UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

October 28, 2005

Carolina Power and Light Company
ATTN: Mr. John Moyer
Vice President - Robinson Plant
H. B. Robinson Steam Electric Plant
Unit 2
3851 West Entrance Road
Hartsville, SC 29550

SUBJECT: H.B. ROBINSON STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT 05000261/2005004

Dear Mr. Moyer:

On September 30, the US Nuclear Regulatory Commission (NRC) completed an inspection at your H.B. Robinson reactor facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 12, 2005, with C. Church, E. Kapopoulos, J. Lucas and other members of your staff, and with you on October 27, 2005.

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 NRC-identified finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it had been entered into your corrective action program, the NRC is treating this issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Robinson 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/

Paul E. Fredrickson, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket No.: 50-261
License No.: DPR-23

Enclosure: Inspection Report 05000261/2005004
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-261
License No:	DPR-23
Report No:	05000261/2005004
Facility:	H. B. Robinson Steam Electric Plant, Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	July 1, 2005 - September 30, 2005
Inspectors:	R. Hagar, Senior Resident Inspector D. Jones, Resident Inspector M. Scott, Senior Reactor Inspector (Section 1R12.2) B. Crowley, Senior Reactor Inspector, Consultant (Sections 4OA5.3, 4OA5.4, & 4OA5.5) J. Rivera-Ortiz, Reactor Inspector (Section 4OA5.3, 4OA5.4, & 4OA5.5) R. Chou, Reactor Inspector (Section 4OA5.2) H. Gepford, Health Physicist (Section 2OS3)
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000261/2005-004, 07/01/2005-09/30/2005; H.B. Robinson Steam Electric Plant, Unit 2; Other Activities.

The report covered a three-month period of inspection by resident inspectors and announced inspections by two Senior Reactor Inspectors, two Reactor Inspectors, and a Health Physicist. One Green non-cited violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

Green. The inspectors identified a green non-cited violation of 10 CFR 50, Appendix B, Criterion V for two procedures which included instructions for restoring reactor coolant pump seal cooling but did not include any requirement or precaution regarding the time at which seal cooling is restored, even though information provided by the Westinghouse Owners’ Group indicated that restoration of RCP seal cooling was time-critical.

This finding was more than minor because it affected the procedure quality attribute of the Barrier Integrity Cornerstone objective of providing reasonable assurance that the reactor coolant system protects the public from radionuclide releases caused by accidents or events. The finding was evaluated using Appendix A to Manual Chapter 0609, Significance Determination Process. Because the finding affects a Barrier Integrity Cornerstone objective, the Phase 1 worksheet requires a Phase 3 risk evaluation be completed. A Phase 3 screening analysis was conducted and determined that because of the low likelihood of a station blackout, and the probable recovery of an offsite or onsite alternating-current power source prior to core damage, the finding was determined to be of very low safety significance. (Section 4OA5.6)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status The unit began the inspection period at full rated thermal power. On September 17, the unit was shut down for a refueling outage. That refueling outage extended through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns

The inspectors performed the following three partial system walkdowns, while the indicated structures, systems, and/or components (SSCs) were out-of-service for maintenance and testing:

System Walked Down	SSC Out of Service	Date Inspected
A emergency diesel generator	B emergency diesel generator	August 2
Service water train A	D service water pump	August 24
B emergency diesel generator	A emergency diesel generator	September 6

To evaluate the operability of the selected trains or systems under these conditions, the inspectors compared observed positions of valves, switches, and electrical power breakers to the procedures and drawings listed in the Attachment.

Complete System Walkdown

The inspectors conducted a detailed review of the alignment and condition of the B motor-driven train of the auxiliary feedwater system to verify that the existing alignment of the system was consistent with the correct alignment. To determine the correct system alignment, the inspectors reviewed the procedures, drawings, and the Updated Final Safety Analysis Report (UFSAR) section listed in the Attachment. The inspectors also walked down the system. During the walkdown, the inspectors reviewed the following:

- Valves were correctly positioned and did not exhibit leakage that would impact the functions of any given valve.
- Electrical power was available as required.
- Major system components were correctly labeled, lubricated, cooled, ventilated, etc.
- Hangers and supports were correctly installed and functional.

- Essential support systems were operational.
- Ancillary equipment or debris did not interfere with system performance.
- Tagging clearances were appropriate.
- Valves were locked as required by the locked valve program.

The inspectors reviewed the documents listed in the Attachment to verify that the ability of the system to perform its functions could not be affected by outstanding design issues, temporary modifications, operator workarounds, adverse conditions, and other system-related issues tracked by the engineering department.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

For the six areas identified below, the inspectors reviewed the control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures to verify that those items were consistent with UFSAR Section 9.5.1, Fire Protection System, and UFSAR Appendix 9.5.A, Fire Hazards Analysis. The inspectors walked down accessible portions of each area and reviewed results from related surveillance tests to verify that conditions in these areas were consistent with descriptions of the areas in the UFSAR. Documents reviewed are listed in the Attachment.

The following areas were inspected:

Fire Zone	Description
25A/B	Turbine building east and west ground floor
17	[Heating, ventilation, and air conditioning] equipment for control room
25F/G	Turbine building east/west mezzanine and operating deck
25D	Dedicated shutdown diesel generator
16	Battery room
19	Cable spreading room

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Internal Flooding

Because the safety injection pump room contains risk-significant SSCs which are susceptible to flooding from postulated pipe breaks, the inspectors walked down that room to verify that the area configuration, features, and equipment functions were consistent with the descriptions and assumptions used in Calculation RNP-F/PSA-0009, Assessment of Internally Initiated Flooding Events and in the supporting basis documents listed in the Attachment. The inspectors reviewed the operator actions credited in the analysis to verify that the desired results could be achieved using the plant procedures listed in the Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed-operator performance during requalification simulator training for crew 2 to verify that operator performance was consistent with expected operator performance, as described in the Continuing Training Simulator Option form dated 6/23/05. This training tested the operators' ability to respond to a loss of reactor coolant system inventory and a subsequent loss-of-coolant accident during shutdown conditions. The inspectors focused on clarity and formality of communication, the use of procedures, alarm response, control board manipulations, group dynamics, and supervisory oversight.

The inspectors observed the post-exercise critique to verify that the licensee identified deficiencies and discrepancies that occurred during the simulator training.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

.1 Routine Inspection of Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two degraded SSC/function performance problems or conditions listed below to verify the appropriate handling of these performance problems or conditions in accordance with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, and 10 CFR 50.65, Maintenance Rule. Documents reviewed are listed in the Attachment.

