UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

November 12, 2010

Mr. Robert J. Duncan, II
Vice President
Carolina Power and Light Company
H.B. Robinson Steam Electric Plant Unit 2
3581 West Entrance Road
Hartsville, SC 29550

SUBJECT: H.B. ROBINSON STEAM ELECTRIC PLANT - NRC INTEGRATED INSPECTION REPORT 05000261/2010004 AND 05000261/2010501; ASSESSMENT FOLLOW-UP LETTER

Dear Mr. Duncan:

On September 30, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your H.B. Robinson Steam Electric Plant. The enclosed integrated inspection report documents the inspection findings, which were discussed on November 12, 2010, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two findings that have potential safety significance greater than very low safety significance. The first finding is associated with a failure to establish and maintain an adequate emergency procedure that ensured reactor coolant pump seal cooling was maintained following a reactor trip. The second finding is associated with a failure to implement the requirement of a Systems Approach to Training in the operator training and remediation program for the implementation of the PATH-1 emergency operating procedure. Although these findings have potential safety significance, they did not represent immediate safety concerns. The findings do not present current safety concerns because prior to plant start-up the licensee performed extensive training to the operators on the entire Path-1 procedure. Two separate NRC inspectors observed the simulator training of three different crews to verify adequate command and control, board awareness and control of critical plant parameters.

In addition, the report documents three self-revealing findings of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it have been entered into your corrective action program (CAP), the NRC is treating this violation as a non-cited violation, in accordance with Section 2.3.2 of the NRC's Enforcement Policy. If you contest the non-cited violation or findings, 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

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Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the H.B. Robinson facility. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Senior Resident Inspector at the H.B. Robinson facility.

Augmented Inspection Team Report 05000261/2010009 issued July 2, 2010 identified 14 Unresolved Items (URI). Eleven of those URIs are dispositioned in this report. One URI (URI 05000261/2010009-09) was dispositioned in the Problem Identification and Resolution Inspection Report (IR) 05000261/2010006. The remaining two URIs (URI 05000261/2010009-01 and 05000261/2010009-03) will be dispositioned in IR 05000261/2010013.

In addition, as a result of its quarterly review of plant performance, which was completed on October 25, 2010, the NRC updated its assessment of H. B. Robinson Steam Electric Plant. The NRC's evaluation consisted of a review of performance indicators and inspection results. This letter informs you of the NRC's assessment of your facility. This letter supplements, but does not supersede, the mid-cycle letter issued on September 1, 2010.

The NRC's review of H. B. Robinson Steam Electric Plant identified that the Unplanned Scrams per 7000 Critical Hours performance indicator has crossed the green-to-white threshold. As a result of our assessment review, we have assessed H.B. Robinson Steam Electric Plant's performance to be in the Regulatory Response Column of the NRC's Action Matrix. We will conduct a supplemental inspection (Inspection Procedure 95001) when you notify us of your readiness for the NRC to review the actions taken to address the White Performance Indicator.

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/

Randall A. Musser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 05000261/2010004 and 05000261/2010501
w/Attachment: Supplemental Information

cc w/Encl: (See page 3)

cc w/encl:

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-261
License No:	DPR-23
Report No:	005000261/2010004 and 05000261/2010501
Facility:	H. B. Robinson Steam Electric Plant, Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	July 1, 2010 - September 30, 2010
Inspectors:	J. Hickey, Senior Resident Inspector D. Bollock, Resident Inspector D. Mills, General Engineer J. Worosilo, Project Engineer A. Nielsen, Senior Health Physicist (Section 2RS8) J. Brady, Senior Resident Inspector, McGuire (Section 1R19) P. Braxton, Reactor Inspector (Section 4OA5) G. Skinner, Contractor (Section 4OA5) M. Bates, Senior Operations Engineer (Section 1R11) G. Laska, Senior Operations Engineer (Section 1R11) C. Kontz, Senior Project Engineer (Section 1R11) J. Beavers, Emergency Preparedness Inspector (Section 1EP2, 1EP3, 1EP4, 1EP5, 4OA1, 4OA6)
Approved by:	R. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000261/2010004 and 05000261/2010501, 07/01/2010 – 09/30/2010; Carolina Power and Light Company; H.B. Robinson Steam Electric Plant, Unit 2; Post Maintenance Testing, Licensed Operator Requalification, and Event Follow-up.

