to verify that the licensee's characterization assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR 20. Additionally, the inspectors reviewed the licensee's calculations used to determine the Department of Transportation sub-typing for the shipment of the RVCH, so as to verify the Low Specific Activity (LSA)-II sub-typing complied with 49 CFR 172, 173, and 177.

No samples under the baseline inspection procedure were completed by this review.

b. Findings

No findings of significance were identified.

.2 Shipment Preparation and Shipping Records for the Old RVCH

a. Inspection Scope

The inspectors reviewed the licensee's procedures and documentation (including photographs) for shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness for the shipment of the old RVCH to the low-level radioactive waste disposal facility, Envirocare of Utah, Inc., in Clive, Utah. The inspectors selectively verified that the requirements of 10 CFR 20 and 61 and those of the Department of Transportation in 49 CFR 170-189 were met for the RVCH shipment to Envirocare.

No samples under the baseline inspection procedure were completed by this review.

b. Findings

No findings of significance were identified.

2PS3 Radioactive Material Control (71122.03)

1. Temporary Storage of the Old RVCH

a. Inspection Scope

The inspectors reviewed licensee controls and surveys of the temporary storage location of the old RVCH in Containment (prior to packaging and shipment), to determine if radiological controls including surveys, postings and barricades were acceptable. Additionally, the inspectors evaluated the licensee’s contamination and engineering controls in place around the temporary storage location of the old RVCH while the containment equipment hatch was open to verify that any contamination on the old RVCH did not contribute to an unmonitored airborne effluent or liquid radioactive material release pathway from the plant.

No samples under the baseline inspection procedure are completed by this review.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors reviewed the licensee submittals for the following PIs, completing two PI verification inspection samples:

- Safety System Functional Failure; and
- High Pressure Injection Unavailability;

The inspectors used PI guidance and definitions contained in Nuclear Energy Institute Document 99-02, Revision 2, “Regulatory Assessment PI Guideline,” to verify the accuracy of the PI data for the first, second and third quarters 2004. The inspectors’ review included, but was not limited to, conditions and data from logs, CRs, and calculations for each PI specified. The inspectors also reviewed CRs to verify that licensee personnel identified issues at an appropriate threshold and entered them into the CAP in accordance with station CA procedures.

b. Findings

No findings of significance were identified.

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicator (PI) listed below for the period May 2003 through mid-December 2004. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The following PI was reviewed:

- Occupational Exposure Control Effectiveness

For the time period reviewed, no reportable occurrences were identified by the licensee. (A TS occurrence involving the unauthorized removal of a flashing red light that was used to control access into a LHRA was identified by the licensee on April 15, 2003, and was reported as required for the second quarter of 2003). To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff the scope and breadth of its PI data review and the results of those reviews. The inspectors independently reviewed selected electronic dosimetry dose alarm reports (radiologically controlled area electronic dosimetry egress transactions), the personnel contamination report for the outage, dose assignments for intakes, and the licensee's CAP database along with individual CAPs generated during the period reviewed to verify there were no unrecognized occurrences. Additionally, as discussed in Sections 2OS1.2 and 2OS1.5, the inspectors walked down the boundaries of selected locked high radiation areas to verify the adequacy of postings and access control physical barriers.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 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 program at an appropriate threshold, that adequate attention was given to timely CAs, and that adverse trends were identified and addressed. The inspectors also reviewed all CAPs written by licensee personnel during the inspection quarter. Minor issues entered into the licensee's CAP program as a result of inspectors' observations were included in the list of documents in the Attachment in the section entitled "Condition Reports Initiated for NRC Identified Issues."

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope:

The inspectors selected Condition Report CAP 021915, "Hydrogen and Propane Gas Lines Are Not Identified in the Fire Strategies," for an annual sample review of the licensee's problem identification and resolution program. This constitutes one annual review inspection procedure sample.

b. Findings

Introduction:

The inspectors identified a NCV of License Condition fire protection requirements having very low safety significance (Green) for not identifying pertinent information, such as the presence of compressed flammable gas cylinders, on fire area strategies.

Description:

The failure to identify hydrogen and propane gas lines passing trough a fire zone in a pre-fire plan (PFP) had been identified by the NRC as part of a triennial fire protection inspection (documented in Section 1R05.10.b.2 of Inspection Report 05000305/2004005). The licensee entered this issue in their corrective action program under CAP 021915 at that time and subsequently revised one PFP to note the existence of the hydrogen and propane lines. Based on discussions with fire protection personnel, the inspectors learned that no other PFP's were reviewed to verify that they included pertinent information or determined the extent of condition.

During this inspection, the inspectors identified that PFP-17, as of October 22, 2004, did not identify that there were combustible gas cylinders within Fire Zone 23B. On October 22, 2004, and December 1, 2004, the inspectors observed a number of compressed gas cylinders with combustible concentrations of flammable gas. The gas cylinders included a propane cylinder and a number cylinders containing mixtures of hydrogen and nitrogen. These compressed flammable gas cylinders were located near doors 196 and 255 on the 586 foot elevation of Fire Zone 23B within the auxiliary building. However, the layout diagram for PFP-17, the applicable fire area strategy for the 586 foot elevation of Fire Zone 23B, only identified that compressed gases (versus compressed flammable gases) were stored in this area of the auxiliary building. The text for PFP-17 identified lubricating oil in pumps as the only flammable or combustible gas or liquid. PFP-17 did not mention the presence of compressed hydrogen and propane gases as being present. In addition, the inspectors identified a number of discrepancies between the hazards identified in the fire zone summaries of the Fire Protection Program Analysis versus what had been identified in the applicable PFP for the fire zone.

Analysis:

In accordance with IMC 0612, "Power Reactor Inspection Reports," dated January 14, 2004, the inspectors determined that the issue of not maintaining acceptable fire pre-plans was a performance deficiency. This performance deficiency was determined to be greater than minor because it affected the mitigating systems cornerstone attribute of protection against external factors (fire). Specifically, the failure to provide adequate warnings and guidance relating to hazards associated with hydrogen and propane compressed gas cylinders in fire strategies could adversely impact fire fighting strategies used by the fire brigade in fighting a fire. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated, March 18, 2002, the inspectors performed a SDP Phase 1 screening and determined that the finding affected fire protection defense-in-depth. As such, the inspectors determined that a Phase 2 analysis in accordance with IMC 0609, Appendix F, "Fire Protection SDP," dated May 28, 2004, was required. As discussed by IMC 0308, Attachment 3, Appendix F, "Technical Basis, Fire Protection Significance Determination Process (IMC 609 App. F) At Power Operations," the current significance determination process did not address findings which affected the performance of the fire brigade. As such, the inspectors used judgement based on experience to determine the safety significance of the issue. The inspectors determined that the issue was of very low safety significance (Green) due to extensive training provided to fire brigade members to deal with unexpected contingencies.

Enforcement:

KNPP License Condition 2.C(3), required, in part, that NMC implement and maintain in effect all provisions of the approved fire protection program as described in the KNPP Fire Plan. Section 10.3, "Fire Area Strategies," of the KNPP Fire Protection Program Plan specified that fire area strategies were documents which provided the fire brigade pertinent information on a given plant area to help the brigade to be better prepared for fire fighting within that area. PFP-17 was the fire area strategy for the 586 foot elevation of Fire Zone AX-23B. Contrary to the above, PFP-17 did not contain pertinent information on a given plant area, the 586 foot elevation of Fire Zone 23B, in that the fire area strategy did not identify that compressed gas cylinders in the area contained flammable gases. Once this issue was identified, the licensee entered the issue into their corrective action program under CAP 023479. The licensee subsequently informed the inspectors that PFP-17 had been specifically revised to note the presence of the compressed flammable gas cylinders. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000305/2004009-08).

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 50-305/2004-003-00: Control Room Boundary Door Found Ajar - Accident Analysis Assumptions Impacted - Personnel Error

On August 12, 2004, while the plant was operating at full power, on-shift plant Operations department personnel, during a normal operating equipment tour, discovered a control room emergency zone barrier door (Door #152) not fully closed. Full closure of this door is required to ensure operability of the Control Room Post Accident Recirculation system. The underlying causes of the failure were considered to be deficiencies in the plants overall barrier control program. This finding was more than minor because, if left uncorrected, the issue would have become a more significant safety concern. In addition, it affected the Mitigating Systems attributes of equipment performance reliability and the Mitigating Systems Cornerstone objective of insuring the reliability of systems. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening. Phase 1 screening required that this finding be evaluated using Phase 3. The Phase 3 result identified this finding as Green due to the very short duration in which the situation associated with this finding existed. Therefore, the finding was determined to be of very low safety significance (Green). This licensee-identified finding involved a violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

.2 (Closed) LER 50-305/2004-002-00: Missed TS Surveillance for Leak Testing In-Core Detectors Prior to Use or Transfer, Due Inadequate Procedural Guidance

On July 21, 2004, in response to an Operating Experience Notice (CAP 019783) review, NMC Nuclear Oversight personnel discovered that TS 4.13 requirement for in-core detectors containing byproduct materials greater than 0.1 microcuries was not met for 13 detectors that were transferred to another licensee. Technical Specification 4.13 required leak test detectors that were in storage prior to removal for use or prior to shipment to other licensed entities. The licensee identified that the root cause for the missed leak test was the failure of the existing procedures to properly control required TS requirements and the inadequate degree of instructional details in the procedure RE 05, “In-core Instrumentation Periodic Hardware Maintenance,” and RE-24, “Special Nuclear Material Control.” Corrective Actions prescribed by the licensee included:

- present and discuss the issue with the radiation protection staff;
- revise the two procedures (RE-05 and RE-24);
- contact the licensee that received the in-core detectors from Kewaunee and request that they perform a leak test on those in-core detectors received from Kewaunee.

The subsequent leak tests did not identify any leaking detectors. The failure to leak test the in-core detectors prior to transfer constituted a violation of minor significance that was 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 leak test the detectors in CAP 021686. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R05.1.b.1 of this report was related to the cross-cutting area of problem identification and resolution, related to the performance characteristic of corrective actions. Specifically, the licensee's corrective actions were ineffective in that the NRC had previously identified incidents involving unauthorized storage of combustible materials above shelves in the materials storage working area and nearby. The inspectors identified additional examples during this inspection.

.2 A finding described in Section 4OA5.2.c.1 of this report was related to the cross-cutting area of problem identification and resolution, related to the performance characteristic of corrective actions. Specifically, the licensee failed to take corrective actions for conditions adverse to quality related to the sump screen openings.

4OA5 Other Activities

.1 Reactor Pressure Vessel (RPV) Lower Head Penetration Nozzles (TI 2515/152)

a. Inspection Scope

The inspectors performed a review of licensee activities in response to NRC Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," in accordance with NRC Temporary Instruction (TI) 2515/152, "Reactor Pressure Vessel Lower Head Penetration Nozzles." The reviewed the licensee's procedures, equipment, and personnel used for RPV lower head penetration examinations to confirm that the licensee met commitments associated with Bulletin 2003-02. The results of the inspectors' review included documentation of observations and conclusions in response to the questions identified in TI 2515/152.

b. Findings:

Based upon a bare metal visual (BMV) examination of the lower head, the licensee did not identify evidence of reactor coolant system leakage near the instrument nozzle penetrations. Several areas of white streaking and rust colored residue were observed on the bare metal of the reactor vessel bottom head located around the 36 bottom mounted instrumentation. The licensee believed that these stained areas were caused by liquid which had rundown from reactor vessel cavity leakage.

Evaluation of Inspection Requirements

In accordance with requirements of TI 2515/152, the inspectors evaluated and answered the following questions:

For each of the examinations methods used during the outage, was the examination:

1. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes. The licensee conducted a direct visual examination of the RPV lower head penetration interface and RPV lower head surface for leakage or boric acid deposits with knowledgeable staff members certified to Level III and Level II as VT-3 examiners. One examiner was a licensee staff member certified to licensee procedure FP-PE-NDE-3, “Written Practice For Qualification And Certification For NDE Personnel,” and GNP-01.05.05, Revision A, “Written Practice for Qualification and Certification of Kewaunee Nuclear Power Plant Personnel in Visual Examination Methods (VT);” the other was a licensee contractor certified to the contractors procedure QA-45 Revision 1, “Qualification and Certification of NDE and Visual Examination Personnel per ASME Section XI, 2000 Addendum.” These qualification and certification procedures were consistent with the requirements of industry standard ANSI/ANST CP-189, “Standard for Qualification and Certification of Nondestructive Testing Personnel,” and/or ASNT-SNT-TC-1A-1984. Additionally, each of the VT-2 examination personnel had reviewed photographs of the boric acid deposits indicative of penetration leakage found at the South Texas Nuclear Power Plant.