The problems/conditions and their corresponding action requests (ARs) were:

Performance Problem/Condition	AR
Containment isolation valve V12-11 failed to open	143554
Letdown orifice isolation valve regulator setpoint changed	135101

During the reviews, the inspectors focused on the following:

- Appropriate work practices.
- Identifying and addressing common cause failures.
- Scoping in accordance with 10 CFR 50.65(b).
- Characterizing reliability issues (performance).
- Charging unavailability (performance).
- Trending key parameters (condition monitoring).
- 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification.
- Appropriateness of performance criteria for SSCs/functions classified (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified (a)(1).

b. Findings

No findings of significance were identified.

.2 Periodic Evaluation (Biennial)

a. Inspection Scope

The inspectors reviewed the licensee's Maintenance Rule (MR) periodic assessment, Self-Assessment Report 147347, dates of assessment May 2-5, 2005. This report was issued to satisfy paragraph (a)(3) of 10 CFR 50.65, and covered the 18 month period of October 1, 2003 to March 31, 2005. The inspection was to determine the effectiveness of the assessment and that it was issued in accordance with the time requirement of the MR and included evaluation of: balancing reliability and unavailability, (a)(1) activities, (a)(2) activities, and use of industry operating experience. To verify compliance with 10 CFR 50.65, the inspectors reviewed selected MR activities covered by the assessment period for the following maintenance rule components and systems: radiation monitors, reactor protection system, Regulatory Guide 1.97 instrumentation, switchyard components, and auxiliary feedwater system. Additionally, the inspectors reviewed a section of the partially completed structural inspection report (containment building) and inspected select plant structures. Specific procedures and documents reviewed are listed in the Attachment.

During the inspection, the inspectors reviewed selected plant work order data, assessments, modifications, the site guidance implementing procedures, discussed and reviewed relevant corrective action issues, reviewed generic operations event data, attendant MR related meeting minutes, probabilistic risk reports, and discussed issues with system engineers. Operational event information was evaluated by the inspectors in its use in MR functions. The inspectors selected work orders and other corrective action documents on systems recently removed from 10 CFR 50.65 a(1) status and those in a(2) status for some period to assess the justification for their status. The inspectors toured and inspected repaired component locations. The documents were compared to the site's MR program criteria, and the MR a(1) evaluations and rule related data bases.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For the four time periods listed below, the inspectors reviewed risk assessments and related activities to verify that the licensee performed adequate risk assessments and implemented appropriate risk-management actions when required by 10 CFR 50.65(a)(4). For emergent work, the inspectors also verified that any increase in risk was promptly assessed, and that appropriate risk-management actions were promptly implemented. Documents reviewed are listed in the Attachment. Those periods included the following:

- The work week of July 30 - August 5, including emergent work which rendered the B emergency diesel generator unavailable
- The work week of August 6 - August 12, including emergent work which included unavailability of the steam-driven auxiliary feedwater pump
- The work week of August 13 - August 19, including emergent work which included unavailability of the C safety injection pump
- The work week of August 26 - September 2, including emergent work which included unavailability of the A emergency diesel generator

b. Findings

No findings of significance were identified.

Enclosure

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability determination associated with AR 164063. This AR addressed the operability of the B auxiliary feedwater pump when an associated flow indicating controller was degraded. The inspectors assessed the accuracy of the evaluation, the use and control of any necessary compensatory measures, and compliance with the Technical Specifications (TS). The inspectors verified that the operability determination was made as specified by procedure PLP-102, Operability Determinations. The inspectors compared the justifications provided in the determination to the requirements from the TS, the UFSAR, and associated design-basis documents, to verify that operability was properly justified and the auxiliary feedwater system remained available, such that no unrecognized increase in risk occurred.

Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

For the six post-maintenance tests listed below, the inspectors witnessed the test and/or reviewed the test data to verify that test results adequately demonstrated restoration of the affected safety functions described in the UFSAR and TS. Documents reviewed are listed in the Attachment.

The following tests were witnessed/reviewed:

Test Procedure	Title	Related Maintenance Activity	Date Inspected
OST-603	Motor Driven Fire Water and Engine Driven Fire Water Pump Test (Weekly)	Routine maintenance on the engine driven fire pump	July 11
OST-101-1	[Chemical and Volume Control System] Component Test, Charging Pump A	Repair the start/stop switch and calibrate the fluid drive oil pressure gauge	July 26

OST-401-2	[Emergency Diesel Generator] B Slow Speed Start	Replace air start solenoid valve, DA-19B	August 2
OST-302-2	Service Water Pumps C and D Inservice Test	Breaker replacement and electrical testing	August 24
OP-604	Diesel Generators A and B	Replace coil for air start solenoid valve, DA-19A	September 6
OST-636	Flow Test for [Reactor Coolant Pump] B Pre-Action Sprinkler System (Refueling)	Replace pressure switch PS-7008	September 22

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR 154987, Lead Deterioration Found On Motor Removed From [Service Water] Pump A
- AR 160935, [Post Maintenance Test] Requirement Not Identified

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

For the outage that began on September 17, the inspectors evaluated licensee outage activities as described below to verify that licensee considered risk in developing outage schedules, adhered to administrative risk reduction methodologies they developed to control plant configuration, and adhered to operating license and technical specification requirements that maintained defense-in-depth. The inspectors also verified that the licensee developed mitigation strategies for losses of the following key safety functions:

- decay heat removal
- inventory control
- power availability
- reactivity control
- containment

Documents reviewed are listed in the Attachment.

Review of Outage Plan

Prior to the outage, the inspectors reviewed the outage risk control plan to verify that the licensee had performed adequate risk assessments, and had implemented appropriate risk-management strategies when required by 10 CFR 50.65(a)(4).

Monitoring of Shutdown Activities

The inspectors observed portions of the plant shutdown and cooldown process to verify that technical specification cooldown restrictions were followed.

Licensee Control of Outage Activities

Periodically during the outage, the inspectors observed the items or activities described below to verify that the licensee maintained defense-in-depth commensurate with the outage risk-control plan for key safety functions and applicable technical specifications when taking equipment out of service.

- Clearance Activities
- Reactor Coolant System Instrumentation
- Electrical Power
- Decay Heat Removal (DHR)
- Inventory Control
- Reactivity Control
- Containment Closure

The inspectors also reviewed responses to emergent work and unexpected conditions to verify that resulting configuration changes were controlled in accordance with the outage risk control plan, and to verify that control-room operators were kept cognizant of the plant configuration.

Refueling Activities

The inspectors observed fuel handling operations (removal) and related activities to verify that those operations and activities were being performed in accordance with technical specifications and approved procedures. Also, the inspectors verified that the locations of the fuel assemblies were tracked during core offload.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the six surveillance tests listed below, the inspectors witnessed testing and/or reviewed the test data to verify that the systems, structures, and components involved in these tests satisfied the requirements described in the TS, the UFSAR, and applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions. Documents reviewed are listed in the Attachment.