The report covered a three month period of inspection by resident inspectors, senior operations engineers, emergency preparedness inspector, senior health physicist, and a reactor inspector. Two NRC identified findings, two self revealing findings and one self revealing violation were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross cutting aspects were determined using IMC 0310, "Components within the Cross Cutting Areas". Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- (TBD) The inspectors identified an Apparent Violation (AV) of 10 CFR 55.59(c), "Requalification program requirements", for the licensee's failure to properly implement elements of a Commission approved program developed using a systems approach to training (SAT), that was implemented in lieu of meeting the requirements defined in 10 CFR 55.59 (c). The finding was entered into the licensee's corrective action program as NCR-423232, NCR-423238, and NCR-423239. Corrective actions for this finding are still being evaluated.

  The licensee's failure to properly implement elements of a Commission approved requalification program was a performance deficiency. The finding was determined to be more than minor because it was associated with the Initiating Events Cornerstone and affected the cornerstone's objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to implement training requirements for Path-1 and perform adequate retraining of operators that demonstrated areas of weakness during operating tests contributed to operators' failure to identify and implement actions to mitigate a loss of seal cooling to the reactor coolant pumps (RCPs) during the events of March 28, 2010. Contrary to Augmented Inspection Team Report 05000261/2010009, further inspection revealed that RCP seal injection was not adequate coincident with a loss of cooling to the thermal barrier heat exchanger to the "B" RCP. Using Manual Chapter Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding could result in reactor coolant system (RCS) leakage exceeding Technical Specification limits. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis is required by a regional senior reactor analyst due to an increase in the likelihood of an RCP seal LOCA. The significance of this finding is designated as To Be Determined (TBD) until the safety characterization has been completed. The cause of this finding was directly related to the cross cutting aspect of Personnel Training and Qualifications in the Resources component of the Human Performance area, in that the licensee failed to ensure the adequacy of the training provided to operators to assure nuclear safety. (H.2(b)) (Section 1R11.3)

- (TBD) The inspectors identified an apparent violation (AV) of Technical Specifications (TS) 5.4.1, “Procedures”, for the licensee’s failure to establish and maintain an adequate emergency procedure that ensured reactor coolant pump (RCP) seal cooling was maintained following a reactor trip. The licensee has entered this into the CAP as nuclear condition report (NCR) 423147. Corrective actions for this finding are still being evaluated.

  The failure to establish and maintain an emergency procedure that would ensure adequate reactor coolant pump seal cooling, preventing seal degradation and a possible seal LOCA was a performance deficiency. The finding is more than minor because it is associated with the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations, specifically a loss of seal cooling to prevent the initiation of a RCP seal loss of coolant accident (LOCA). Using Manual Chapter Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding could result in RCS leakage exceeding Technical Specification limits. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis is required by a regional senior reactor analyst due to an increase in the likely hood of an RCP seal LOCA. The significance of this finding is designated as To Be Determined (TBD) until the safety characterization has been completed. The cause of this finding had a cross-cutting aspect of Documentation, Procedures, and Component Labeling, in the Resources component of the cross-cutting area of Human Performance, in that the licensee failed to ensure procedures for emergency operations were adequate to assure nuclear safety. (H.2(c)) (Section 4OA3.2)