2. Performed in accordance with demonstrated procedures?

   Yes. The licensee performed a bare metal inspection of the lower head in accordance with procedure NEP 15.05, Revision A, “Visual Examination for Inservice Inspection.” The licensee considered this procedure to be demonstrated because their examination personnel could resolve the lower case alpha numeric characters 0.105 inches in height at a maximum of 4 feet under existing lighting to meet Code VT-3 inspection criterion. In addition, the licensee had specific guidance or reference, written paper, “Sampling and Analysis Guidance for Deposits Found on Reactor Pressure Vessels at Various Locations,” for when and how to take samples of deposits if any had been identified near the interface of lower head penetrations and what analysis would be performed to determine the source of deposits identified.

   However, the inspectors identified parameters that could impact the quality/effectiveness of the inspection and were not controlled by the procedure. Specifically, the procedure did not provide:

   - specific guidance to identify recordable indications of corrosion or wastage if it had been present on the lower head. Note that no significant corrosion or wastage was present based upon the NRC inspectors’ inspection of the head; and
   - useful orientation and penetration numbering figure/schematic for the BMI penetrations.

   The inspectors performed an independent direct bare metal visual examination for most of the 36 lower head penetration nozzles. This inspection was conducted from a platform under the vessel head and the inspectors determined that each penetration was readily accessible such that the visual examination could be performed within a few inches of each penetration location. Additionally, the inspectors reviewed a sample of licensee photographs taken at each penetration nozzle. Based upon this inspection and interviews with inspection staff, the inspectors did not identify any concerns associated with implementation of the visual inspection procedure for the lower head.

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

   Yes. Several areas of white streaking and rust colored residue were observed on the bare metal of the reactor vessel bottom head located around the 36 bottom mounted instrumentation penetrations. The licensee believed that these stained areas were caused by rundown from liquid sources above the bottom of the vessel. Chemistry analysis was performed and the results indicated that the white streaking was attributed to reactor vessel cavity leakage. Based upon the visual examination, the licensee did not identify any penetrations with boric acid deposits indicative of coolant leakage.

4. Capable of identifying pressure boundary leakage as described in the bulletin and/or RPV lower head corrosion?

   Yes. The inspectors performed a direct visual inspection of portions of the 36 lower VHPs. Based on this examination, and interviews with licensee examiners, the inspectors concluded that the visual examination was capable of detecting deposits indicative of pressure boundary leakage as described in the bulletin.

5. Could small boric acid deposits representing reactor coolant system leakage as described in Bulletin 2003-02 be identified and characterized, if present, by the visual examination method used?

   Yes. If small boric acid deposits characteristic/indicative of leakage had existed, the licensee's examination would have identified these. However, no boric acid deposits indicative of leakage were identified.

6. How was the visual inspection conducted (e.g., with video camera or direct visual by examination personnel)?

   Licensee personnel conducted a direct visual examination of each of the lower head penetration nozzles. This examination included a bare metal visual examination of the lower head up to the transition to the vertical vessel shell wall. In addition, photographs were taken of all the instrumentation penetrations and the surface area of the reactor vessel lower head.

7. How complete was the coverage (e.g., 360 degrees around the circumference of all the nozzles)?

   The examination coverage included a 360 degree unobstructed examination of each of the 36 lower head penetration nozzles at the interface of the vessel head. The entire lower head was accessible for a visual inspection to identify corrosion and wastage.

8. What was the physical condition of the RPV lower head (e.g., debris, insulation, dirt, deposits from any source, physical layout, viewing obstructions)? Did it appear that there are any boric acid deposits at the interface between the vessel and the penetrations?

   The Kewaunee reactor pressure vessel was installed with mirror-type insulation at the lower RPV dome. This insulation generally conformed to the contour of the lower RPV dome but had a gap of about 1 - 3 inches between the RPV surface and insulation. Each BMI penetration had a slight gap that varied in size and was normally covered by metal flashing. The licensee intended to install a revised lower head insulation structure with a tub type configuration (e.g., horizontal insulation floor with vertical walls). This revised insulation design provided for access doors in the vertical and horizontal walls to allow access for future bare metal head inspections. For this inspection, all of the lower insulation had been removed to provide unobstructed access to the BMI penetrations. This inspection was conducted from a platform under the vessel head and the inspectors determined that each penetration was readily accessible such that the visual examination could be performed within a few inches of each penetration location. A specific description of the RPV lower head is contained in the answer to Question 3 above. Based upon the inspectors' inspection, they did not identify any boric acid deposits at the interface between the vessel and the penetrations .

9. What material deficiencies (i.e., crack, corrosion, etc.) were identified that required repair?

   None. No boric acid deposits indicative of leakage were identified and thus no repairs were required.

10. What, if any, impediments to effective examinations, for each of the applied nondestructive examination method, were identified (e.g., insulation, instrumentation, nozzle distortion)?

    The direct visual examination required access to the RPV lower head and instrument nozzle penetrations by climbing down a ladder, into the keyway (a sump area under the vessel). This area was a confined space, a high radiation area, and was congested by the instrument tubes and their supports. Scaffold had been installed to support removal of the lower insulation and to allow access for direct inspection of the BMI penetrations. With the insulation removed, each penetration was accessible from this platform for direct visual inspection.

11. Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPV lower head?

As noted in the answer to Question 3 above, the licensee did identify white streaking which they attributed to reactivity cavity seal leakage. However, as noted in the answer to Question 12 below, this leakage was no longer active.

12. Did the licensee take any chemical samples of the deposits? What type of chemical analysis was performed (e.g., Fourier Transform Infrared(FTIR)), what constituents were looked for (e.g., boron, lithium, specific isotopes), and what were the licensee's criteria for determining any boric acid deposits were not from RCS leakage (e.g., Li-7, ratio of specific isotopes, etc.)?

    Yes. The licensee collected samples of deposits from five locations on the reactor lower head. A control swipe was also taken from an area on the head that had no indications of boric acid or other noticeable deposits. All samples were counted for qualitative isotopic analysis. Sampling and analysis methodologies were based on a document prepared by Electric Power Research Institute (EPRI) and member utilities titled, "Sampling and Analysis Guidance for Deposits Found on Reactor Pressure Vessels at Various Locations," dated September 2003. Based on the observations from the Gamma Isotopic analyses, a lack of short-lived isotopes indicated no active leakage. Furthermore, there were only two isotopes present, Co-60 and Cs-137. Using the radionuclide ratio of Co-60/Cs-137 as a method to identify the leakage, all the samples' ratios as well as the typical ratio in the Refueling Water Storage Tank (RWST) were whole numbers equal to 2.6 or greater; the typical ratio for Reactor Coolant System (RCS) ratio is 0.6. The last notable observation from the isotopic data was that the results of the residue and swipe samples #1 - 4 were not noticeably different than swipe #5, the "control swipe" taken from a residue free area of the vessel. Therefore, the licensee concluded that the white streaking was attributable to reactivity cavity seal leakage that was no long active.

13. Is the licensee planning to do any cleaning of the head?

    Yes. The licensee planned to clean the head with demineralized water and scotch-bright pads.

14. What are the licensee's conclusions regarding the origin of any deposits present and what is the licensee's rationale for the conclusions?

    The licensee concluded that the residue was not from an active leak, but that these stained areas were caused by liquid which had rundown from reactor vessel cavity leakage. The licensee's rationale for this was based on the results obtained from isotopic analysis of the samples obtained.

.2 <u>Reactor Containment Sump Blockage</u> (TI 2515/153)

a. <u>Inspection Scope</u>

The inspectors performed a preliminary review of licensee activities in response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump

Recirculation at Pressurized Water Reactors (PWRs)," in accordance with NRC Temporary Instruction (TI) 2515/153, "Reactor Containment Sump Blockage (NRC Bulletin 2003-01)," dated October 3, 2003. The inspectors reviewed the licensee's completed and proposed compensatory measures submitted in accordance with Bulletin 2003-01, Option 2, which were contained in the licensee's correspondence to the NRC dated August 7, 2003, and May 17, 2004. The inspectors verified that the compensatory measures committed to were implemented, or were planned and scheduled for implementation consistent with the licensee's response. In accordance with TI 2515/153 Section 04.02.b, the inspectors discussed the licensee's response with the NRR Project Manager since a NRR acknowledgment letter had not been issued for the licensee's response at the time of the inspection.

Visual inspections of the containment sumps, sump screens and flow paths were performed by the inspectors during the refueling outage. The inspectors also walked down containment to verify that the condition of the containment coatings, piping insulation, post Loss-of-Coolant-Accident (LOCA) drainage paths, and Emergency Core Cooling System (ECCS) recirculation sumps were consistent with the conditions reported and documented by the licensee. The inspectors interviewed operating and engineering personnel and reviewed training records, procedures for foreign material control and containment inspection, and the results of licensee containment coating and debris generation inspections.

b. Findings:

The following information is provided as required by Section 5, "Reporting Requirements," of TI 2515/153.

During this inspection period Kewaunee completed a refueling outage (Refueling Outage Number R27) and subsequently returned to power. In addition, the inspectors verified the licensee had performed similar inspections in the Refueling Outage which occurred 18 months prior in the Spring of 2003 (Refueling Outage Number R26). During the refueling outages, containment walkdowns were conducted by the licensee to further quantify and in some cases remove potential debris sources. During the walkdowns, the inspectors verified that the licensee's current quantification of potential debris sources was accurate. The licensee's walkdown also checked for gaps in the sump screen and for major obstructions in the containment upstream of the sump. The inspectors did identify two issues related to the containment sump during this inspection, which were discussed further in Sections 4OA5.2.c.1 and 4OA5.1.c.2 of this report.

Licensee engineers stated that advance long term preparations were being made to expedite the performance of sump-related modifications, in case the licensee determined modifications were necessary after performing the sump evaluation. At the time of the inspection, these actions included the initiation of additional engineering evaluations by the licensee.

Finally, the inspectors verified that the compensatory measures committed to by the licensee in correspondence to the NRC dated August 7, 2003, and May 17, 2004, were either implemented or scheduled for implementation in accordance with the timetables committed to by the licensee. The inspectors determined that the licensee met the current commitments, with one minor exception. Commitment 1 in the licensee's August 7, 2003, submittal stated, in part, that NMC would develop and implement training on sump clogging by December 19, 2003, as a compensatory measure. The licensee's correspondence further clarified that a sump clogging training module would be developed and administered to license operators, auxiliary operators, and Emergency Directors. The sump clogging training was comprised of seven topics, which included a review of the importance of aggressively cooling the reactor coolant system in order to transition to shutdown cooling as soon as possible to avoid recirculation cooling, and a review of the content and implementation of the severe accident management guidelines, including actions available to respond to sump clogging.

During a review of the training module and records, the inspectors identified that the sump clogging training module given to the licensed operators covered five topics, but did not address the importance of aggressive cooling and did not review the content and implementation of the severe accident management guidelines. In addition, the inspectors noted that the sump clogging training had not been given to the auxiliary operators and Emergency Directors. Finally, the inspectors identified that the licensee had not established a program to ensure that the sump clogging training was given to new licensed operators, auxiliary operators and Emergency Directors while the compensatory measures remained in effect, until the licensee completed the final sump analysis. The inspectors determined the failure to meet this commitment was of minor significance, and the licensee initiated Condition Report CAP 023615. In addition, the licensee conducted the training which was committed to prior to the startup of the plant from Refueling Outage 27.

b.1 <u>Non-conforming Condition on the Safety-Related Containment Sump</u>

<u>Introduction:</u>

A finding of very low safety significance (Green) was identified by the inspectors for a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." During a review of the licensing and design basis of the containment sump screens, the inspectors noted that the screen size allowed particles greater than 1/8 inch to enter the sump, when the original licensing basis of the screens was to prevent any particles greater than 1/8 inch from entering the sump. The inspectors subsequently determined this issue was identified in the licensee's corrective action program; however, adequate corrective actions were not taken to correct this condition adverse to quality.

<u>Description:</u>

The inspectors performed an inspection of the licensee's conical sump screens and noted that the sump screen opening size on both screens was approximately 1/8 inch by 15/32-inch. The inspectors subsequently reviewed the design and licensing basis of the sump screens to verify the original screens were properly constructed in accordance with the design and licensing basis.