Test Procedure	Title	Date Inspected
OST-251-1*	[Residual Heat Removal] Pump A and Components Test	July 14
OST-151-3	Safety Injection System Components Test - Pump C	July 28
OST-151-1	Safety Injection System Components Test - Pump A	August 3
OST-202	Steam Driven Auxiliary Feedwater System Component Test	August 8
OST-409-2	[Emergency Diesel Generator] B Fast Speed Start	August 16
OST-154	Safety Injection System High Head Check Valve Test	September 27

* This procedure included inservice testing requirements.

The inspectors reviewed AR 165893, Steam Driven [Auxiliary Feedwater] Pump Trip to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification described in Engineering Change 60359, Temporary Connection of [Radiation Monitor] 14 - Control Room Alarm Circuits During Engineering Change 52464, to verify that the modification did not affect the safety functions of important safety systems, and to verify that the modification satisfied the requirements of procedure EGR-NGGC-005, Engineering Change, and 10 CFR 50, Appendix B, Criterion III, Design Control. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope

Portable Radiation Monitoring Instrumentation

During the week of August 1, 2005, the inspectors evaluated completion and adequacy of radiation survey instrument calibrations performed by the licensee's central calibration facility located at the Shearon Harris Nuclear Plant. Availability of portable instruments for licensee use was evaluated through discussion with licensee personnel regarding inventory, logistics, and transfer/receipt of instruments. Calibration data for portable instruments staged or recently used for coverage of field tasks were reviewed. Records associated with the annual certifications of the gamma irradiator and neutron source used for performing calibrations and routine response checks were reviewed in detail. In addition, the inspectors observed the calibration facility for neutron instrument calibrations and discussed its adequacy for performing instrument calibrations with cognizant licensee personnel. The inspectors discussed techniques and technical bases applied to the calibration of portable survey instruments, including the use of a 25% grace period, with licensee personnel. Two corrective action program (CAP) nuclear condition documents associated with the instrument calibration activities were reviewed and discussed with responsible licensee representatives.

Operability, reliability, and calibration of selected radiation detection instruments were reviewed against 10 CFR Part 20; Final Safety Analysis Report (FSAR) Chapter 12; ANSI N323-1978, Radiation Protection Instrumentation Test and Calibration; and applicable licensee procedures. The licensee's ability to characterize, prioritize, and resolve identified CAP issues was reviewed against CAP-0200, Corrective Action Program, Rev. 14 and associated guideline documents. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Routine Review of ARs

To aid in the identification of repetitive equipment failures or specific human performance issues for followup, the inspectors performed frequent screenings of items entered into the CAP. The review was accomplished by reviewing daily AR reports. Documents reviewed are listed in the Attachment.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected AR 158738, [Self Assessment] 147347 Issue #1: Repetitive Functional Failures, for detailed review. The inspectors selected this AR because it relates generally to the Mitigating Systems Cornerstone, and involved failures of steam dump valves and spurious actuations of reactor protection system components. The inspectors reviewed this report to verify:

- complete and accurate identification of the problem in a timely manner;
- evaluation and disposition of performance issues;
- evaluation and disposition of operability and reportability issues;
- consideration of extent of condition, generic implications, common cause, and previous occurrences;
- appropriate classification and prioritization of the problem;
- identification of root and contributing causes of the problem;
- identification of corrective actions which were appropriately focused to correct the problem; and
- completion of corrective actions in a timely manner.

The inspectors also reviewed this AR to verify compliance with the requirements of the CAP as delineated in Procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B. Documents reviewed are listed in the Attachment.

b. Observations and Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Operation of an Independent Spent Fuel Storage Installation (ISFSI) (IP 60855.1)

a. Inspection Scope

During the first movement of spent fuel from the storage pool into the new ISFSI, the inspectors observed selected activities associated with inspecting spent fuel; record-keeping; loading spent fuel into storage canisters; and canister welding, vacuum-drying, transport, and insertion into the storage location, to verify that the activities were performed in a safe manner and in accordance with approved procedures. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Review of Reactor Vessel Closure Head Replacement Project (RVCHRP) Lifting and Transportation Program Activities

a. Inspection Scope

The inspectors reviewed the adequacy of the RVCHRP lifting program as described in Engineering Change Package 56266, Rev. 2, "Reactor Closure Head & Service Structure Replacement", to assure that it was prepared in accordance with regulatory requirements, appropriate industrial codes and standards, and to verify that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

The inspectors reviewed and partially examined the RVCHRP lifting equipment including a crane, a down/up-ender, a head lifting tripod and rigs, runway skid systems, load paths, and a transporter, for material condition and adequacy. In addition, the inspectors reviewed the adequacy of the transport programs, procedures, work packages and load test records, to assure that they had been prepared and tested in accordance with regulatory requirements, appropriate industrial codes, and standards. The inspectors also reviewed the adequacy of the licensee's analyses and calculations for handling loads during the rigging and lifting, the load path analysis, polar crane maintenance records, and lifting and rigging drawings.

b. Findings

No findings of significance were identified.

.3 Reactor Pressure Vessel Head (RPVH) Replacement (IP 71007)

a. Inspection Scope

The inspectors observed/reviewed the activities detailed below for the replacement RPVH to verify compliance with applicable construction and inspection Codes (ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda and Section XI, 1995 Edition through 1996 Addenda) as defined in the Engineering Change (EC) document EC 56266R1, “Reactor Head and Service Structure Replacement.”

RPVH and Control Rod Drive Mechanism (CRDM) Housing Fabrication Records

The inspectors reviewed fabrication records for the RPVH and the CRDM housings including Certified Material Test Reports, Non-Destructive Examination (NDE) reports, hydrostatic testing and dimensional examinations to verify compliance with the applicable construction and inspection codes. The records reviewed are listed in the Attachment. For the following welds, the inspectors reviewed: fabrication process sheets, welding work record sheets, fitup inspection records, PT examination records, RT examination records, and UT examination records, as applicable, to verify compliance with the applicable construction and inspection codes. Records reviewed for the selected welds are listed in the Attachment.

- RPVH J-groove Butter welds WO-S107-1A, 8A, 18A, 25A, and 35A
- RPVH J-groove welds WC-S109-1A, 6A, 16A, 28A, and 62A
- RPVH Clad welds WO-S103-1, and -2
- CRDM Rod Travel Housing to Latch Housing welds WC-L009-1A, 9A, 17A, 26A, 30A, and 38A
- CRDM Latch Housing to RPVH Adapter welds WC-L202-1A, 14A, 24A, 34A, 62A, and 65A

Preservice Inspection (PSI) and Baseline Inspections

The inspectors reviewed selected NDE records, which documented the ASME Section XI PSI and baseline inspections performed to provide baseline conditions for future inspections in accordance with NRC Order EA-03-09.