- Green. A self revealing Green finding was identified for a failure to have adequate work orders to properly configure and post maintenance test the volume control tank (VCT) level comparator module. The licensee’s procedure ADM-NGGC-0104, Work Implementation and Completion, required that work orders contain all work activities necessary to perform all related work activities including Post Maintenance Testing (PMT). The licensee’s work orders for installing a jumper on the VCT level comparator module and for post maintenance testing failed to contain adequate instructions to properly configure (place jumper in correct location) and post maintenance test the volume control tank level comparator module. This resulted in the failure of the charging pump suction to automatically transfer from the volume control tank to the refueling water storage tank (RWST) when the auto transfer VCT low level setpoint was reached. The licensee’s identified corrective actions included repairing the subject VCT level module, reviewing the adequacy of other replacement NUS modules that have non-safety control functions and revising the site specific PMT procedures to provide more specific guidance for ensuring that the control loop circuit is adequately tested.

The failure to have adequate work order instructions to properly configure and post maintenance test the volume control tank level comparator module is a performance deficiency. This finding is greater than minor because the failure to auto transfer from the VCT to the RWST could cause a failure of the charging pump, resulting in the loss of seal injection which is a precursor to a seal LOCA. Using IMC 0609, “Significance Determination Process,” (SDP) Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required since the finding could have likely affected other mitigation systems resulting in a total loss of their safety function. This issue was evaluated using IMC 0609, Appendix A (SDP Phase 2) as being potentially greater than green with loss of component cooling water (LOCCW) and loss of service water (LOSW) as the dominant sequences. A phase 3 SDP risk evaluation was performed by a regional senior reactor analyst in accordance with the guidance in IMC 0609 Appendix A utilizing the NRC’s Robinson Standardized Plant Analysis Risk (SPAR) model. The VCT level comparator module performance deficiency resulted in a core damage frequency increase of less than 1E-6, Green. The risk was mitigated by the availability of the letdown and normal makeup charging pump suction sources, which would be available under certain conditions reducing the likelihood of an autoswap demand. Another factor which mitigated the risk is that the fire shutdown procedures for most fire areas specify use of a manual RWST supply valve. The performance deficiency is characterized as Green, a finding of very low safety significance. This issue has a cross-cutting aspect in the resources component of the human performance area because the licensee did not provide complete, accurate, and up-to-date work packages for the configuration and testing of the VCT comparator module. (H.2.(c)) (Section 1R19)

- Green. A self-revealing Green finding was identified for the licensee’s failure to adequately follow guidance in a design change package for the installation of non safety-related 4kV cables. This resulted in cables with design features inappropriate for the application being installed and eventually led to a fire and a reactor trip. Specifically, the licensee failed to follow the cable vendor recommendations and a self-imposed administrative requirement/standard for cable installation contained in cable specification L2-E-035, “Specification for 5,000 Volt Power Cable”. The licensee entered this into the CAP as NCR 390095. As corrective actions, the licensee replaced the cable, conduit and other damaged equipment, including evaluation on damage to cables in overhead, and the feeder cables to station service transformer (SST) 2E and 4kV bus 5.

  The failure to follow the guidance in the design change package to install non safety-related cables between Bus 4 and Bus 5 in accordance with their design change program and vendor and cable installation specifications was a performance deficiency. This finding was determined to be more than minor because it affected the Initiating Events Cornerstone objective of limiting events that upset plant stability, and was related to the attribute of Design Control (i.e., Plant Modifications). Specifically, the inadequate cable modification was determined to be the root cause of the reactor trip that occurred on March 28, 2010. This deficiency also paralleled Inspection Manual Chapter 0612, Appendix E, Example 2.e, as the licensee did not follow their own administrative requirements and vendor recommendations for cable installation. The performance deficiency was screened using Phase 1 of Inspection Manual Chapter 0609, Significance Determination Process, which determined that because the finding increases the likelihood of a fire, a Phase 3 SDP analysis was required. A phase 3 SDP risk evaluation was performed by a regional senior reactor analyst in accordance with the guidance in IMC 0609 Appendix F utilizing the NRC's Robinson SPAR model. The Phase 3 analysis determined the finding to be of very low safety significance (Green) because the core damage frequency increase was less than 1E-6. There is not a cross-cutting aspect associated with the finding because the performance deficiency involving the cable installation occurred greater than 20 years ago and does not reflect current licensee performance. (Section 4OA5.11)