On September 23, 1971, the Atomic Energy Commission (AEC) issued a request for additional information related to the review of the Final Safety Analysis Report (FSAR) for the Kewaunee plant. Question 6.19 in the request from the AEC to Wisconsin Public Service (WPS) Corporation stated, “Provide a description of the screens installed for the containment sumps, including the size of foreign matter that will be precluded from entering the recirculation system.” Wisconsin Public Service Corporation responded to the AEC’s Question 6.19 in a docketed letter dated December 15, 1971, which transmitted FSAR Amendment 13, and annotated that the response to Question 6.19 was located on Page 6.2-9. Section 6.2.2, “Recirculation Phase,” stated, “Foreign matter is prevented from entering the recirculation system by two screens mounted over the sump inlet. These screens are conical in shape, manufactured of Johnson Screen material and sized to prevent any particles larger than 1/8 inch from entering the sump.” Based on licensee documentation for the purchase of the safety related screens in May 1973, the screens were ordered with a 1/8 inch slot opening and support rods placed on 5/8 inch center which created a 1/8 inch by 15/32 inch screen opening. Therefore, at the time of installation in 1973, the two conical sump screens would have allowed particles larger than 1/8 inch to enter the sump. The inspectors determined that no modifications were made to the sump since the time of original installation, and no correspondence was submitted to the AEC discussing the change in the size of particle which could enter the sump.

The inspectors noted that the current Updated Safety Analysis Report (USAR), Revision 18, Section 6.2.2, stated, “These screens are conical in shape, manufactured of Johnson Screen material and sized to prevent any particles with a mean diameter greater than 1/8 inch from entering the sump.” The inspectors determined that the change from the original FSAR occurred with USAR Change Request R16-029, in September 2000, which was processed without a 10CFR50.59 evaluation based on condition report evaluation KAP 97-0885. Condition Report KAP 97-0885 was written in May 1997 when the licensee discovered that the actual conical sump screen size was 1/8 inch by 15/32 inch which conflicted with the USAR Section 6.2, which stated that the screens were sized to prevent particles larger than 1/8 inch from entering the sump.

The inspectors determined the evaluation for Condition Report KAP 97-0885 erroneously concluded that the current screen design met the intent of the USAR statement and therefore a change to the USAR was warranted for clarification. The conclusion was based, in part, on internal correspondence from November 1973 from a licensee contractor to WPS Corporation which stated the response to AEC Question 6.19 was, “The screens installed over the containment sumps, which provide a source of suction for the residual heat removal pumps, are of a conical shape, manufactured of Johnson Screen material, which will admit particles having a mean diameter of 1/8 inch or smaller.” The 1997 condition report evaluation failed to recognize that the AEC, based on WPS Corporation’s December 1971 response, reviewed and approved a sump screen which was sized to prevent any particles larger than 1/8 inch from entering the sump.

The inspectors identified the sump screen discrepancies to the licensee. The licensee initiated a condition report to address the issue and performed an operability evaluation, based on current ECCS recirculation performance characteristics (including flow restrictions) which concluded the sump screens were operable but nonconforming, in accordance with Generic Letter 91-18.

Analysis:

The inspectors determined that the failure to promptly correct this condition adverse to quality was a licensee performance deficiency warranting a significance evaluation. This issue was more than minor because the issue affected the Mitigating System cornerstone attributes of design control for initial design and equipment performance reliability and affected the associated cornerstone objective to ensure the reliability and capability of systems that responded to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening and determined that the finding was of very low safety significance because it was not a design or qualification deficiency that had been confirmed to result in a loss of function per Generic Letter 91-18. The inspectors confirmed this through review and verification of the licensee's operability determination which concluded the containment sump screens were nonconforming per Generic Letter 91-18.

The inspectors also concluded that the primary cause of this finding was related to the cross-cutting area of problem identification and resolution, specifically the performance characteristic of corrective actions.

Enforcement:

10 CFR 50, Appendix B, Criterion XVI," Corrective Action," required, in part, that measures be established to assure that conditions adverse to quality, such as deficiencies, deviations, and nonconformances were promptly corrected. Contrary to this, the inspectors identified that conditions adverse to quality related to the sump screen openings were not promptly corrected. Therefore, the inspectors determined that this finding was a violation of 10 CFR 50, Appendix B, Criterion XVI. Because this violation was of very low safety significance (Green) and documented in the licensees corrective action program as CAP 023621 and CAP 023771, this finding was being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000305/2004009-09)

The licensee took immediate corrective actions which included performing an operability determination to determine if there were any immediate operability issues associated with the larger screen size. In addition, the licensee was taking long term corrective actions which would evaluate this issue in conjunction with the resolution of Generic Safety Issue 191 and NRC Generic Letter 2004-02.

b.2 Inadequate Instructions and Procedures for Inspections and Cleaning of the Safety-Related Containment Sump

Introduction:

A finding of very low safety significance (Green) was identified by the inspectors for a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, And Drawings," regarding licensee instructions and procedures for containment sump inspections. Specifically, the inspectors identified that current licensee procedures did not require inspection or cleaning when boric acid or small debris might be present in the containment sump; the licensee's procedures for containment coatings did not require inspection of the coating located inside the containment sump and had not been inspected since initial application; and the licensee's procedure for containment sump gap inspections did not specify acceptance criteria to ensure this activity was satisfactorily accomplished.

Description:

While performing an inspection of the containment sump, the inspectors noted that there appeared to be standing water with very small debris and residual boric acid residue in the safety-related containment sump directly below the sump screens. The inspectors noted that the licensee's outage schedule did not include an activity for routine cleaning of the safety-related containment sump.

The inspectors determined that during refueling outages prior to 2001, the containment sump was inspected and cleaned; however, a revision was made to the preventive maintenance activity instruction PM34-037 in 2001, to only clean the sump if external screen damage was verified. The inspectors questioned the licensee on the adequacy of this condition, in light of industry operating experience regarding boric acid accumulations in containment sumps and the residual boric acid currently located in the sump. The licensee initiated CAP 023679 and concluded that the safety-related sump required cleaning during the current refueling outage.

Following the cleaning of the containment sump the inspectors entered the containment sump as part of the inspection. The inspectors noted that the containment sump was concrete and had a thin clear coating (approximately 1-2 mils thick) which was later determined to be Carboline 1340. The inspectors identified that residual boric acid remained in certain sections of the sump which prohibited inspection of the containment sump coating in those areas. The inspectors questioned the licensee regarding the types of coating inspections performed in the containment sump and noted that General Nuclear Procedure (GNP), GNP-08.22.03, "Containment Walkdown to Monitor the Performance of Service Level I Coatings," listed all the safety-related areas with coatings in containment, except the containment sump. The licensee subsequently determined that the coating was applied approximately 10 years prior and that a coating inspection had never been performed since the original application of the coating. Based on the inspectors questions, CAP 023840 was initiated and subsequent cleaning of the remaining boric acid was performed. The licensee then performed a coating inspection and identified some missing coating under the two containment sump suction intakes; however, the remaining coating was intact.

The inspectors then verified the licensee's procedure for inspection of the containment sump screens, performed under GNP-12.17.01, Step 6.1.4 which required, an operator to verify that there were no breaches of integrity in the Containment Sump B conical screens and base plate attachments. The inspectors questioned the licensee whether the acceptance criteria would ensure that the containment sump recirculation function was maintained. The licensee initiated CAP 023816 and determined that the acceptance criteria was not explicit enough to ensure satisfactory completion of the activity, and additional clarifications were added to procedure GNP-12.17.01.

Analysis:

The inspectors determined that the failure to assure that inspections of the containment sump and screens were prescribed by instructions or procedures appropriate to the circumstances and containing appropriate acceptance criteria was a performance deficiency warranting a significance evaluation. This finding was more than minor because if left uncorrected the finding would become a more significant safety concern and the issue affected the Mitigating System cornerstone attributes of equipment performance reliability and procedure quality and affected the associated cornerstone objective to ensure the reliability and capability of systems that responded to initiating events to prevent undesirable consequences. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening and determined that the finding was of very low safety significance because it was not a design or qualification deficiency that had been confirmed to result in a loss of function per Generic Letter 91-18.

Enforcement:

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, And Drawings," required, in part, that activities affecting quality be prescribed by documented instructions, or procedures, of a type appropriate to the circumstances and shall include appropriate quantitative or qualitative acceptance criteria. Contrary to this, inspections of the containment sump and sump screens, activities affecting quality, were not prescribed by documented instructions, procedures or drawings of a type appropriate to the circumstances with appropriate acceptance criteria. Specifically, GNP-08.22.03, "Containment Walkdown to Monitor the Performance of Service Level I Coatings," was not appropriate to the circumstances, in that, the procedure did not require routine inspections of the coatings used on the internal portion of the safety-related containment sump. Preventive maintenance activity PM34-037, was not appropriate to the circumstances, in that, containment sump cleaning was not required if boric acid or debris was located in the containment sump. General Nuclear Procedure GNP-12.17.01 did not contain appropriate acceptance criteria for determining that important activities had been satisfactorily accomplished. The inspectors determined that this finding was a violation of 10 CFR 50 Appendix B, Criterion V. Because this violation was of very low safety significance (Green) and documented in the licensee's corrective action program as CAP 023679, CAP 023840 and CAP 023816, this finding was being treated as an NCV, consistent with Section VI of the NRC Enforcement Policy.
(NCV 05000305/2004009-10)

The licensee subsequently initiated several corrective actions to address these issues which included, but were not limited to:

- inspection and cleaning of the safety-related containment sump;
- inspection and assessment of the safety-related sump concrete coating;
- revision of preventive maintenance activity PM34-037 to require inspection and cleaning of the safety-related containment sump every refueling outage;
- revision of GNP-08.22.03 to include inspection of the safety-related containment sump concrete coating every refueling outage; and
- revision of GNP-12.17.01 to include appropriate acceptance criteria for determining that important activities were satisfactorily accomplished.

.3 Replacement Reactor Vessel Closure Head (RVCH) Fabrication (IP 71007)

a. Inspection Scope

The original RVCH penetrations nozzles were fabricated from Inconel Alloy 600 material. These nozzles were welded to the RVCH with a partial penetration weld fabricated from Inconel Alloy 182 weld filler metal. In recent years, several pressurized water reactors have experienced pressure boundary leakage caused by primary water stress corrosion cracking (PWSCC) of these materials.

During the 2004 refueling outage, the licensee elected to replace the RVCH and CRDM housings. The design of the replacement RVCH is similar to the original RVCH, with some notable exceptions as follows:

- the new RVCH is constructed from a single piece forging which eliminates the dome-to-flange weld;
- the new CRDM housing design eliminates vents and seal welds;
- the new RVCH design eliminates the spare and part length control rod penetrations; and
- the use of Inconel Alloy 600 was prohibited in fabrication of the new RVCH; for example, the RVCH penetration tube material was changed from Inconel Alloy 600 to Inconel Alloy 690 which is more resistant to PWSCC.

From August 9, 2004, through August 13, 2004, and from October 18, 2004, through October 28, 2004, the inspectors performed an on-site review of fabrication and preservice nondestructive examination (NDE) records related to fabrication of the replacement RVCH in accordance with Section 02.03 and Step 02.05.e of IP 71007, "Reactor Vessel Head Replacement Inspection." This review was performed to confirm that the manufacture and fabrication of the vessel head was completed in accordance with Section III of the ASME Code, 1998 Edition through 2000 Addenda. Specifically, the inspectors reviewed:

- contract and Code specifications for materials used in the head forging, and vessel head penetration nozzles and copies of heat treatment records including plots of furnace temperature verses time and related documentation that demonstrated the required temperatures and times were achieved to meet the material specifications;

- fabrication process sheets, fabrication drawings, and NDE records to verify that this manufacturing process control plan included provisions for NDE in accordance with applicable Code requirements;

- fabrication process sheets, fabrication drawings and welding procedures to ensure an appropriate sequence of welding operations and procedures existed to support cladding the inside of the reactor vessel head with stainless steel to meet Code requirements, design specifications and drawings;

- certified material test reports for materials used in fabrication of the reactor vessel head including weld materials to ensure Code material specifications were met;

- Nuclear Management Company surveillance audit records of the head fabricator and subcontractors associated with welding activities (welding of J-groove welds, head adaptor welds and head cladding), NDE activities, part identification/ traceability and drawing controls to confirm that these activities had been properly controlled in accordance with the contract specifications or Code requirements; and

- deviation notices, subcontractor corrective action notices and Nuclear Management Company communication issue resolution sheets to ensure that fabrication related deviations were appropriately tracked, evaluated and resolved.

b. Findings

No findings of significance were identified.