Relative to ASME Section XI PSI of the replacement RPVH, the inspectors reviewed the completed PT records of :

- 28 peripheral Category B-O CRDM Latch Housing to Rod travel Housing welds
- 28 peripheral Category B-O CRDM Latch Housing to RPVH Adapter welds

In addition, the inspectors reviewed personnel certification records of two (2) Level III NDE Examiners, and PT material certification records for the welds listed above.

In order to support future inspections required by NRC Order EA-03-09, the baseline NDE inspections consisted of:

- Automated open bore inside diameter UT and eddy current (ET) examination of the CRDM penetrations covering from the bottom of the penetrations to a minimum of 2" above the J-groove welds
- J-groove surface ET examination and inside diameter UT and ET examination of the Reactor Vessel Level Indication System (RVLIS) line and vent nozzle penetrations
- ET examination of the outside diameter of the penetrations below the J-groove welds and the surface of the J-groove welds
- Under head PT inspection of all penetration to head J-groove welds using “PT white” acceptance criteria (performed to obtain the best surface possible to prevent future primary stress corrosion cracking)

The inspectors observed in-process NDE baseline examinations and reviewed the results of a sample UT and ET data. Specifically, the inspectors conducted the following activities:

- Partial observation of open bore scanning UT/ET examination for penetration No. 62
- Partial observation of outside diameter ET examination of penetration No. 37
- Partial observation of ET examination on J-groove weld surface for penetration Nos. 29 and 44
- Review of completed UT and ET reports for penetration Nos. 7, 39, and 63, including saved computer data
- Review of automated UT and ET procedures, including equipment specifications
- Review of personnel qualifications for UT and ET examiners performing baseline inspections
- Review of under head PT examination reports of all penetration to head J-groove welds using “PT white” acceptance criteria, including personnel qualifications and PT material certifications
- Visual inspection on top of the RPVH, specifically penetration Nos. 62, 38, 30, 22, 14, and 26

b. Findings

No findings of significance were identified.

.4 Review of 10 CFR 50.59 Screening/Evaluation for the Replacement RPVH

a. Inspection Scope

The inspectors reviewed EC 56266, Reactor Head and Service Structure Replacement, Rev. 1, including the associated 10 CFR 50.59 screening to verify that changes between the original RPVH and the replacement RPVH, and modifications resulting from installation of the replacement RPVH were properly evaluated in accordance with 10 CFR 50.59. Specifically, the inspectors reviewed the impact of the replacement RPVH weight on the reactor vessel supports and seismic analysis. The inspectors verified that the weight of the replacement RPVH assembly, as described in EC 56266R1, was bounded within the safety margin of the design stress calculations for static and seismic loads.

b. Findings

No findings of significance were identified.

.5 Review of Quality Assurance (QA) Activities for the fabrication of the RPVH

a. Inspection Scope

The inspectors reviewed surveillance reports of licensee QA activities at the vendor facilities to verify that the licensee evaluated the QA activities performed by the fabricator (Mitsubishi Heavy Industries, MHI) and the contractor for RPVH fabrication (Westinghouse) at the MHI facilities. The inspectors performed a review to verify that the licensee conducted QA surveillance such as: (1) independent review of fabricator procedures and fabrication records, (2) independent verification of RPVH dimensions, (3) witnessing NDE examinations, material testing, and manufacturing processes, (4) verification of fabricator personnel knowledge through interviews, (5) verification that non-conformance conditions and deviations were identified and dispositioned, and (6) assessment of Westinghouse QA surveillance on MHI QA program.

b. Findings

No findings of significance were identified.

.6 (Closed) URI 05000261/2005003-01: Failure of Two Procedures to Have Appropriate Acceptance Criteria for Restoration of Reactor Coolant Pump (RCP) Seal Cooling

In NRC Inspection Report 05000261/2005003, a URI was identified involving two examples of a violation of 10 CFR 50, Appendix B, Criterion V, Procedures, for failure to include appropriate acceptance criteria in two procedures for restoration of cooling to the reactor coolant pump seals following a loss of all seal cooling. The finding was unresolved because it had potential safety significance greater than very low significance and required the completion of a significance determination process Phase 3 review, prior to finalizing the finding's significance.

As discussed in Section 1R17 of Inspection Report 05000261/2005003, the inspectors identified that Procedures EPP-22, Revision 20, Energizing Plant Equipment Using Dedicated Shutdown Diesel Generator, and DSP-002, Revision 30, Hot Shutdown Using the Dedicated/Alternate Shutdown System, both included instructions for restoring RCP seal cooling but did not include any requirement or precaution regarding the time at which RCP seal cooling is restored, even though information provided by the

Westinghouse Owners' Group indicated that restoration of RCP seal cooling was time-critical.

During this inspection period, a Phase 3 screening analysis was completed by using a bounding calculation to determine the potential limits of the risk associated with the finding. That analysis included consideration of results from licensee-completed thermal hydraulic calculations which determined the time to core uncovery following a loss-of-coolant accident from the RCP seals induced by restoring RCP seal cooling later than would be advisable. Because of the low likelihood of a station blackout, and the probable recovery of an offsite or onsite alternating-current power source prior to core damage, the finding was determined to be of very low safety significance (GREEN).

10 CFR 50, Appendix B, Criterion V, Procedures, requires, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances. It further requires that these procedures include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, Procedures EPP-22, Rev. 20 and DSP-002, Rev. 30, do not include appropriate acceptance criteria for restoration of cooling to the RCP seals following a loss of all seal cooling. However, because of the very low safety significance and because this issue was entered into the corrective action program (AR 160357), this finding is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy, and has been designated NCV 05000261/2005004-01, Failure of Two Procedures to Have Appropriate Acceptance Criteria for Restoration of RCP Seal Cooling. URI 05000261/2005003-01 is closed.

.7 Operational Readiness of Offsite Power (Temporary Instruction (TI) 2515/163)

Completion of this TI was documented in NRC Inspection Report 05000261/2005003. However, after an NRC headquarters review of the data provided, additional information related to the TI was requested. The inspectors collected this information from licensee discussions, site procedures and licensee documentation. The information was subsequently provided to the headquarters staff for further analysis.