- Green. A self-revealing Green NCV of 10 CFR 55.46(c), "Simulation Facilities", was identified for a plant referenced simulator used for administration of operating tests not correctly modeling the reference plant. A loss of electrical power that resulted in a loss of component cooling water (CCW) to the reactor coolant pump seals was not properly modeled in the simulator. When power to safety-related 480 volt bus E-2 was transferred to the emergency diesel generator in the reference-plant, FCV-626, thermal barrier heat exchanger outlet isolation flow control valve, closed. The simulator modeled FCV-626 to respond to CCW flow through the valve and did not model the effect of a loss of power to the valve operator and associated control circuit. Consequently, with a loss of power to bus E-2, the simulator model allowed this valve to remain open. The licensee documented the issue in Significant Adverse Condition Investigation Report, 390095. As corrective action the licensee changed the simulator modeling to match the plant configuration.

  The inspectors determined that the failure of the simulator to accurately demonstrate reference plant response was a performance deficiency. This finding was more than minor because it affected the human performance attribute of the initiating events cornerstone in that the unexpected closure of FCV-626 raises the likelihood of human error in response to a loss and subsequent re-energization of the E-2 Bus. This could challenge reactor coolant pump seal cooling and result in reactor coolant pump seal failure. The finding was evaluated using the Operator Requalification Human Performance SDP (MC 0609, Appendix I) because it was a requalification training issue related to simulator fidelity. The finding was of very low safety significance (Green) because the discrepancy did not have an impact on operator actions resulting in a total loss of RCP seal cooling and subsequent increase in reactor coolant system (RCS) leakage. There is not a cross-cutting aspect associated with the finding because the performance deficiency involving the simulator modeling occurred over 3 years ago and does not reflect current licensee performance. (Section 1R11.2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period in a refueling outage. The unit was returned to service on July 20, 2010. A reactor trip occurred on September 9, 2010, due to a turbine control valve problem and was returned to service on September 14, 2010. The unit operated at full power for the remainder 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:

- Residual Heat Removal (RHR) Train "A" while performing maintenance on RHR "B" pump breaker
- "B" Emergency Diesel Generator (EDG) while performing maintenance on the "A" EDG
- "B" CCW Pump while performing maintenance on the "C" CCW Pump

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" Emergency Diesel Generator 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.
- Breakers were correctly positioned.
- Cabinets, cable trays, and conduits were correctly installed and functional.
- Visible cabling appeared to be in good material condition.

Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R05 Fire Protection

a. Inspection Scope

For the five 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 5, Component Cooling Pump Room
- Fire Zone 2, “A” Diesel Generator Room
- Fire Zones 25 E&F, Turbine Building Mezzanine Level
- Fire Zones 25 A&B, Turbine Building Ground Level
- Fire Zone 22, Control Room

b. Findings

No findings were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Internal Flooding

Because the RHR pump pit and CCW rooms contain risk-significant SSCs which are susceptible to flooding from postulated pipe breaks in the Auxiliary building, the inspectors used remotely operated cameras to scan the RHR pump pit and walked down the CCW room to verify that the area configurations, 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 were identified.