.4 RVCH and CRDM Housing Replacement (71007)

a. Inspection Scope

From October 18, 2004, through October 22, 2004, and November 30, 2004, through December 3, 2004, the inspectors reviewed the licensee's design changes associated with the replacement of the RVCH and CRDM housings.

The inspectors reviewed replacement RVCH and CRDM housing certified design specifications, certified design reports, American Society of Mechanical Engineers (ASME) Code reconciliation reports, fabrication deviation notices, non-conformance reports, and design calculations to confirm that the replacement RVCH and CRDM housings were in compliance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (1998 Edition including addenda through 2000 Addendum). Specifically, the inspectors confirmed that the design specifications and design reports for the replacement RVCH and CRDM housings were certified by registered professional engineers competent in ASME Code requirements. The inspectors confirmed that adequate documentation existed to demonstrate the certifying registered professional engineers were qualified in accordance with the requirements of the ASME Code Section III (Appendix XXIII of Section III Appendices). The inspectors also confirmed that the replacement RVCH and CRDM housings were provided as Code NPT stamped components.

b. <u>Findings</u>

<u>Introduction</u>:

The inspectors identified an unresolved item (URI) for potential non-compliance with the ASME Code design requirements governing the attachment of RVCH nozzles with partial penetration welds.

<u>Description</u>:

Partial penetration welds may be used to attach nozzles to the RVCH as permitted by the ASME Code Section III, Paragraph NB-3337.3. For this joint design, Paragraph NB-3337.3(b) allows the stress intensities resulting from pressure induced strains (dilation of hole) to be treated as secondary provided that the requirements of NB-3352.4(d), “Attachment of Nozzles Using Partial Penetration Welds,” and figure NB-4244(d)-1, “Partial Penetration Nozzle, Branch, and Piping Connections,” are fulfilled. In Design Calculation CN-RCDA-03-120, “CRDM Head Adapter - ASME Code Evaluation,” Section 6.3.4, the licensee evaluated stresses in the J-groove weld region resulting from pressure induced strains as secondary. However, the inspectors identified that the licensee’s RVCH design may have deviated from the requirements of NB 3352.4(d) and Figure NB-4244(d)-1.

For the attachment of nozzles using partial penetration welds, Section III Paragraph NB-3352.4(d)(2) specifies that the minimum dimensions of Figure NB-4244(d)-1 shall be met. In part, the corners of the end of each nozzle shall be rounded to a minimum radius of one-fourth of the nominal thickness of the penetrating part, or 3/4 inch, whichever is less. In addition, NB-3352.4(d)(3) specifies that the corners of the end of each nozzle, extending less than $(dt_n)^{0.5}$ (where d is the outside diameter and t_n is the nominal thickness of the penetrating part) beyond the inner surface of the part penetrated, shall be rounded to a minimum radius of one-half of the nominal thickness of the penetrating part, or 3/4 inch, whichever is less.

The inspectors identified the following discrepancies with respect to these requirements:

- The vent nozzles were ground flush with the inner surface of the RVCH. As such, the inside corner should have been rounded using a minimum ½ t_n (0.126 inch) radius in accordance with NB-3352.4(d)(3). However, as indicated on drawing L5-01DE109, the actual installed minimum radius was only 0.062 inch or approximately 1/4 t_n.

- The head adapter nozzles have a 4 inch outside diameter and 0.625 inch nominal wall thickness. All corners were rounded with a minimum 0.177 inch radius which is greater than 1/4 t_n but less than ½ t_n. Therefore, in accordance with NB-3352.4(d)(3), these nozzles should extend not less than $(dt_n)^{0.5}$ (1.5811 inch) beyond the inner surface of the part penetrated. The inspectors defined the inner surface of the part penetrated to be the J-groove weld toe. As indicated on drawing L5-01DE173, the actual extension dimensions (column L_6)

measured at nozzle location Nos. 28, 31, and 33 were less than the 1.5811 inch requirement. Therefore, the minimum corner radius should have been ½ t_n (0.3125 inch) at these locations in accordance with NB-3352.4(d)(3).

- The threads of the bottom of the instrumentation port head adapter tubes were removed by machining which resulted in an outside diameter step change. The measured distance from the J-groove weld toe to the diameter step change at these locations was less than the 1.5811 inch cutoff specified by NB-3352.4(d)(3). As such, the diameter step change corners should have been rounded using ½ t_n minimum radii. In addition, the corners at the bottom of the instrumentation port head adapter tubes should have been rounded using a minimum 1/4 t_n radius in accordance with NB-3352.4(d)(2). Instead, as shown on drawing L5-01DE111, two nozzle corner edges were chamfered between 0.005 inch and 0.03 inch, and the inside bottom corner edge was beveled at 30 degrees.

The inspectors judged that these potentially non-conforming conditions did not represent a degraded condition which would affect operability of the new RVCH. However, the inspectors considered these potential deviations from the design Code to be an unresolved item (URI 05000305/2004009-04) pending further review by the licensee to determine their position on application of these Code requirements. The licensee has entered this issue into their corrective action system (CAP 024611).

.5 Activities Associated With Reactor Vessel Head Replacement (IP71007)

a. Inspection Scope

The inspectors reviewed design and construction of Reactor Vessel Head (RV Head) lifting and rigging equipment used to transport the new RV head along the ground, through the containment equipment hatch, and into position in containment. In addition, the inspectors directly observed rigging activities associated with all phases of the new RV head being placed in containment. Crane and rigging equipment testing documents and procedures for rigging the new RV head into position in the containment were reviewed for adequacy. The inspectors directly observed the new RV head being placed into position on the reactor vessel.

The inspectors observed preparations for setting the new RV Head onto the reactor vessel. These preparations included:

- RCS draindown to 6" below the Reactor Vessel flange;
- Decontamination of the refueling cavity;
- Foreign material exclusion controls utilized for the reactor cavity work;
- Preparation of the New RV head, including installation of CRDM Coils and ARPI coil stacks; and
- Attachment and testing of rigging used to lift the New RV Head into position on the vessel;

The inspectors also observed post-installation testing of the new RV Head including:

- The licensee's testing program and results;
- Inspection of test records from CRDM and ARPI coil installation; and
- Inspection for RV Head leakage at plant normal operating temperature and pressure.

b. Findings

No findings of significance were identified.

.6 Review of Institute of Nuclear Power Operations Report

The inspectors completed a review of the Institute of Nuclear Power Operations, April 2004 Evaluation and Assistance Report for the Kewaunee Nuclear Power Plant, received by the licensee in October 2004.

4OA6 Meetings

.1 Exit Meeting

On December 17, 2004, the resident inspectors presented the inspection results to Mr. T. Coutu and other members of licensee management, who acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meeting

Interim exit meetings were conducted for:

- TI 2515/152, and the ISI procedure (IP 71111.08) inspections with Mr. T. Coutu on October 22, 2004.
- The reactor vessel head replacement fabrication review (IP 71007) with Mr. T. Coutu and other Members of your staff on October 28, 2004, and the reactor vessel head replacement safety evaluation and design reviews (IP 71007) on December 3, 2004, and December 17, 2004.

- Occupational Radiation Safety Access Control, ALARA and limited portions of the transportation and radioactive material control programs during and immediately following the licensee's extended refueling and reactor head replacement outage with Mr. T. Coutu on October 15, 22 and December 17, 2004.

4OA7 Licensee-Identified Violations

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

- Technical Specifications 3.3.a.1.B required that prior to exceeding 1000 psig RCS Pressure that the SI System Accumulator Isolation Motor Operated Valves (MOVs) be opened with power to the MOVs "locked out". Contrary to this requirement, the licensee exceeded 1000 psig RCS Pressure with the SI System Accumulator Isolation Valves still closed. This violation of Plant TSs was of low safety significance since no actual condition existed that required the Accumulators to be functional to mitigate an event or accident condition. This was documented in licensee's CAP as CAP 024241.

- 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality be prescribed by document instructions or procedures, of the type appropriate to the circumstances and shall include appropriate acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this requirement, the licensee failed to ensure that Procedure FPP-08-09, associated with the plants control room emergency zone envelope barrier control program was appropriate to the circumstances and included sufficiently detailed guidance to ensure all control room barriers were in their required positions. This violation was of low safety significance due to the very short duration in which the situation associated with this finding existed. This was documented in licensee's CA program as CAP 022205, ACE 002735, Maintenance Rule Evaluation 002409, and RCE 000658.

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
R. Adams, ALARA Supervisor
L. Armstrong, Engineering Director
S. Baker, Radiation Protection Manager
J. Bennett, EP Instructor
A. Bolyen, QA Supervisor
J. Coleman, EP Manager
J. Egdorf, EP Supervisor
D. Fitzwater, Operations Training Supervisor
W. Flint, Chemistry Manager
D. Franson, Service Water System Engineer
S. Forsha, Quality Assurance Oversight Lead NMC Head Replacement
L. Gerner, Licensing Supervisor
E. Gilson, Security Manager
W. Goder, Operations Training General Supervisor
G. Harrington, Licensing
W. Hunt, Training Manager
D. Lohman, Operations Manager
K. Peveler, Manager, Engineering Programs
J. Pollock, Design Engineering Manager
B. Presl, NMC Security Consultant
S. Putman, Maintenance Manager
A. Rahn, SW and FAC Inspection Program Engineer
R. Repshas, Site Services Manager
J. Riste, Licensing Supervisor
J. Rozell, Simulator Support Team
D. Scherwinski, Training Instructor
T. Schmidli, Radiation Protection General Supervisor, Field Operations
J. Stafford, Assistant Operations Manager
J. Rozell, Simulator Support Team
J. Stoeger, Operations Training Supervisor
D. Scherwinski, Training Instructor
P. Sunderland, EP Coordinator
C. Tomes, Fleet Lead NMC Engineer Head Replacement
S. Zepplin, Simulator Support Team

NRC Personnel

T. Kozak, Team Leader, Technical Support Section
J. Cameron, Project Engineer
J. Lamb, Project Manager
S. Reynolds, Acting Director, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000305/2004009-01	NCV	Inadequate Control of Combustible Materials (Section 1R05.1.b.1)
05000305/2004009-02	NCV	Inadequate Corrective Action to Preclude Storage of Oxygen Cylinders Next to Flammable Gas Cylinders (Section 1R05.1.b.2)
05000305/2004009-03	URI	Potential Flooding in the Turbine Building Basement (Section 1R06.2.b)
05000305/2004009-04	URI	Potential Non-compliance with ASME Code Governing the Attachment of RVCH Nozzles with Partial Penetration Welds (Section 1R17.2.b)
05000305/2004009-05	NCV	Scaffolding Erected Too Close to Safety-Related Equipment Required To be Operable (Section 1R20.1.b.1)
05000305/2004009-06	AV	Inability to Close Containment Equipment Hatch (Section 1R20.1.b.2)
05000305/2004009-07	NCV	Reactor Building Ventilation Isolation Function Not Available When Required (Section 1R20.1.b.3)
05000305/2004009-08	NCV	Failure to Identify Inadequate Pre-Fire Strategies (Section (4OA2.3.b)
05000305/2004009-09	NCV	Non-conforming Condition on the Safety-Related Containment Sump (Section 4OA5.2.c.1)
05000305/2004009-10	NCV	Inadequate Instructions and Procedures for Inspections and Cleaning of the Safety-related Containment Sump (Section 4OA5.2.c.2)

Closed

05000305/2004009-01	NCV	Inadequate Control of Combustible Materials (Section 1R05.1.b.1)
05000305/2004009-02	NCV	Inadequate Corrective Action to Preclude Storage of Oxygen Cylinders Next to Flammable Gas Cylinders (Section 1R05.1.b.2)
05000305/2004009-05	NCV	Scaffolding Erected Too Close to Safety-Related Equipment Required To be Operable (Section 1R20.1.b.1)

05000305/2004009-07	NCV	Reactor Building Ventilation Isolation Function Not Available When Required (Section 1R20.1.b.3)
05000305/2004009-08	NCV	Failure to Identify Inadequate Pre-Fire Strategies (Section (4OA2.3)
05000305/2004009-09	NCV	Non-conforming Condition on the Safety-Related Containment Sump (Section 4OA5.2.c.1)
05000305/2004009-10	NCV	Inadequate Instructions and Procedures for Inspections and Cleaning of the Safety-related Containment Sump (Section 4OA5.2.c.2)

Discussed

05000305/2004009-03	URI	Potential Flooding in the Turbine Building Basement (Section 1R06.2.b)
05000305/2004009-04	URI	Potential Non-compliance with ASME Code Governing the Attachment of RVCH Nozzles with Partial Penetration Welds (Section 1R17.2.b)
05000305/2004009-06	AV	Inability to Close Containment Equipment Hatch (Section 1R20.1.b.2)

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.