4OA6 Meetings, Including Exit

On October 12, the resident inspectors presented the inspection results to C. Church, E. Kapopoulos, J. Lucas, and other members of the Robinson staff. In addition a supplemental exit was conducted with J. Moyer on October 27, 2005. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

M. Blew, Inservice Inspection Coordinator
E. Caba, Engineering Superintendent
C. Castell, Licensing
A. Cheatham, Radiation Protection Superintendent
C. Church, Engineering Manager
B. Clark, Nuclear Assurance Manager
R. Cline, Non-Destructive Examination Level III Examiner
D. Etheridge, Lead Engineer RPVH Replacement Project
W. Farmer, Engineering Superintendent
J. Huegel, Maintenance Manager
R. Ivey , Operations Manager
E. Kapopoulos, Outage and Scheduling Manager
J. Lucas, Manager, Support Services - Nuclear
G. Ludlum, Training Manager
J. Moyer, Vice President, Robinson Nuclear Plant
W. Noll, Director of Site Operations
D. Stoddard, Plant General Manager
V. Wagoner, Reactor Head Replacement Project Manager
S. Wheeler, Supervisor, Regulatory Support

Contractor Personnel

R. Driscoll, Westinghouse Principal Engineer
P. Lancaster, Westdyne Non-Destructive Examination Level III Examiner
R. Vestovich, Westdyne Project Manager

NRC personnel

P. Fredrickson, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Opened and Closed

05000261/2005004-01	NCV	Failure of Two Procedures to Have Appropriate Acceptance Criteria for Restoration of RCP Seal Cooling (Section 4OA5.6)

Closed

05000261/2005003-01	URI	Failure of Two Procedures to Have Appropriate Acceptance Criteria for Restoration of Reactor Coolant Pump (RCP) Seal Cooling(Section 4OA5.6)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Partial System Walkdown

A emergency diesel generator system
Procedure OP-604, Diesel Generators A & B, Revision 61

Service water train A
Procedure OP-903, Service Water System, Rev. 95
Clearance Order Checklist No. 93063, Motor Lead Inspection and [High] Potential Testing
Drawing G-190199 Service and Cooling Water System Flow Diagram, Sheet 2 of 13, Rev. 63
Drawing G-190199 Service and Cooling Water System Flow Diagram, Sheet 5 of 13, Rev. 44
Drawing G-190199 Service and Cooling Water System Flow Diagram, Sheet 6 of 13, Rev. 44
Drawing G-190199 Service and Cooling Water System Flow Diagram, Sheet 7 of 13, Rev. 38
Drawing G-190199 Service and Cooling Water System Flow Diagram, Sheet 9 of 13, Rev. 52
Drawing G-190199 Service and Cooling Water System Flow Diagram, Sheet 10 of 13, Rev. 43

B emergency diesel generator system
Procedure OP-604, Diesel Generators A & B, Revision 61

Complete System Walkdown

Procedure OP-402, Auxiliary Feedwater System, Rev. 62
System Description SD-042, Auxiliary Feedwater System, Rev. 10
Drawing G-190197, Feedwater Condensate and Air Evacuation System Flow Diagram, Sheet 4, Rev. 54
FSAR section10.4.8, Auxiliary Feedwater System
A list of ARs that involved Auxiliary Feedwater System components during 2004-2005
The system health report for Auxiliary Feedwater, dated March 22, 2005

Section 1R05: Fire Protection

Sections in UFSAR Appendix 9.5.1A, Fire Hazards Analysis
3.7.1, Fire Zone 25A - Turbine Building East Ground Floor
3.1.3.1, Fire Zone 25B - Turbine Building West Ground Floor
3.1.5.3, Fire Zone 17 - [Heating, Ventilation, and Air Conditioning] Equipment Room for Control Room
3.1.5.5, Fire Zone 19 - Unit 2 Cable Spreading Room
3.7.5, Fire Zone 25F - Turbine Building East Mezzanine
3.7.6, Fire Zone 25F - Turbine Building West Mezzanine
3.7.7, Fire Zone 25G - Turbine Building Operating Deck
3.7.4, Fire Zone 25D - Dedicated Shutdown Diesel Generator
3.1.5.2, Fire Zone 16 - Battery Room

Procedures
FP-012, Fire Protection Systems Minimum Equipment and Compensatory Actions, Rev. 9
OMM-003, Fire Protection Preplans/Unit No. 2, Rev. 44

Completed Procedures
results from OST-645, Turbine Lube Oil Deluge System Flow Test (Annual), Rev. 17
results from OST-611-10, Low Voltage Fire Detection and Actuation System Zones 16, 17, 18, 29, & 30 (Semi-Annual), Rev. 6
results from OST-611-11, Low Voltage Fire Detection and Actuation System Zones 19 & 20 (Semi-Annual), Rev. 4

Other documents
Drawing HBR2-8255, Fire Protection System Flow Diagram, Sheet 2 of 6, Rev. 27
Clearance Order Checklist No. 95228, Clean/Inspect S-105B, [Turbine Lube Oil Supply Strainer]

Section 1R06: Flood Protection Measures

UFSAR Sections
3.6A.6, Flooding Analysis
9.5.1.4.2, Fire Suppression Systems.

Calculations
RNP-F/PSA-0009, Assessment of Internally Initiated Flooding Events

Procedures
AOP-014, Component Cooling Water Malfunction, Rev. 21
AOP-022, Loss of Service Water, Rev. 28
AOP-032, Response to Flooding from the Fire Protection System, Rev. 35

Other Documents
AOP-014-BD, Basis Document Component Cooling Water Malfunction, Rev. 21
AOP-022-BD, Basis Document Loss of Service Water, Rev. 28
AOP-032-BD, Response to Flooding from the Fire Protection System, Rev. 35

Section 1R12: Maintenance Effectiveness

Action Requests
30516, pressurizer thermal insulation
30920, fault pressure relay
86712, radiation monitors in a(1) for 5 years
104566, N42 spiking
115823, spurious RPS problems
126165, “C” main transformer fault pressure relay failure
128014, [steam driven auxiliary feedwater] pump emitting sparks
133713, steam generator narrow range low level failure
135101, CVC-200C, letdown orifice isolation, regulator setpoint changed
139933, RC-525 leaking

143554, Containment isolation valve V12-11 failed to open 158738, steam dump solenoid coil failures
147649, [over-temperature delta-T]/[over-pressure delta-T] alarms
158734, unlogged functional failures

Work Orders
54312, CVC-200C: valve was not set up properly
333105, CVC-200C has a diaphragm leak
141188, CVC-200C failed to open
20010830, R-11 check source
20041214, R-20 zero flow
20050718, R-11 reading high
20050720, R-14 computer locked up
20050726, R-14 flow alarm

Procedures
ADM-NGGC-0101, Maintenance Rule Program, Rev. 18
PM-492, E0555 Air Regulator Maintenance, Rev. 0
CAP-NGGC-0200, Corrective Action Program, Rev. 16
CM-101, Quick Change Trim Air Operated Control Valve Maintenance, Rev. 21