1R11 Licensed Operator Requalification

.1 Quarterly Simulator Performance

a. Inspection Scope

The inspectors observed licensed-operator performance during requalification simulator training to verify that operator performance was consistent with expected operator performance, as described in scenario LOC 0007R Rev. 11B. This training tested the operators' ability to operate components from the control room, direct auxiliary operator actions, and determine the appropriate emergency action level classifications while responding to Volume Control Tank level failing high, high radiation in the auxiliary building following loose parts monitor alarm, and a loss of coolant accident. The inspectors focused on clarity and formality of communication, the use of procedures, alarm response, control board manipulations, group dynamics, and supervisory oversight. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Simulator Fidelity

(Closed) Unresolved Item (URI) 05000261/2010009-04, Fidelity of Plant-Referenced Simulator for Conduct of Component Cooling Manipulations

a. Inspection Scope

The inspectors reviewed simulator performance and reference plant event data in order to compare the response of the simulator to the reference plant. The simulator performance was inspected using the criteria listed in Inspection Procedure 71111.11, Licensed Operator Requalification Program.

b. Findings

Introduction: A self-revealing Green NCV of 10 CFR 55.46(c), “Simulation Facilities”, was identified for a plant-referenced simulator used for administration of operating tests not correctly modeling the reference plant. A loss of electrical power that resulted in a loss of CCW to the reactor coolant pump seals was not properly modeled.

Description: The licensee did not design and implement a simulator model that accurately demonstrated the reference plant response to a loss of safety-related 480 volt bus E-2. Following an actual event on the reference plant, it was determined that when power to bus E-2 was transferred to the emergency diesel generator, FCV-626, thermal barrier heat exchanger outlet isolation flow control valve, closed. The simulator modeled FCV-626 to respond to CCW flow through the valve and did not model the effect of a loss of power to the valve operator and associated control circuit. Consequently, with a loss of power to bus E-2, the simulator model caused this valve to remain open.

Analysis: The inspectors determined that the failure of the simulator to accurately demonstrate reference plant response was a performance deficiency. Specifically, the simulator did not accurately model the closure of FCV-626 when the safety related 480 volt bus E-2 was momentarily de-energized. The issue was self revealing because it was identified during a review of the loss of power event which occurred on March 28, 2010. This finding was more than minor because it affected the human performance attribute of the initiating events cornerstone in that the unexpected closure of FCV-626 raises the likelihood of human error in response to a loss and subsequent re-energization of the E-2 Bus. This could challenge reactor coolant pump seal cooling and result in reactor coolant pump seal failure.

The finding was evaluated using the Operator Requalification Human Performance SDP (IMC 0609, Appendix I) because it was a requalification training issue related to simulator fidelity. The SDP, Appendix I, Block 12, required the inspector to determine if deviations between the plant and simulator could impact operator actions. The noted difference was determined to be a contributing cause to the operator’s delay in re-establishing thermal barrier heat exchanger cooling prior to the loss of seal injection during the March 28, 2010 event. Therefore, the answer to the Block 12 question was yes, which resulted in a finding of very low safety significance (Green). The finding was of very low safety significance (Green) because the discrepancy did not have an impact on operator actions resulting in a total loss of RCP seal cooling and subsequent increase in reactor coolant system (RCS) leakage. There is not a cross-cutting aspect associated with the finding because the performance deficiency involving the simulator modeling occurred over 3 years ago and does not reflect current licensee performance.

Enforcement: 10 CFR 55.46(c) states that a plant-referenced simulator is required to be used for the administration of operating tests and that the simulator must demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond. The plant-referenced simulator must be designed and implemented so that it is sufficient in scope and fidelity to allow conduct of the evolutions listed in 10 CFR 55.59(c)(3)(i)(A) through (AA), including the loss of electrical power and loss of component cooling to an individual component.

Contrary to the above, the licensee did not design and implement a simulator model that accurately demonstrated the reference plant response to a loss of safety-related 480 volt bus E-2. Following an actual event on the reference plant on March 28, 2010, it was determined that when power to bus E-2 was transferred to the emergency diesel generator in the reference-plant, FCV-626, thermal barrier heat exchanger outlet isolation flow control valve, closed. The simulator modeled FCV-626 to respond to CCW flow through the valve and did not model the effect of a loss of power to the valve operator and associated control circuit.