1R01 Adverse Weather Protection

Procedure GNP-12.06.01; Cold Weather Operations; Revision B
Procedure A-AAC-15; Abnormal Auxiliary Building Air Conditioning System Operation; Revision E
Procedure A-TAV-16; Abnormal Turbine Building and Screenhouse Ventilation System Operation; Revision P
Procedure PMP-08-07; FP - Hydrant Discharge House Test and House Station and Floor Drain Inspection; Revision W
CAP 013674; PB Level A issue - PBNP Facility not prepared for Cold Weather on 1 November 2002; Sept. 26, 2003
CAP 018586; Adverse Weather Protection activities have not been timely; Oct. 23, 2003

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

DCR 3481; Reactor Vessel Head Replacement Project; Revision 0
50.59 Applicability Review; DCR 3481; dated August 16, 2004
50.59 Pre-Screening; DCR 3481; dated August 16, 2004
SCRN No. 04-103; 10 CFR 50.59 Screening for DCR 3481; Revision 0
50.59 Applicability Review; DCR 3481 - Vendor (Westinghouse) Supporting Calculations; dated November 3, 2004
50.59 Pre-Screening; DCR 3481 - Vendor (Westinghouse) Supporting Calculations; dated November 10, 2004
Westinghouse Letter LTR-RCPL-04-145; Revision 1; Subject: Review of RRVCH and CRDM Reference Documents for USAR-Related Methods of Evaluation; dated November 15, 2004
50.59 Applicability Review; DCR 3481 - Vendor (Bigge Power Constructors) Supporting Calculations; dated September 13, 2004
50.59 Pre-Screening; DCR 3481 - Vendor (Bigge Power Constructors) Supporting Calculations; dated August 13, 2004
Procedure GNP-04.04.01; 50.59 Applicability Review and Pre-Screening; Revision C
Procedure GNP-04.04.02; 50.59 Screening and Evaluation; Revision C

1R04 Equipment Alignment

N-FW-05B-CL; Auxiliary Feedwater System Pre-startup Checklist; Revision AI
OPERM-205; Flow Diagram Feedwater System; Revision AX
N-SI-33-CL; SI System Prestartup Checklist, Revision AG
N-FW-05B-CL; Auxiliary Feedwater System Pre-startup Checklist; Revision AI

OPERM-205; Flow Diagram Feedwater System; Revision AX
N-SI-33-CL; SI System Prestartup Checklist; Revision AG
OPERXK-100-2B; Flow Diagram SI System; Revision AM
OPERXK–100-29; Flow Diagram SI System; Revision AA

1R05 Fire Protection

Fire Protection Program Analysis; Revision 5
Fire Protection Program Plan; Revision 5
Operational Quality Assurance Program Description; Revision 22.a
FPP-08-08; FP - Control of Transient Combustible Materials; Revision D

1R06 Flood Protection Measures

USAR Section 2.6; Hydrology; Revision 18.
Letter from WPS to NRC; Letter No. NRC-98-102; Response to Supplemental Request for Additional Information Regarding Individual Plant Examination for External Events Submittal; September 28, 1998
Procedure E-0-5; Response to Natural Events; Revision K
CAP 003858; OEA 2001-082 - Temporary Flood Barriers Not Installed Following Removal of; April 11, 2002
CAP 003187; Flooding Issue Screenhouse; February 20, 2002
CAP 002050; OEA 2001-082; May 31, 2001
CAP 008836; OEA 2001-061 - Flooding; May 10, 2001
CAP 013154; Potential Screenhouse Flooding Paths; Oct. 1, 2002
Letter from Pioneer Service & Engineering Co. to WPS.; Letter No. KP-S-2351; Check List Item 9 Draft - Screenhouse High Water Protection; May 2, 1972

1R07 Heat Sink Performance

PMP-10-11; DGM - Diesel Generator Cooling Water Heat Exchanger Performance Monitoring (QA-1); Revision C; November 20, 2003
GMP-137; Brush/Tube Scrubber Cleaning Heat Exchanger Tubes and Inspection; Revision H; July 29, 2004

1R08 Inservice Inspection Activities

SP-06-258; Main Steam and Auxiliary Feedwater System Pressure Test; Revision G
SP-36-267; ASME Boiler and Pressure Vessel Code Class I System Pressure Test; Revision 0

1R11 Licensed Operator Requalification

LRC-04-DY501; Cycle 04-05 Simulator Dynamic Scenario
Differences List Between Simulator and the Reference Plant

1R12 Maintenance Effectiveness

GNP-08.20.04; Maintenance Rule MRFF and MPFF Evaluations; Rev. E
NAP-08.20; Maintenance Rule Implementation; Revision. D
Residual Heat Removal Unavailability Hours; April 2003 to September 2004
CAP 016291; RHR 11 Operability Following Actuator Replacement
CAP 021806; RHR 299A Failed to Open During SP-33-098A
MRE 1830; RHR 110 Bushing Fell off While Attempting to Open; 4/15/2003
MRE 1968; RHR System Leak Near RHR-500B; 4/5/2003
MRE 1597; RHR A Pressure Xmtr Out of Tolerance Low; 9/17/2002
OPERXK-100-18; Residual Heat Removal System; Revision AQ
OPERXK-100-29; SI System; Revision AA
WO 04-05792; RHR HX Outlet Loop Hdr Temp has failed low; 5/17/2004
WO 04-06887; TM-627A removed - Repair module; 6/1/2004

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

GNP 08.04.01; Shutdown Safety Assessment, Revision L
Shutdown Safety Assessment Checklists, Control Room Logs, and Integrated Work Schedule For The Week Of October 11, 2004
Shutdown Safety Assessment Checklists, Control Room Logs, and Integrated Work Schedule For The Week Of October 18, 2004
Shutdown Safety Assessment Checklists, Control Room Logs, and Integrated Work Schedule For The Week Of October 25, 2004
Shutdown Safety Assessment Checklists, Control Room Logs, and Integrated Work Schedule For The Week Of November 1, 2004

1R14 Personnel Performance During Non-Routine Plant Evolutions

CAP 022946; Containment Sump A High Level Received; September 30, 2004
CAP 022983; Shroud Cooling Coil(s) Suspected Source of Leakage to Containment Sump A; October 3, 2004
CAP 022984; Increased Indication on Containment Particular Monitor R-11; October 3, 2004
GNP 12.17.02; Containment Inspection During Operations; Revision C; August 26, 2004
Operational Decision-Making Exercise; Increased in-leakage into Containment Sump A

1R15 Operability Evaluations

CAP 023009; Turbine Driven AFW Pump OB Bearing Oil Level Above Normal Level
NMAC TR-1007461; Terry Turbine Maintenance Guide
GNP-11.08.03; Operability Determinations
CAP 023009; Turbine Driven AFW Pump OB Bearing Oil Level Above Normal Level
NMAC TR-1007461; Terry Turbine Maintenance Guide
GNP-11.08.03; Operability Determinations
CAP 023695; Shroud Cooling Coil Damaged During Installation; October 30, 2004
CAP 023333; FHA Control Room Boundary Damper Analysis Assumptions; October 17, 2004

CAP 023124; As Found ACC System Flows Not Per Design Values; October 9, 2004
OBD 000100; As Found ACC System Flows Not Per Design Values; October 12, 2004
OPR 000076; As Found ACC System Flows Not Per Design Values; October 10, 2004

1R16 Operator Work-Arounds

Operator Workaround Status Sheet Dated October 25, 2004
Operator Workaround 04-08
Operator Workaround 04-07
Operator Workaround 04-06
Operator Workaround 04-04
Operator Workaround 04-03
Operator Workaround 04-02
NAD-12.07; Operator Workaround Rev B

1R17 Permanent Plant Modifications

LTR-RCUMP-04-61; Kewaunee Support Pin Design Equivalency Report; 11/10/04
Design Change Request 3494; Revision 1; 50.50 Applicability Review; 9/13/04
QF-0525 (FP-E-MOD-06); Revision 0
Final Guide Tube Replacement Support Pin Design Specifications and Supporting Documents; 4/30/04
DCR-3481; Reactor Vessel Head Replacement Project; Revision 0
Design Specification No. 414A85; Control Rod Drive Mechanism (CRDM) Model L106A; Revision 2
Document No. KW-KCS-04-0006; Design Report KW-KCS-04-0002; Revision 0 Addendum; Revision 3
Document No. KW-KCS-04-0002; Control Rod Drive Mechanism Design Report; Revision 0
WCAP-16238-P; Kewaunee Nuclear Power Plant Replacement Control Rod Drive Mechanism - Design Report; Revision 1
Addendum 1 to WCAP-16238-P; Revision 1; Kewaunee Nuclear Power Plant Replacement Control Rod Drive Mechanism - Design Report; dated June 2004
Calculation Note No. WB-CN-ENG-04-4; Kewaunee CRDM ASME Code Section XI Reconciliation; Revision 1
Calculation Note No. WB-CN-ENG-03-79; Kewaunee CRDM - Upper Latch Housing and Lower Latch Housing (LLH) - ASME Qualification; Revision 1
Calculation Note No. WB-CN-ENG-04-19; Kewaunee CRDM - Rod Travel Housing (RTH) ASME Qualification; Revision 0
Document No. KW-KCS-04-0004; Justification for Nonconformance Reports of Replacement Control Rod Drive Mechanism; Revision 2
Document No. KW-KCS-04-0008; Additional Reconciliation of Design Report with Latest Drawing Revision; Revision 0
Surveillance SP-49-074A; Control Rod Drop Time Test - Startup Measurements; dated November 23, 2004
MHI-NMC-1630K; L5-03BJ040, Revision 2 - CRDM As-built Dimension Drawing; dated June 14, 2004
NMC Letter KE-RRVCH-04-0290; Subject: CRDM As-built Dimensional Drawing, L5-03BJ040, Revision 2, MHI-NMC-1630; dated June 22, 2004

WEC Letter LTR-RCDA-04-596; Subject: Document Submittals MHI-NMC-1456K, MHI-NMC-1457K, MHI-NMC-1458K, MHI-NMC-1593K, and MHI-NMC-1630K; dated June 17, 2004
MHI Drawing L5-03BJ040; Control Rod Drive Mechanism, As-built Dimensional Drawing; Revision 2
Reactor Vessel Closure Head and Control Rod Drive Mechanisms; ASME NPT Component Certification; Mitsubishi Heavy Industries, Ltd.; dated June 14, 2004
DCR-3481; Reactor Vessel Head Replacement Project; Revision 0
Design Specification No. 414A84; Replacement Reactor Vessel Closure Head (RRVCH), (ASME B&PV Code, Section III, Class 1, Subsection NB); Revision 3
Design Specification No. 418A75; Addendum to Design Specification 414A84; Revision 3,
Replacement Reactor Vessel Closure Head (RRVCH), (ASME B&PV Code, Section III, Class 1, Subsection NB); Revision 0
Document No. L5-01DE510; Kewaunee Nuclear Power Plant, Replacement Reactor Vessel Closure Head Design Report; Revision 1
Document No. L5-01DE511; Kewaunee Nuclear Power Plant, Replacement Reactor Vessel Closure Head, Design Report L5-01DE510 Revision 1 Addendum; Revision 1
WCAP-16237-P; Kewaunee Nuclear Power Plant Replacement Reactor Vessel Closure Head - Design Report; Revision 1
Addendum 1 to WCAP-16237-P; Revision 1; Kewaunee Nuclear Power Plant Replacement Reactor Vessel Closure Head - Design Report; dated June 2004
Calculation Note No. CN-RCDA-04-20; Kewaunee RRVCH, ASME Section XI Code Reconciliation; Revision 0
Calculation Note No. CN-RCDA-03-106; Kewaunee Nuclear Power Plant RVCH Analysis Procedure; Revision 3
Calculation Note No. CN-RCDA-03-120; NMC Kewaunee Replacement Reactor Vessel Closure Head, CRDM Head Adapter ASME Code Evaluation; Revision 0
Calculation Note No. CN-RCUWF-04-1; Kewaunee Replacement Head Project - Closure Head Flange ASME Code and Leakage Evaluation; Revision 1
Document No. KBS-20040284; Justification for Nonconformance Reports of Replacement Reactor Vessel Closure Head; Revision 2
Document No. KBS-20040336; Fracture Evaluation of Kewaunee RRVCH and Point Beach Unit 2 RRVCH; Revision 1
Deviation Notice No. 60749; UT Requirement for Head Forging; Revision 1
Deviation Notice No. 62421; Number of Samples for Vent Pipe/Height of CRDM Housing; Revision 1
NMC Letter KE-RRVCH-04-0372; Subject: NMC Review of WEC DN 62421 Revision 1, WPS-04-130, Deviation Notice; July 2, 2004
NMC Letter KE-RRVCH-04-0425; Subject: NMC Review of WEC Letter LTR-RCDA-04-803 Regarding DN 62421; Revision 1; August 16, 2004
WEC Letter LTR-RCDA-04-803; Subject: Response to NMC Letter KE-RRVCH-04-0372 Regarding DN 62421; dated July 27, 2004
Nikko Inspection Report No. 2033-01-13; Closure Head Forging, Archive Sample, Coupons C1 & C2; dated April 10, 2003
MHI-NMC-1456K, L5-01DE171; Revision 2 - As-built Drawing (1/3); dated June 11, 2004
MHI-NMC-1457K, L5-01DE172; Revision 3 - As-built Drawing (2/3); dated June 11, 2004