Maintenance Rule Documents
For system 2060 (chemical and volume control system)
- Event Log Report for 1/1/03 - 8/22/05
- Scoping and Performance Criteria

For system 1000 (containment isolation valves):
- Event Log Report for 5/1/03 - 5/1/05
- Scoping and Performance Criteria

Expert panel meeting minutes excerpts on radiation monitors 1998 to present

Other Documents
System Health Report, Chemical and Volume Control System, January 26, 2005
Long Range Plan Project Control Report - By Selected System, for chemical and volume control system
System health reports on RPS, RM, HVAC, AFW, Dedicated Shutdown System
HBRSEP Unit 2, Table 3.3.3-1, Post Accident Monitoring Instrumentation, Amendment No. 176
LCO log for Technical Specification 3.3.3 (last two years)

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Procedure OMM-048, Work Coordination and Risk Assessment, Rev. 24
Procedure MST-913, Emergency Fire Pump Batteries (Quarterly Test), Rev. 11
[Robinson Nuclear Plant] Unit #2 Shift Logs, dated 8/31 - 9/1/2005

Section 1R15: Operability Evaluations

AR 164063, B [Auxiliary Feedwater] Pump Unavailability Time Extended
Design Basis Document, DBD/R87038/SD32 Auxiliary Feedwater
Drawing G-190197, Feedwater Condensate and Air Evacuation System, Sheet 4 of 4, Rev. 54
System Description SD-042, Auxiliary Feedwater System, Rev. 9
Technical Specification 3.3.8, Auxiliary Feedwater System Instrumentation

Section 1R19: Post Maintenance Testing

Procedures
OST-101-1, [Chemical and Volume Control System] Component Test Charging Pump A, Rev. 38
OST-603, Motor Driven Fire Water and Engine Driven Fire Water Pump Test (Weekly), Rev. 26
OST-401-2, [Emergency Diesel Generator] B Slow Speed Start, Rev. 28
OST-302-2, Service Water Pumps C and D Inservice Test, Rev. 36
OP-604, Diesel Generators A and B, Rev. 61
OST-636, Flow Test for [Reactor Coolant Pump] B Pre-Action Sprinkler System (Refueling), Rev. 18
MMM-006, Appendix B-13 Calibration Data Sheets, Rev. 6
PIC-301, Pressure Switches and Vacuum Switches, Rev. 7

Drawings
B-190628, Control Wiring Diagram, Sheet No. 950, Rev. 26
5379-1153, Electrical Schematic Diagram for Diesel Generator, Sheet 1 of 1, Rev. 26

Work Orders
633082-01, A Charging Pump Start/Stop Switch
678323-01, Diesel Fire Pump Universal Joint Looseness Check
678324-01, Diesel Fire Pump Semi-Annual Lubrication
741058, [Instrumentation and Control] Found Damaged Wiring to DA-19B
691868-01, [Service Water Pump] D Motor Lead Inspection
321582-01, Trip Testing of [Motor Control Center]-6(7C) 12D Service Water D
724267-01, [Service Water] D Pump [Motor] [Direct Current] Stepped Hi-[Potential]/Surge Testing
490193-03, 52/25B Replace Breaker 52/25B With Refurbished DB-50
752249-03, Repair/Return to Stock Solenoid Valve for DA-19A
562419-01, [Refueling Outage] 23 - Replace Pressure Switch ([Reactor Coolant Pump] B Sprinkler Low Flow Alarm)

Other
Clearance Order Checklist 96377, [Emergency Diesel Generator] B Air Start Solenoid Valve Replacement
Material Evaluation 01761, Valve - Solenoid, 1-1/2 Inch, Rev. 1
Material Evaluation 07027, Valve - Solenoid, 1-1/2 Inch, Rev. 0
AR 161546, Weakness #1 from Assessment R-OM-05-01
System Description SD-041, Fire Water System, Rev. 3

Section 1R20: Refueling and Outage Activities

Procedures
OMP-004, Outage Risk Assessment, Rev. 15
OMP-003, Shutdown Safety Function Guidelines, Rev. 23
OMM-046, Control of Key Safety Functions During Shutdown, Rev. 13
OMM-033, Implementation of CV Closure, Rev. 15
MMM-009, Operation, Testing and Inspection of Cranes and Material Handling Equipment, Rev. 50
GP-006, Normal Plant Shutdown From Power Operation to Hot Shutdown, Rev. 48
GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown, Rev. 65
FMP-019, Fuel and Insert Shuffle, Rev. 30
SP-1527, Low Pressure Draining, Inspection and Repair, and Refilling of the North Service Water Header, Rev. 1

Other
Refueling Outage-23 Outage Schedule
[Refueling Outage-23] Pre-Outage Risk Assessment, dated 8/5/2005
OST-410 Contingency Plan
[Spent Fuel Pool] Level and Temperature Monitoring Contingency Plan
IN 2005-16, Outage Planning and Scheduling - Impacts on Risk
Clearance Order Checklist 95337, R223-7110B 480 [Volt] Bus 3 and [Dedicated Shutdown] Bus
Results from Procedure GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown, Attachment 10.1, [Reactor Coolant System] and [Pressurizer] Cooldown Data Table
Clearance Order Checklist 94554, R223-4060B North Service Water Header
Clearance Order Checklist 93309, R223-5175B 480 [Volt] Bus E2
System Description, SD-51, Inadequate Core Cooling Monitor System, Rev. 3

Action Requests
AR 166245, [Self Assessment] 141805 W-1 RO-23 Schedule [Equipment Out Of Service] Flags
AR 166259, [Self Assessment] 141805 W-3 Risk Window and Code Problems
AR 166257, [Self Assessment] 141805W-2 [Refueling Outage]-23 Activities Not Meeting [Procedure] OMP-003 [requirements]
AR 162214, NRC IN-2005-16 Outage Planning and Scheduling

Section 1R22: Surveillance Testing

Procedures
OST-151-3, Safety Injection System Components Test - Pump C, Rev. 25
OST-202, Steam Driven Auxiliary Feedwater System Component Test, Rev. 62
OST-251-1, [Residual Heat Removal] Pump A and Components Test, Rev. 19
OST-252-1, [Residual Heat Removal] System Valve Test, Rev. 12
MMM-006 Appendix B-5, Calibration Data Sheets, Rev.8
TMM-004, Inservice Testing Program, Rev. 64
OST-151-1, Safety Injection System Components Test - Pump A, Rev. 24
OST-409-2, [Emergency Diesel Generator] B Fast Speed Start, Rev. 24
GP-009-1, Filling the Refueling Cavity with Fuel in the Reactor Vessel, Rev. 5
PPP-104, SI-870A and SI-870B [Motor Operator Valve] [Differential Pressure Test], Rev. 7