Consequently, with a loss of power to bus E-2, the simulator model caused this valve to remain open. Therefore, the plant-referenced simulator did not correctly model the reference plant's response to a loss of safety-related 480 volt bus E-2 and the subsequent loss of component cooling to the reactor coolant pump thermal barrier heat exchanger. As corrective action the licensee changed the simulator modeling to match the plant configuration. Because this issue is of very low safety significance and has been entered into the licensee's corrective action program as NCR 390095, the violation is being treated as a Non-Cited Violation consistent with Section 2.3.2 of the NRC Enforcement Policy and is designated as NCV 50-261/2010004-02, Failure to Design and Implement a Simulator Model that Demonstrated Reference Plant Response.

URI 05000261/2010009-04, Fidelity of Plant-Referenced Simulator for Conduct of Component Cooling Manipulations, is closed

.3 Systems Approach to Training

(Closed) URI 05000261/2010009-05, Corrective Action for Operating Crew Performance Issues

a. Inspection Scope.

During the week of September 20-24, 2010, the inspectors reviewed training records from 2007 through 2010 associated with the operating crew on watch the evening of March 28, 2010 during a plant fire, reactor trip, safety injection, among other complications. The inspectors reviewed licensee procedures associated with their implementation of Licensed Operator Continuing Training (LOCT), training lesson plans, operator evaluation records and interviewed personnel. Each of these activities was performed to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in their own procedures. Inspection activities were conducted in accordance with Inspection Procedure 71111.11, "Licensed Operator Requalification Program," and 41500, "Training and Qualification Effectiveness." Documents reviewed during the inspection are stated in the Attachment.

b. Findings

Introduction: The inspectors identified an Apparent Violation (AV) of 10 CFR 55.59(c), "Requalification program requirements", for the licensee's failure to properly implement elements of a Commission approved program developed using a systems approach to training (SAT), that was implemented in lieu of meeting the requirements defined in 10 CFR 55.59 (c).

Description: The inspectors reviewed licensee training procedures, lesson plans, and operator evaluation documentation to determine if the licensee was meeting the requirements of 10 CFR 55.59, "Requalification," as well as their own procedure requirements pertaining to Licensed Operator Continuing Training (LOCT). Training Program Procedure TPP-200, "Licensed Operator/Shift Technical Advisor Continuing

Training Program" required the licensee to develop, maintain, and implement their LOCT program using the SAT process, as well as meet the requirements of 10 CFR 55.59.

During the review of training material, the inspectors identified that the Path-1 training material was not developed to train on all of the procedure steps within Path-1. The lesson material was developed to train primarily on immediate action steps, rules for procedure implementation, procedure transitions, and some relevant operating experience. The lesson plan did not contain supporting training information for subsequent action steps to ensure that operators not only knew the wording of the steps, but also that they understood the intent and basis behind those subsequent action steps. Path-1 training enabling objective, PATH-1-005, required operators to explain the basis of steps, cautions, and notes of Path-1. This enabling objective applied to the entire licensed operator population, as well as the Shift Technical Advisor (STA) position. Design and implementation of training based on learning objectives is a required element of a SAT as defined in Element (3).

Inadequate training on emergency operating procedures contributed to operators inadequately implementing Path-1 procedure steps during the events of March 28, 2010. The training on Path-1 procedure steps, as well as the basis and intent of those steps, is especially important at Robinson because the Path-1 procedure, although primarily written in accordance with Westinghouse Owners Group (WOG) guidance, does not contain the level of detail required to ensure performance meets the intent of the procedure steps. Proper implementation of their Path-1 Emergency Operating Procedure (EOP) requires the operator to retain and apply much of the knowledge from memory, as opposed to performing specific details prescribed within a procedure, details that also ensure compliance with the intent of the procedure.