MHI-NMC-1458K, L5-01DE173; Revision 4 - As-built Drawing (3/3); dated June 11, 2004
NMC Letter KE-RRVCH-04-0285; Subject: As-built Dimensional Drawing (1/3), L5-01DE171, Revision 2, HI-NMC-1456; dated June 22, 2004
NMC Letter KE-RRVCH-04-0286; Subject: As-built Dimensional Drawing (2/3), L5-01DE172; Revision 3, MHI-NMC-1457; dated June 22, 2004
NMC Letter KE-RRVCH-04-0287; Subject: As-built Dimensional Drawing (3/3), L5-01DE173, Revision 4, MHI-NMC-1457; June 22, 2004
WEC Letter LTR-RCDA-04-596; Subject: Document Submittals MHI-NMC-1456K, MHI-NMC-1457K, MHI-NMC-1458K, MHI-NMC-1593K, and MHI-NMC-1630K; June 17, 2004
MHI Drawing L5-01DE109; Replacement Reactor Vessel Closure Head, Closure Head and Adapter Housing Assembly; Revision 4
MHI Drawing L5-01DE111; Replacement Reactor Vessel Closure Head, Instrumentation Port Head Adapter 2/2; Revision 2
MHI Drawing L5-01DE115; Replacement Reactor Vessel Closure Head, Spare CRDM Adapter; Revision 1
MHI Drawing L5-01DE171; Replacement Reactor Vessel Closure Head, As-built Drawing (RV Closure Head) 1/3; Revision 2
MHI Drawing L5-01DE172; Replacement Reactor Vessel Closure Head, As-built Drawing (RV Closure Head) 2/3; Revision 3
MHI Drawing L5-01DE173; Replacement Reactor Vessel Closure Head, As-built Drawing (RV Closure Head) 3/3; Revision 4
Reactor Vessel Closure Head and Control Rod Drive Mechanisms, ASME NPT Component Certification, Mitsubishi Heavy Industries, Ltd.; dated June 14, 2004

1R19 Post-Maintenance Testing

SP-10-211-1; Inspection of Diesel Generator B- Electrical
SP-10-211-2; Inspection of Diesel Generator B- Mechanical
SP-10-211-3; Inspection of Diesel Generator B- Component Retest
SP-42-047B; Diesel Generator B Operational Test
SP-34-339B; RHR Pump B Full Flow Test at Refueling Shutdown - IST
CMP-34-01; RHR - RHR Pump Overhaul

1R20 Refueling and Outage Activities

N-O-04; 35 percent To HSD Condition
N-O-05; Plant Cooldown From Hot Shutdown To Cold Shutdown Condition
N-RHR-34; Residual Heat Removal System Operation
N-TB-54; Turbine and Generator Operation
N-O-02; Plant Start Up From Hot Shutdown To 35 percent Power
MRS-SSP-1637; Replacement RV Head Field Installation
MRS- GEN-1148; CRDM and ARPI Coil Resistance and Insulation Resistance Testing
PMP-57-26; Reactor Building Polar Crane Mechanical Maintenance
Bigge Document No. 2100-P7; Procedure For Load Tests of Kewaunee and Ginna Upending/Downending Frames and Spreader Bar SB-224 As Lift Rigging In Vertical Position

MRS-SSP-1690; Kewaunee RRVH Procedure To Haul New Head To Containment and Upend
MRS-SSP-1678; Kewaunee RRVH Procedure for Off-load from Delivery Vehicle, Transfer to Bigge Transporter, Transfer to Assembly Site, and Remove MHI Container
MRS-SSP-1687; Kewaunee RRVH Procedure to Install and Remove Containment Building Runway
Bigge Mechanical Drawings Package For Job Number 2100 (Kewaunee Head Replacement)
Carpenter Rigging and Supply Company Certificate of Test Serial Number 44840-01
Carpenter Rigging and Supply Company Certificate of Test Serial Number 44897-1
GNP-08.22.03; Containment alkdown to Monitor the Performance of Service Level I Coatings; Revision A; May 4, 2004
GNP-08.04.01; Shutdown Safety Assessment; Revision K; March 9, 2004
Shutdown Safety Assessment Checklist; October 8, 2004
Tagout Tag List; Tagout Group: Refueling Outage 27; Tagout 50-11-CONT-00001-(001)
GNP 02.07.01; Refueling Operations - Logkeeping, Watchstanding, and Shift Turnover; Revision A; May 25, 2004
E-FH-53A; Dropped or Damaged Fuel Assembly; Revision D; August 17, 2001
E-FH-53B; Loss of Reactor Cavity Inventory During Fuel Movement; Revision D; February 19, 2004
FP-OP-COO-01; Conduct of Operations; Revision 1
Operations Department Instruction Book; Protected Equipment; Revision 6; October 11, 2004
RCE 000616; Damaged Rod Control Cluster Assembly (RCCA)
Shutdown Safety Assessment Checklist; October 12, 2004; Time 0600-1800
Shutdown Safety Assessment Checklist; October 10, 2004; Time 0600-1800
Shutdown Safety Assessment Checklist; October 10, 2004; Time 1800-0600
Shutdown Safety Assessment Checklist; October 11, 2004; Time 0600-1800
Shutdown Safety Assessment Checklist; October 11, 2004; Time 1800 -0600; Rev 1
N-0-01-CLE; Backseated Valves Checklist; Revision D; April 18, 2002

1R22 Surveillance Testing

SP-33-110; Diesel Generator Automatic Test
SP-56-078; Containment Isolation Trip Test
SP-33-191; SI Flow Test - IST; Revision V; August 26, 2004
SP-05B-283A; Motor Driven AFW Pump A Full Flow Test - IST; Revision H; September 30, 2004
SP-05B-283B; Motor Driven AFW Pump B Full Flow Test - IST; Revision H; September 30, 2004
CAP024309; Relay Chatter During SP-49-074A, Control Rod Drop Timing Test; November 29, 2004
SP-49-074A; Control Rod Drop Time Test - Startup Measurements; Revision S; November 29, 2004
SP-36-082; Reactor Coolant System Leak Rate Check; December 19 and December 20,2004

1R23 Temporary Plant Modifications

TCR 04-13; Raise the setpoint of SFP Temperature Switches 12007 and 12012
Engineering Change Notice (ECN)-04-13-01; Raise the setpoint of SFP Temperature Switches 12007 and 12012
CAP 023890; 50.59 not updated on TCR 04-13 (Spent) Fuel High Temperature Alarm; November 8, 2004

1EP6 Drill Evaluation

Emergency Preparedness Drill and Exercise Manual; 4th Quarter 2004 Drill

2OS1 Access Control to Radiologically Significant Areas

CAP 019414; Unqualified Individual Attempting to Fee Release Material from the Radiologically Controlled Area; dated January 5, 2004
CAP 019896; Barriers for Locked High Radiation Areas; dated February 9, 2004
CAP 020143; Unnecessary Dose Received for Plant Inspection; dated February 25, 2004
CAP 020885; Posting Practices; dated April 20, 2004
CAP 021140; Exposure Received During Quarterly Plant Inspection in Locked High Radiation Area; dated May 10, 2004
CAP 021464; Human Error Traps in Requirements for Issuance of Neutron Bubble Dosimetry; dated June 7, 2004
CAP 022816; Radiation Protection Department Missed an Inventory of the Locked High Radiation Keys; dated September 22, 2004
CAP 022966; Foreign Material in Spent Fuel Transfer Canal; dated October 1, 2004
CAP 023148; HP Technician Performed Initial Confined Space Survey Without Support Person; dated October 10, 2004
CAP 023247; Electronic Dose Alarm Received; dated October 13, 2004
CAP 023254; Inadequate ALARA Brief; dated October 14, 2004
LER 050-305/2004-002-0; TS Sections 4.13 (b) and (e) Requirement for Leak Tests of Sources Transferred from Storage for Use or to Another Licensee
KNPP HP-01.019; Radiological Postings, Boundaries and Barricades; Revision F
KNPP HP-01.021; Issuance and Control of Locked High Radiation Area Keys; Revision C
RWP 7; General Decontamination and Support of Decontamination; Revision 0
RWP 11 RVCH Disassembly/Reassembly
RWP 15; NDE Testing; Revision 0
RWP 36; Transfer Canal Inspect and Decontamination; Revision 0
RWP 93; 626 Containment-Seal Table Area; Revision 0
RWP 98; Conoseal Work; Revision 0
RWP 115; 592 Containment Sump-C Sump Area; Revision 0
RWP 182; Reactor Coolant Pump Seal Work; Revision 0
RWP 200; General Clean-up and Decontamination of Containment; Revision 0
NAD-01.11; Dosimetry and Personnel Monitoring; Revision L
KNPP HP-01.016; Radiation Work Permit - Preparation, Issuance and Termination; Revision H

Personnel Contamination Outage Report for 2004 (undated draft)
KNPP HP-03.001; Shallow Dose Equivalent Calculation; Revision H
KNPP HP-03.008; Evaluation of Inhalations or Ingestions; Revision C
KNPP HP-03.009; Calculating Internal Dose from Whole Body Counter Results; Revision D
Selected Whole Body Count Results and Intake Dose Assessment Records for October 9, 2004 - December 12, 2004
CAP 023163; LHRA Entry Without Recorded Brief; dated October 11, 2004
CAP 024161; LHRA Key Issued in Violation of Procedure; dated November 22, 2004

2OS2 <u>As Low As Is Reasonably Achievable Planning And Controls</u>

ALARA Plan 04-005; Reactor Head Replacement Project ALARA Plan; dated September 29, 2004
ALARA Plan 04-006; Reactor Coolant Pump Work and Support; dated September 29, 2004
CAP 019748; Tag-out for Work Not Identified; dated January 28, 2004
CAP 019820; Poor Timing of Work for ALARA Considerations; dated February 2, 2004
CAP 020642; New Procedure Issue With No Prior Briefing or Training; dated April 1, 2004
CAP020664; Possible Violation of Newly Issued Procedure; dated April 2, 2004
KNPP HP-01.017; Self-Assessment of Radiation Protection Program; Revision D
KNPP HP-03.011; Special Dosimetry Issuance; Revision F
KNPP HP-04.006; Control and Use of HEPA Vacuums and Portable Air Filtration Units in Radiologically Controlled Areas; Revision B
KNPP HP-05.004; Radiation/Contamination Survey and Airborne Radioactivity Sampling Schedules; Revision Q
NAD-01.23; ALARA Program; Revision F
Historical Outage Exposure Performance Data (undated)
Exposure Performance Summary for all Outage RWPs for Various Periods Between October 9 and December 4, 2004
KNPP HP-02.003; Evaluation for Use and Issuance of Respiratory Protection Equipment; Revision G
KNPP HP-04.007; ALARA Plan Writers Guide; Revision A
KNPP HP-04-001; ALARA Plan; Revision G
NAD-08.03; Radiation Work Permit; Revision I
ALARA Plan 04-011 (dated September 24, 2004), associated Pre-Job ALARA Planning Checklist, ALARA Comment Sheet, and RWP 113 along with its Briefing Form; Bottom Mount Insulation Inspection and Replacement Plan
Minutes of November 4, 2004 Radiological Performance Committee; Bottom Mount Insulation Project Issues; dated November 15, 2004
Work In-Progress ALARA Reviews for ALARA Plan 04-011 and RWP 113; Bottom Mount Insulation Inspection and Replacement; dated October 17, November 4, 22 and 25, 2004
Daily Cumulative and Individual Worker Exposures for RWP 113; Bottom Mount Insulation Project
ALARA Plan 04-001 (dated September 29, 2004), associated Pre-Job ALARA Planning Checklist, ALARA Comment Sheet, and RWP 92 along with its Briefing Form; Refueling ALARA Plan