OST-154, Safety Injection System High Head Check Valve Test, Rev. 37

Drawings
5379-1082, Safety Injection System Flow Diagram, Sheet 1 of 5, Rev. 42
5379-1082, Safety Injection System Flow Diagram, Sheet 2 of 5, Rev. 45
5379-1082, Safety Injection System Flow Diagram, Sheet 3 of 5, Rev. 25
5379-1082, Safety Injection System Flow Diagram, Sheet 4 of 5, Rev. 28

Other
AR 165893, Steam Driven [Auxiliary Feedwater] Pump Trip
Work Order 399443-01, Calibrate the [Residual Heat Removal] Pump A Miniflow Recirculation Flow Indicator, FI-608A
[Condition Report] 93-15019, OST-051 and OST-010 Violate OMM-040 Readability Requirements
Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment
RNP-L/LR-0602, Aging Management Program, Open Cycle Cooling Water System Program
Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance
Design Basis Document, DBD/R87038/SD02, Safety Injection System, Rev. 0

Section 1R23: Temporary Plant Modifications

Engineering Change 60359, Temporary Connection of [Radiation Monitor] 14 Control Room Alarm Circuits During Engineering Change 52464
Procedure EMP-013, Operation of R-14 and F-14, Rev. 29
Work Order 729619-01, Implement [Temporary Modification] [Engineering Change] 60359 for R-14

Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment

Procedures and Technical Documents
ERC-114, Control of Radiation Instruments and Equipment, Rev. 6
HPS-0005, Calibration of Portable Radiation and Contamination, Rev. 5
HPS-0009, Operation of Radiation & Contamination Survey Instruments, Rev. 2
HPS-0011, Cs-137 Calibration Source Standardization, Rev. 2
SIC-700, Operation and Certification of Calibration Standards, Rev. 9
Radiation Protection Technical Note 04-001, Use of a “Grace Period” for Calibrations, Rev. 0,

Action Requests
127084, Instrument Source Check, 5/15/04
67211, Calibration Performed with Source not Traceable to NBS, 7/26/02

Data and Records Reviewed
Certificates of Calibration for Calibration Sources: 03-021B, 00-068, 00-072B, 00-072A, 86-001, 98-003, 99-052
Neutron Calibration Source Certification Data Sheet, 7/11/05
Shepherd Model 89 Recertification Spreadsheet, 2/22/05

Section 4OA2: Identification and Resolution of Problems

AR 158738, [Self Assessment] 147347 Issue #1: Repetitive Functional Failures
CAP-NGGC-0200, Corrective Action Program, Rev. 16
ADM-NGGC-0101, Maintenance Rule Program, Rev. 18
Maintenance Rule Scoping and Performance Criteria for System 1080 (Reactor Protection and Safeguards System)

Section 4OA5: Other Activities

Procedures
ISFS-011, 24P-ISFSI Transfer Cask and Dry Shielded Canister Preparation for Loading, Rev. 2
ISFS-012, 24P-ISFSI Transfer Cask Handling Operations for Fuel Loading, Rev. 3
ISFS-013, 24P-ISFSI Dry Shielded Canister Fuel Loading, Rev. 2
ISFS-012, 24P-ISFSI [Dry Shielded Canister] Sealing Operations, Rev. 4
Progress Energy Manual MMM-009, Operation, Testing, and Inspection of Cranes and Material Handling Equipment, Rev. 50
Progress Energy Manual PM-132, Containment Polar Gantry Crane Semiannual at Hot or Cold Shutdown, Rev. 11
Bigge Procedure 2085-P3, Procedure to Downend Old Head in Containment, Transfer Out of Containment, Tie Down, Haul and Move to Mausoleum, Rev. 1
Bigge Procedure 2085-P4, Procedure to Haul New Head to Containment, Transfer into Containment, Upend and Set on Runway, Rev. 1.
Bigge Procedure 2085-5 (including test data), Procedure for Component Load Test Qualification, Rev. 0
Westinghouse Procedure MRS-DFD-1685-RVHPIS-DP, Component Handling Equipment - Load Test Procedure for the H. B. Robinson Reactor Vessel Head Tripod Lift Rig Assembly - Drawing 10010D26, Rev. 0

Other Documents
Engineering Change (EC) Package 56266 & 50.59 Evaluation, Reactor Head & Service Structure Replacement, Rev. 2
CP&L Response, Dated May 19, 1981, to an Unnumbered Generic Letter- Dated December 22, 1980, Control of Heavy Loads at Nuclear Power Plants
CP&L Response, Dated December 15, 1982 to NUREG-0612, Control of Heavy Loads
NRC letter, Dated May 29, 1984, to CP&L for Control of Heavy Loads - Phase I
CP&L Response, Dated May 13, 1996, to Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment
Westinghouse Calculation CN-RVHP-04-93, H. B. Robinson Unit 2 Reactor Vessel Head Lift Rig NUREG 0612 Stress Calculation, Rev. 2
Robinson Nuclear Generation Group Calculation RNP-C/STRU-1199, Containment Runway System Analysis, Rev.2
Bigge Calculation 2085-C1.1, Crane and Rigging for Lifting / Upending of MHI Package, Rev. 2
Bigge Calculation 2085-C1.2, Crane and Rigging for Lifting / Downend New RVCH Assembly, Rev. 1
Bigge Calculation 2085-C5.1, Lifting New / Old Heads with Spreader and Rigging Trusses, Rev. 1