The inspectors also identified weaknesses in the licensee's remedial training process. Element (4) of the SAT process requires evaluation of trainee mastery of learning objectives during training. The licensee did not adequately identify and document operator weaknesses so that the underlying cause of operator errors could be effectively remediated and re-evaluated.

Specifically, the inspectors identified that the licensee did not perform adequate retraining for operators that demonstrated areas of weakness on February 13, 2007, during their annual operating test. During the scenario, operators were expected to recognize that a complete loss of RCP seal cooling had occurred when charging and CCW flow were lost to all RCPs. The licensee's crew evaluation documentation states that the crew did not recognize, for a period of 21 minutes, that Reactor Coolant Pump (RCP) seal cooling had been lost. Review of training records indicated that the operators were not remediated on their failure to identify the loss of RCP seal cooling. One of these operators was on shift as a control room operator during the events of March 28, 2010, when inadequate seal cooling to the RCPs was experienced and not identified by the operators.

The inspectors identified additional examples of inadequate retraining for operators that demonstrated areas of weakness during operating tests. The inspectors identified instances where the training program was not identifying and documenting operator weaknesses so that the underlying cause of operator errors could be remediated and re-evaluated prior to those operators performing licensed duties. The licensee's training documentation contains instances where operator errors were identified; however, the underlying reason for the error was not documented. The underlying reason causing operators to make errors is required to be addressed by remediation training and re-evaluation. When underlying reasons are not understood and documented, it is difficult to tailor a remediation plan to correcting the weakness and also difficult to tailor a re-examination that provides confidence that the previous weakness no longer exists. Asking and documenting follow-up questions is one of the most direct methods of determining the nature of the operator weakness. Documentation of follow-up questions is largely absent from the completed evaluation forms.

Representative examples of inadequate evaluation and remediation documentation were noted in the following evaluated simulator scenarios:

- On December 8, 2009, licensed operators incorrectly diagnosed a feedwater transient that resulted in a manual reactor trip. Operators then failed to ensure complete phase B isolation. Remediation documentation is essentially the same for each operator on the crew and did not indicate specific individual weaknesses. The remediation documentation does not discuss the individual operator weaknesses that resulted in the operational errors.

- On May 9, 2010, a Shift Manager made an incorrect emergency classification. The individual's evaluation documentation only states that an emergency classification was made incorrectly. The report does not indicate why the classification was incorrect, or what operator weaknesses may have been associated with the incorrect classification. The individual was required to take two practice classification Job Performance Measures (JPMs) followed by a re-examination simulator scenario. The individual again did not correctly classify the event during his re-examination. The lack of specific documentation on the weaknesses resulting in the first misclassification brings into question the effectiveness of that remediation, considering that fact that the individual then did not classify the event correctly in the re-evaluation.

- On May 9, 2010, a Reactor Operator in the balance of plant position had evaluated weaknesses of not responding to Annunciator Panel Procedures (APPs), not recognizing that the turbine would not runback when in manual, not controlling charging to prevent letdown alarms, and not using the governor valve fast action push button during Path-1 immediate operator actions. The remediation plan included a brief and review of the failed scenario, two scenarios containing immediate action drills, followed by a re-evaluation. The documentation does not contain information that addresses the operator weaknesses associated with all of the documented operator errors.

- On May 26, 2010, a Shift Manager made an incorrect emergency classification. Similar to the May 9, 2010 misclassification, the individual's evaluation documentation only states that an emergency classification was made incorrectly. The report does not indicate why the classification was incorrect, or what operator weaknesses may have been associated with the incorrect classification. The individual's remediation documentation states that he was required to perform three practice classification JPMs. The individual then was required to pass another simulator scenario. The evaluation and remediation documentation does not discuss the probable causes for the incorrect classification;

therefore, there is no method to determine whether the operator weakness was identified, corrected, and accurately re-evaluated.

The licensee has entered these issues into their corrective action program as NCR-423232, NCR-423238, and NCR-423239.