Work In-Progress ALARA Reviews for ALARA Plan 04-001 and RWP 92; Reactor Head Disassembly/Reassembly and Support; dated October 17 and November 25, 2004
ALARA Plan 04-014 (dated September 20, 2004), associated Pre-Job ALARA Planning Checklist, ALARA Comment Sheet, and RWP 103 and 106 along with its Briefing Forms; In-Service Inspection ALARA Plan
ALARA Plan 04-019 (dated September 20, 2004), associated Pre-Job ALARA Planning Checklist, RWP 12 along with its Briefing Form; Motor Operated Valve Maintenance and Testing Work Scope ALARA Plan
ALARA Plan 04-012 (dated September 29, 2004), Associated Pre-job ALARA Planning Checklist, ALARA Comment Sheet, and RWP 199 along with its Briefing Form; Scaffold and Support
CAP 023254; Inadequate ALARA Brief; dated October 14, 2004
CAP 023678; Inconsistent Expectations for Proper Attire in Containment; dated October 29, 2004
KSA - KIPP-04-01; Source Term Reduction Program Self-Assessment; dated February 24, 2004
KNPP HP-04.008; Hot Spot/Hot Line Tracking, Trending and Mitigation; Revision B
Steam Generator Loop Marker Survey Results for 1982 - 1993

2PS2 <u>Radioactive Material Processing and Transportation</u>

CAP 024203; Wrong Revision of Form Used for Radioactive Shipment; dated November 24, 2004 [Self-Revealed Issue based on NRC Questions]
EC-0230; Envirocare of Utah, Inc. Radioactive Waste Profile Record; dated September 1, 2004
EC-0230-SNM; Envirocare of Utah, Inc. Special Nuclear Material Exemption Certificate; dated September 1, 2004
EC-1800; Envirocare of Utah, Inc. Notice to Transport; dated September 28, 2004
ER-03-010; Duratek Engineering Report - Characterization of Kewaunee Nuclear Power Plant Reactor Pressure Vessel Head; dated March 5, 2004
E&L-037-04; Update of Characterization of Kewaunee RPVH; dated October 19, 2004
HP-09.031; Radioactive Material Shipping; Revisions A and B
Manifest 0845-08-0001; Uniform Low-Level Radioactive Waste Manifest, Shipping Paper, and Vehicle/Package Surveys for the Old RVCH (LSA-II), Shipped to Envirocare of Utah, Inc., Clive, UT; dated November 15, 2004
PL-DTK-04-002; Transportation and Emergency Response Plan - Kewaunee Reactor Head Disposal Project; dated June 28, 2004

2PS3 <u>Radioactive Material Control</u>

RPJG-40; RP Job Guideline - Reactor Head Replacement Project; dated September 2004
RWP 110; Remove Old RVCH from Containment to North Lot and Prepare for Off-Site Shipment; Revision 0

4OA1 PI Verification

NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 2;
LER 2003-002; Diesel Generator Failed Start Test Caused by Start Relay; Revision 0;
LER 2003-006; Component Cooling Water R-17 Radiation Detector Pipe Assembly Leakage; Revision 0;
LER 2004-008; Control Room Boundary Door Found Ajar; Revision 0
Various Dosimetry Egress Transactions, Personal Contamination Outage (Draft) Report, and Selected Intake Dose Assessments for the period mid-2003 through December 15, 2004
GNP-03.18.01; NRC Performance Indicators Reporting Instructions; Revision H
CAP Database Listing for Selected Keyword Searches for the period May 2003 - December 12, 2004

4OA2 Identification and Resolution of Problems

CAP 021901; Lack of Warnings or Training for Actions Needed if a Loss of Fire Water Occurs; dated July 20, 2004
CAP 021915; Hydrogen and Propane Gas Lines Are Not Identified in the Fire Strategies; dated July 21, 2004
PFP-17; Charging Pump, Boric Acid Concentrate Pump & Residual Heat Removal Pump Pit Areas; dated May 7, 2004
Fire Protection Program Analysis; Revision 5
Fire Protection Program Plan; Revision 5

4OA3 Event Followup

Licensee Event Report 2004-003; Control Room Boundary Door Found Ajar-Accident Analysis Assumptions Impacted
Licensee Event Report 2004-003-01; Supplemental Report to Control Room Boundary Door Found Ajar-Accident Analysis Assumptions Impacted
FPP-08-09; Fire Plan Procedure "Barrier Control"; Revision F
CAP 022205; Door 152 Control Room HVAC Elevator Door found open by NAO
ACE 002735; Door 152 Control Room HVAC Elevator Door found open by NAO
RCE 000658; Door 152 Control Room HVAC Elevator Door found open by NAO
CAP 021686; TS Surveillance Violation, T.S. 4.13. E or T.S. 4.13. F; June 25, 2004

4OA5 Other Activities

TI 2515/153

RFT012563; Sump Debris Bulletin 2003-01 - Option 2 Analysis
RFT013870; Develop and Implement Training on Sump Clogging by 12/19/03-NRC Commitment
Attendance Report by LP ID for LRC-03-SE601; Monday, October 25, 2004
Simulator Exercise Guide; SEG LRC-03-SE601; LB LOCA, Containment Sump Recirculation; October 8, 2003

CAP 023615; Potential Gap Between Training Conducted and NRC Commitment
Correspondence from KNPP to NRC; NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors”; 60-Day Response; August 7, 2003
Correspondence from KNPP to NRC; Supplement to 60-Day Response to NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors”; May 17, 2004
Correspondence from NRC to KNPP; Request for Additional Information Regarding Response to NRC Bulletin 2003-01 (TAC No. MB9584); September 7, 2004
KNPP Document Data Sheet; Containment Survey Screens; P.O. No. K-651; September 26, 1973
Drawing XK-651-1; Conical Screen; August 1, 1973
Drawing 237127A-S-237; Reactor Building Concrete-Sections and Details
Correspondence from Pioneer Service and Engineering Company to Wisconsin Public Service Corporation; KP-S-1966; Answers to AEC Questions #10
USAR Change Request B16-029; September 19, 2000
Correspondence from U.S. Atomic Energy Commission to Wisconsin Public Service Corporation; September 23, 1971
Correspondence from Wisconsin Public Service Corporation to U.S. Atomic Energy Commission; Amendment No. 13 to the Application for Construction Permit and Operating License for the KNPP; December 15, 1971
CAP 023816; Containment Sump Screen Inspection Criteria Unclear
CAP 023840; Containment Sump B Epoxy Coating
CAP 023679; Containment Sump B Requires Cleaning
Action Request Form; November 16, 2004; Void in Concrete Under East Inlet to 1A RHR Pump
KNPP Work Request Form; Sequence Number 209941; Apply Carboline 1340 to Concrete Floor; Ref. HP 8.05; Contact Rich Bardon; November 11, 1996
KNPP Work Request Form; Sequence Number 206649; Remove Remaining Coatings Found on Concrete and Steel Surfaces Inside Containment Sump B; May 12, 1995
DCR 2736; Removal of Containment Sump B Protective Coatings
Correspondence from Wisconsin Public Service Corporation to NRC; 120-Day Response to NRC Generic Letter 98-04; November 12, 1998
Incident Report 87-29; Identified Damage to Containment Sump B Floor Coating
CAP 023771; Potential Nonconformance Containment Sump B
KAP 471; Unqualified Coatings in Containment
CAP 013455; OEA 2003-230 - Potential ECCS Sump Blockage Due to Born Accumulation
CA 007256; Evaluate Need for Periodic Inspection of Containment Sump B and, if Necessary, Generate Work Instructions
CAP 017108; Bulletin 2003-01 Response Quantities of Containment Debris
CA 012689; Volume of Debris in Containment
CAP 006773; Potential Discrepancy was Discovered Between the Containment Sump B Design
CAP 023932; Missed Opportunity to Inspect Containment Sump B - Rework
RFT 012562; Sump Debris Bulletin 2003-01 – Option 2 Analysis
Attendance Report by LPID; BR03-109; Containment Sump Blockage (NRC Bulletin 2003-01); Monday, October 25, 2004

CAP 018297; Response to NRC Bulletin 2003-001: Impact of Debris Blockage Emergency Sump Recirculation

<u>Reactor Vessel Head Replacement</u>

Certified Material Test Reports:

Reactor Vessel Closure Head; dated April 11, 2003
Closure Cap; dated December 11, 2003
Instrument Port Head Adaptor; dated April 7, 2003
Vent Pipe; dated November 5, 2003
Latch Housings NKM806A,B,C,J; dated May 9, 2003
NX3167JK, WELTIG 52; dated November 18, 2003
NX2686JK, WELTIG S52; dated November 5, 2003
304372, WELAC 152; dated September 4, 2003
A3302N, LBL-96; dated May 30, 2003
A3301N, NC-38LK; dated May 30, 2003
A1071213N, NC-39LK; dated May 30, 2003
2L6712025, USB-308L; dated May 30, 2003
2L6612029, USB-309L; dated May 30, 2003
BF060331, ER308L; dated June 9, 2004
BHA0406, THS-316LK; dated November 5, 2003
BF36099, SATY-316LK; dated January 7, 2004
AH4218, DW-100; dated April 22, 2004

Communication Issues Resolution Sheets:

CIRS 03-088N; No Angle Beam Test for Latch Housing; dated September 11, 2003
CIRS 03-121N; 100 Percent DAC Exceeded on Latch Housing; dated October 24, 2003
CIRS 03-122N; Two 50 Percent DAC Indications on CRDM Head Adaptor; dated October 24, 2003
CIRS-03-125N; Vent Pipe Drawing Error; dated October 30, 2003
CIRS-03-127N; Straight Beam UT for Bi-Metallic Weld; dated October 30, 2004
CIRS-04-027N; Strip Cladding Weld Metal Certification; dated February 4, 2004
CIRS-04-068N; Weld Data Sheets for J-Groove Welds; dated March 17, 2004
CIRS-04-097M; Kewaunee CMTR Issues; dated May 26, 2004
CIRS-04-109N; Weld Filler Metal CMTRs; dated April 16, 2004

Deviation Notices:

DN 60679, Unsat UT on Keyway and Mating Surface; dated September 22, 2003
DN 60681, Unclear PT Procedure; dated August 27, 2003
DN 60684, Use of Demineralized Water; dated October 20, 2003
DN 60741; Repair Vent Pipe; dated December 4, 2003
DN 60749; 20 percent DAC UT of Head; dated March 11, 2004
DN 60751; Inadequate UT Procedure; dated March 2, 2004
DN 60846; Uncontrolled Welding and PT; dated April 30, 2004

Drawings:

JSW Drawing, N148737-1; Closure Head Forging Configuration at QT; Revision 2
JSW Drawing, N148737-M; Closure Head Forging Detail of Test Coupons; Revision 2
MHI Drawing, L5-01DE 201; 2-Loop Closure Head Forging; Revision 3
MHI Drawing, L5-01DE 202; Vent Pipe; Revision 1
MHI Drawing, L5-01DE 204; Instrument Port Head Adaptor Flange; Revision 1
MHI Drawing, L5-01DE 205; Spare CRDM Head Adaptor Flange; Revision 0
MHI Drawing, L5-01DE 206; Closure Cap; Revision 1
MHI Drawing, L5-01DE 001; Closure Head Outline Drawing; Revision 7
MHI Drawing, L5-01DE 002; Closure Head Outline Drawing; Revision 3
MHI Drawing, L5-01DE 101 &102; Closure Head General Assembly; Revision 5
MHI Drawing, L5-01DE 103; Closure Head Welding; Revision 1
MHI Drawing, L5-01DE 104, Closure Head Welding, Revision 1
MHI Drawing, L5-01DE 105 &106; Closure Head Machining; Revision 1
MHI Drawing, L5-01DE 107 & 108; Closure Head Penetration Position; Revision 1
MHI Drawing, L5-01DE 109 &110, Closure Head & Adaptor Housing Assembly, Revision 4
MHI Drawing, L5-01DE 111; Instrument Port Adaptor; Revision 2
MHI Drawing, L5-01DE 112; Vent Pipe; Revision 2
MHI Drawing, L5-01DE 114 &115; Spare CRDM Head Adaptor; Revision 2