Bigge Calculation 2085-C7.1, Bigge Transport Configuration, Pull Force, & Wheel Loads for New / Old Heads & MHI Package, Rev. 1
Bigge Calculation 2085-C7.2, Tiedown System New / Old Head, Rev. 1
Bigge Calculation 2085-C8.1, Qualification / Test of Upender / Downender, Rev. 1
Whiting Corporation Evaluation for CP&L Robinson 155 Ton Polar Crane, Dated October 12, 1982
Bigge ENGR. No. Drawing No. 002, Sheets 1 to 6, Installation, Storage, & Removal of A-Frame, Rev.1
Bigge ENGR. No. Drawing No. 004, Sheets 1 to 14, Downend Old RVCH From Containment, Rev.1
Bigge ENGR. No. Drawing No. 005, Sheets 1 to 13, Upend New RVCH Into Containment, Rev.1
Bigge ENGR. No. Drawing No. 021, Sheets 1 to 4, Upend New RVCH at Head Assembly Building, Rev.1
Bigge ENGR. No. Drawing No. 022, Sheets 1 to 3, Downend New RVCH at Head Assembly Building, Rev.1
Work Order 00327076-01, Inspection of the CV Polar Crane and Hooks, Dated April 23, 2004
Work Order 00137457-01, Inspection of the CV Polar Crane and Hooks, Dated October 12, 2002
Adverse Condition Investigation Action Request Nos. 164027-02 & 00167574, Polar Crane Problems
Crane Certification Company Certificate CCC-36909, Spreader Beam Load Test, Dated August 17, 2004
Quality Assurance Documents Package - Replacement Reactor Vessel Closure Head, H.B. Robinson P.O. No. 3382-0039
CPL_2_OH01_007_01, Baseline UT/ET Report for Penetration No. 7
CPL_2_OH01_039_01, Baseline UT/ET Report for Penetration No. 39
CPL_2_OH01_063_01, Baseline UT/ET Report for Penetration No. 63
Engineering Change (EC) 56266R1, Reactor Head and Service Structure Replacement
REG-NGGC-0010, 10 CFR 50.59 Screening for EC 56266R1, Rev. 7
Westinghouse Calculation CN-RCDA-04-157, RRVCH ASME Section XI Code Reconciliation, Rev. 2
QAA/0300-2004-SUR-01, Surveillance at Japan Steel Works, Muroran, Japan
QAA/0300-2004-SUR-02, Surveillance at Mitsubishi Heavy Industries, Kobe and Futami, Japan
Letter from Ted Huminski to Vaughn Wagoner dated 9/7/04, Trip Report - Mitsubishi Heavy Industries - Kobe
RNAS 05-031, MHI Reactor Head Replacement Manufacturing Surveillance - Shipping Document Review Surveillance Trip Report

<u>RPVH and Control Rod Drive Mechanism (CRDM) Housing Fabrication Records</u>

Certified Material Test Report (CMTR) for the RPVH forging (Material: SA-508, Grade 3, Class 1, Heat No. 03W159-1-1), including Magnetic Particle (MT) and Ultrasonic (UT) examinations reports of the RPVH forging prior to welding
CMTR for twelve (12) Latch Housings (Material: SA-182, Grade F316, Heat/Lot Nos. RK0E01A, RK0E01D, RK0E01G), including Liquid Penetrant (PT) and UT examinations reports
CMTR for twelve (12) CRDM Rod Travel Housings (Material: SA-182, Grade F316, Heat/Lot Nos. RK0E02A, RK0E02D, RK0E02G), including PT and UT examinations reports

CMTR for seventeen (17) CRDM to RPVH Adapters (Material: SB-167, N06690, Heat/Lot Nos. ONNC6146, ONNC6150, ONNC6151), including PT and UT examinations, hydrostatic testing, and dimensional examinations reports
CMTR for RPVH Lifting Lugs (Material: SA-533, Type B, Class 1, Heat No. 7-4679), including UT and dimensional examination reports
RT films and NDE examiner certification records for CRDM Rod Travel Housing to Latch Housing welds WC-L009-16A, 20A, 38A, 42A, 63A, and 67A, and CRDM Latch Housing to RPVH Adapter welds WC-L202-18A, 21A, 22A, 42A and 62A

Non Conformance Report (NCR) UGNR-HBN2-RHV-00 covering repair of PT indications in J-groove welds
Repair records of J-groove weld WC-S109-41A-R-A on RPVH penetration No. 41 after Hydrostatic Test
Hydrostatic Test Report, including the chemical analysis report of the test water and reports of post hydrostatic test NDEs performed on: (1) exterior of the RPVH (MT), (2) all RPVH Lifting Lugs welds (MT/PT), (3) all J-groove welds (PT), (4) all inner cladding welds (PT), (5) all keyway and mating surface clad welds (PT), (6) all Latch Housing to RPVH Adapter welds (PT), and (7) all Rod Travel Housing to Latch Housing welds (PT)
Final post weld heat treatment (PWHT) records, including time-temperature strip charts for the RPVH
Deviation notice (DN) 60791 on material selected for RPVH Adapters
Westinghouse calculation for reconciliation of the replacement RPVH with the original construction Code as required by ASME Section XI Code

<u>Records Examined for RPVH and CRDM Welds Reviewed</u>

RT films and NDE examiner certification records for CRDM Rod Travel Housing to Latch Housing welds WC-L009-16A, 20A, 38A, 42A, 63A, and 67A, and CRDM Latch Housing to RPVH Adapter welds WC-L202-18A, 21A, 22A, 42A and 62A
Non Conformance Report (NCR) UGNR-HBN2-RHV-00 covering repair of PT indications in J-groove welds
Repair records of J-groove weld WC-S109-41A-R-A on RPVH penetration No. 41 after Hydrostatic Test
Hydrostatic Test Report, including the chemical analysis report of the test water and reports of post hydrostatic test NDEs performed on: (1) exterior of the RPVH (MT), (2) all RPVH Lifting Lugs welds (MT/PT), (3) all J-groove welds (PT), (4) all inner cladding welds (PT), (5) all keyway and mating surface clad welds (PT), (6) all Latch Housing to RPVH Adapter welds (PT), and (7) all Rod Travel Housing to Latch Housing welds (PT)
Final post weld heat treatment (PWHT) records, including time-temperature strip charts for the RPVH
Deviation notice (DN) 60791 on material selected for RPVH Adapters
Westinghouse calculation for reconciliation of the replacement RPVH with the original construction Code as required by ASME Section XI Code
Welder qualification records for 23 welders
Welding material certification records for: (1) Electroslag Cladding Material - ER309L Wire Heat/Lot No. 1B118 and Flux PFB-7FK Lot No. 3L6112025, (2) E308-16 - Heat/Lot A3Z01N, (3) E309-16 - Heat/Lot A1221905N, (4) ERNiCrFe - Heat/Lot Nos. Nx3668JK, Nx3609JK, and Nx3474JK, (5) ER316L - Heat/Lot No. BF36099, and (6) EniCrFe-7 - Heat/Lot Nos. 312392, 310388, and 304373

Certification records for PT examination materials: (1) Penetrant Batch Nos. 05B02k, 2L1089, 3D1491, 4C2494, and 3I0964, (2) Cleaner/Remover Batch Nos. 05B01K, 3H93, 4B0295, 4E96, 4E0696, 4H97, 3H93, 4H0297, and 3J02, and (3) Developer Batch Nos. 05B07K, 4O95, 4E96, 4G97, and 4H96

Nondestructive Examination (NDE) and General Inspection Personnel Certification records for: 14 Level II PT examiners, 3 MT examiners, 5 Level II UT examiners, 4 Level II Radiographic (RT) examiners, and 4 Level I General Inspectors