Nuclear Management Company Surveillance Reports:

2003-0124; Review of Qualified Welders and Weld Operators, Review of Calibration Records for the Measuring Devices for Weld Overlay Cladding; dated June 6, 2003
2003-0132; Monitor Weld Overlay Cladding on Inner Surface of RVCH; dated June 20, 2003
2003-0163; Witness UT for Overlay Weld of Keyway; dated July 25, 2003
2003-0185; Witness Activities for the Kewaunee Nuclear Power Plant Head Replacement Project; dated August 21, 2003
2003-0188; Fit-Up Inspection of Butt Joint Between Latch Housing and Head Adapter; dated August 19, 2003
2003-0252; Remoter Visual Inspection for Inner Surface of Rod Travel Housings, UT for Seamless Stainless Steel Pipe for Vent Line; dated October 3, 2003.
2004-0020; PT for J-weld at MHI, Review Final PWHT Chart for RVCH; dated January 16, 2004
2004-0021; Witness PT for J-Groove Welds, Review of Welder Certification Records; dated January 23, 2004
2004-0048; PT for J-Weld of Head Adapter and Vent Pipe to RVCH, Monitoring of Welding for J-weld of Head Adaptor to RVCH; dated February 13, 2004
2004-0062; PT on Closure Cap after Machining, Monitoring of Welding of Butt Joint of Latch Housing to Rod Travel Housing; dated March 12, 2004
2004-0087; Monitored Welding Between Latch Housing and Rod Travel Housing; dated March 19, 2004
2004-0089; Manufacturing Process for CETNA Parts; dated February 4, 2004
2004-0093; Monitoring of Welding Operations for Closure Cap, UT of Butt Weld of Rod Travel Housing to Latch Housing, UT and ECT for Vent Pipe Penetration at MHI; dated April 9, 2004

2004-0102; UT for Butt Weld of Rod Travel Housing to Latch Housing; dated April 24, 2004
2004-0115; PT for Upper Surfaces of Flange and Stud Holes of RVCH, PT for J-welds and Alloy 690 Tubes, RT Films for Butt Welds of CRDM Housings, UT for Closure Cap, Hydro Pressure Test for RVCH; dated May 8, 2004

Nondestructive Examination Records:

Pressure Test Record- Reactor Vessel Closure Head; dated May 6, 2004
Magnetic Particle Examination Record, Exterior Surface of RVCH; dated May 25, 2004
Magnetic Particle Examination Record, Lift Lug Welds; dated May 24, 2004
Liquid Penetrant Examination Record, Vent Pipe Welds; dated May 24, 2004
Liquid Penetrant Examination Record, RVCH Cladding; dated May 25, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 25, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld 4 Inch Tube; dated May 24, 2004
Liquid Penetrant Examination Record, Indication J-Groove Weld Vent Pipe; dated May 24, 2004
Liquid Penetrant Examination Record, Welds WC-E109-1A, 8A,13A, 15A, 17A, 22A, 33A; dated May 29, 2004
Westinghouse Report - Kewaunee Unit 1 Replacement Vessel Head Inspection Final Report; dated June 17, 2004
J-Groove Weld Eddy Current Report Sheets; dated May 17-31, 2004
Radiographic Examination Records, Welds No. WC-J202-1A, 2A, 3A, 4A & 5A; dated October 29, 2003
Ultrasonic Examination Record, Weld No. WC-J202-1A; dated October 30, 2003
Ultrasonic Examination Record, Latch Housing NKM806A; dated March 18, 2003
Ultrasonic Examination Record, Latch Housing NKM806A; dated March 27, 2003
Ultrasonic Examination Record, Closure Head Forging; dated April 9, 2003
Magnetic Particle Examination Record, Closure Head Forging; dated March 26, & 27, 2003

Radiographic Film Records:

Radiograph KEN-CRDM-WC-E110-34A; Instrument Port Head Adaptor to Adaptor Flange Weld

Radiograph KEN-CRDM-WC-E114-9A; Spare CRDM Head Adapter to Extension Pipe Weld
Radiograph KEN-RVCH-WC-E116-9A; Closure Cap to Extension Pipe Weld
Radiograph KEN-RVCH-WC-J009-3A; Rod Travel Housing to Latch Housing Weld
Radiograph KEN-RVCH-WC-J009-28A; Rod Travel Housing to Latch Housing Weld
Radiograph KEN-CRDM-WC-J202-3A; CRDM Head Adaptor to Latch Housing Weld
Radiograph KEN-CRDM-WC-J202-28A; CRDM Head Adaptor to Latch Housing Weld

Other Documents:

MHI Specification No. L3-01AA409; Standard Material Purchase Specification for Head Adaptor Material (SB167 UNS N06690); Revision 4
Westinghouse Electric Co. Specification No. 676413; General Reactor Vessel Specification; Revision 1
MHI Purchase Specification No. KCE-20020111; Stainless Steel Forging for Pressure Vessel (ASME SA 182 Gr F316); Revision 2
Sumitomo Metal Industries LTD Certificates No’s. ONNC9498, ONNC9499, ONNC9503, ONNC9505, ONNC9506, ONNC9507; dated May 28, 2003
Reactor Vessel Head Design Specification No. 418A75; Revision 0; and No. 414A84; Revision 3
PO. No. P015276; Revisions 0 through 8
Calculation No. CN-RCDA-04-20; Nuclear Management Company Kewaunee RRVCH ASME Section XI Code Reconciliation; Revision 0.
Letter, from Westinghouse to Nuclear Management Co.; dated May 2, 2003
Manufacturing Specification No. –7474-10; Closure Head Forging; Revision 2
Heat Treatment Strip Chart for Reactor Vessel Closure Head; No. 03-518; dated March 13, 2003
Heat Treatment Strip Chart for Reactor Vessel Closure Head, No. 03-001; dated January 16, 2003
Heat Treatment Strip Chart for Reactor Vessel Closure Head; No. 03-569; dated March 24, 2003
2033-9; Record of Quenching and Tempering- Closure Head Archive Sample; dated March 14, 2003
2033-1-11; Record of Postweld Heat Treatment-Closure Head Archive Sample; dated March 25, 2003
2033-1.3; Record of Normalizing and tempering-Closure Head Archive Sample; dated January 7, 2003
MHI Document No. –7474-20; Quality Plan for Closure Head Forging; Revision 3
Reactor Head Replacement Project Guidelines-Fabrication Plan; dated June 29, 2004
Reactor Head Replacement Project Oversight Plan; Revision 2
Fabrication Process Sheets for Latch Housings Activities; April through July of 2003
Fabrication Process Sheets for CRDM Head Adaptor Activities; July through August of 2003
Fabrication Process Sheets for Spare CRDM Head Adaptor Flanges; July through August of 2003
Fabrication Process Sheets for Instrument Port Head Adaptor Flange Activities; July through September of 2003
Fabrication Process Sheets for the Reactor Vessel Closure Head Activities; July through October of 2003

Technical Specification - Design, Fabrication, and Installation of Replacement Reactor Vessel Closure Heads; Revision 4
Record of Dimensional Inspection and Visual Examination; dated April 8, 2003
Record of Markings Heat Number of Closure Forging 02D973-1-1Test Coupons; dated April 10, 2003

Subcontractor Corrective Action Notices:

MHI-03-008; dated August 4, 2003
MHI-03-010; dated August 22, 2003
MHI-03-022; dated September 24, 2003
MHI-03-023; dated September 26, 2003

Welding Procedures and Procedure Qualifications:

WPS Es0-3-5N; dated December 11, 2002
WPS Es0-3-6N; dated February 11, 2003
WPS A0-3-4N; dated December 11, 2002
WPS TO-3-4N; dated July 9, 2003
WPS A3.3-1N; dated January 16, 2003
WPS TA-3.43-11N; dated March 11, 2003
WPS TaTb-3.43-11N; dated February 3, 2004
PQR RE303V3; dated May 18, 1998
PQR RE303V4; dated May 18, 1998
PQR RE 03m1; dated December 6, 1991
PQR RE 03m2; dated December 6, 1991
PQR RE 03m3; dated November 7, 1991
PQR RT0m5; dated April 11, 1987
PQR RA0TaTa343R1; dated June 9, 2003
PQR RT 843m10; dated June 9, 2003

Weld Repair Records:

WC-E103-5, 2601-RVH-10A-R-1-R0-5; dated September 29,2003
WC-E109-3A/35A, 2601-RVH-10F-R4-39AC; dated May 20, 2004
WC-E109-35A, 2601-RVH-10F-R4-39AI; dated May 21, 2004
WC-E109-2, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-1A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-8A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-13A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-15A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-17A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-22A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004
WC-E109-33A, 2601-RVH-10F-R1-R0-9; dated May 29, 2004

Welder Qualification Records for Weld Repairs:

B285, Qualification Record TW-6h F6; dated June 6, 2000
B304, Qualification Record TW-6h F6; dated June 6, 2000
B320, Qualification Record TW-3r F-43a; dated October 27, 2003
B275, Qualification Record TW-3r F-43; dated August 29, 1997

Condition Reports Initiated for NRC Identified Issues

CAP 023477; Certified Design Specification 414A84 Revision 3 Inadequacy; dated October, 21, 2004
CAP 023040; Scaffolding too close to safety related equipment; dated October 6, 2004
CAP 023062; NRC Questions Shutdown Safety Assessment Orange Condition; October 7, 2004
CAP 023228, 023235; NRC Questions equipment protection guidance; October 13, 2004
CAP 023388; Oil in the Diesel Generator Room without permit and in excess of FPPA values; dated October 19, 2004
CAP 023418; Materials found above the Sprinkler Line in the Working Materials Storage Area; dated October 20, 2004
CAP 023428; Lube Oil improperly stored in 'A' EDG room; October 20, 2004
CAP 023478; Combustibles found stored on cabinet in AX-32; dated October 21, 2004
CAP 023479; Potential Hazards not identified in Fire Area Strategies; dated October 21, 2004
CAP 023480; Improper storage of combustible gas cylinders; dated October 21, 2004
CAP 023483; Inadequate Corrective Action to CAP 16329; dated October 22, 2004
CAP 023501; Verification of Fire Area Strategies to FPPA; dated October 23, 2004
CAP 024553; Flammable gas cylinders found stored in AX-23B; dated December 14, 2004
CAP 023512; SFP Pump A motor oil leak; October 23, 2004
CAP 023582; CAP 023274 Reportability basis statement is incorrect; October 26, 2004
CAP 023606; CAP apparently not written for an instrument failure;October 27, 2004
CAP 023621; SRI questions spacings in Emergency Recirc Sump; October 27, 2004
CAP 023679; Containment Sump B Requires Cleaning; October 29, 2004
CAP 023771; Potential Non-Conformance With Containment Sump B; November 2, 2004
CAP 023787; Cold Weather Operations Preps; November 3, 2004
CAP 023797; Flammable Materials Storage Cabinet Left Open; November 3, 2004
CAP 023814; Annulus Area Review For Adverse Weather; November 4, 2004
CAP 023816; Containment Sump Inspection Criteria Unclear; November 4, 2004
CAP 023839; Screws Missing On Diamond Plate In SW/CW Screen House; November 5, 2004
CAP 023840; Containment Sump B Epoxy Coating; November 5, 2004
CAP 023908; Containment Closure Problem Encountered; November 9, 2004
CAP 023950; Containment Hatch Closure Interference; November 11, 2004
CAP 024365; Turbine Building flooding concern with AFW trench; December 1, 2004

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AEC	Atomic Energy Commission
AR	Action Request
AFW	Auxiliary Feedwater
CA	Corrective Action
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
KNPP	Kewaunee Nuclear Power Plant
kV	kilovolt
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOR	Licensed Operator Requalification
NCV	Non-Cited Violation
NMC	Nuclear Management Company
NRC	Nuclear Regulatory Commission
PARS	Public Availability Records
PFP	Pre-Fire Plan
PI	Performance Indicator
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RO	Reactor Operator
ROP	Reactor Oversight Process
SAT	Systematic Approach to Training
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SI	Safety Injection
SRO	Senior Reactor Operator
STA	Shift Technical Advisor
SW	Service Water
TAC	Training Advisory Committee
TI	Temporary Instruction
TS	Technical Specification
TSC	Technical Support Center
TAC	Training Advisory Committee
TLD	Thermoluminescence Dosimeter
USAR	Updated Safety Analysis Report
VETIP	Vendor Technical Information Program

VCT	Volume Control Tank
WO	Work Order
WPS	Wisconsin Public Service