Analysis: The licensee's failure to properly implement elements of a Commission approved requalification program was a performance deficiency. The finding was determined to be more than minor because it was associated with the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The failure to implement training requirements for Path-1 and perform adequate retraining of operators that demonstrated areas of weakness during operating tests contributed to operators' failure to identify and implement actions to mitigate a loss of seal cooling to the reactor coolant pumps during the events of March 28, 2010.

Using Manual Chapter Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding could result in RCS leakage exceeding Technical Specification limits. The Phase 2 analysis determined that this finding was Potentially Greater than Green; therefore, a Phase 3 analysis is required by a regional senior reactor analyst due to an increase in the likelihood of an RCP seal LOCA. The significance of this finding is designated as To Be Determined (TBD) until the safety characterization has been completed. The finding was directly related to the cross-cutting aspect of Training of Personnel of the Resources component in the cross-cutting area of Human Performance, in that the licensee failed to ensure the adequacy of the training provided to operators to assure nuclear safety. (H.2(b))

Enforcement: 10 CFR 55.59(c), "Requalification program requirements," states, in part, that a Commission approved program developed using a systems approach to training, can be implemented in lieu of meeting the requirements defined in 10 CFR 55.59 (c). Contrary to the above, the licensee failed to adequately implement the Commission approved program developed using a systems approach to training that was implemented in lieu of meeting the requirements defined in 10 CFR 55.59 (c). Specifically, the licensee failed to implement Elements (3) and (4) of an approved SAT program. 10 CFR 55.4, "Definitions," defines these two items as: Element (3) Training design and implementation based on the learning objectives and Element (4) Evaluation of trainee mastery of the objectives during training. The licensee failed to implement Element (3) by not developing adequate PATH-1 training material that thoroughly covered the training enabling objective PATH-1-005. Also, the licensee failed to implement Element (4) by not identifying, documenting, and evaluating operator weaknesses exhibited during evaluated scenarios. This issue has been entered into the licensee's corrective action system as NCR-423232, NCR-423238, and NCR-423239. This finding is identified as Apparent Violation (AV) 0500261/2010004-05, Failure to Correctly Implement a Systems Approach to Training for the Licensed Operator Requalification Program.

URI 05000261/2010009-05, Corrective Action for Operating Crew Performance Issues, is closed.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two system, structure and component (SSC)/function performance problems, conditions or the overall system performance history listed below to verify the appropriate handling of 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/systems:

- Action Request (AR) 390027-28, Condenser Vacuum Pump Motor Fire
- Overall system performance history of the Safety Injection System

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, and
- 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).

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

- 404094, “C” Safety Injection Pump Appears Mechanically Bound
- 390072, Safety Injection Automatically Initiated

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

For the six samples 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:

- Reactor Startup, physics testing and power ascension, July 17-23, 2010.
- Pressurizer Relief Valve (PRV) block valve PCV-455 shut with power due to leakage of Pressurizer Pressure Relief Valve, PCV-456 on July 8, 2010.
- Work Week August 9 through August 16, 2010, including maintenance on the "A" Charging Pump, "B" Motor Driven Auxiliary Feedwater Pump, and "C" CCW Pump breaker inspection.
- Work Week September 6 through September 12, 2010, including maintenance on the 'D' Service Water Pump breaker, Power Range Instrumentation calibration, and Control Rod Exercising.
- Work Week September 20 through September 24, 2010, including the inoperable "A" Emergency Diesel Generator orange risk associated with running the "B" Emergency Diesel Generator for common cause.
- Emergent inoperability of the "B" CCW Pump due to a pump seal leak repair on September 30, 2010.

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

- 422546, "A" EDG failed to reach rated Voltage during OST-409-1

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the four operability determinations associated with the ARs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of any necessary compensatory measures, and compliance with the Technical Specification (TS). The inspectors verified that the operability determinations were made as specified by Procedure OPS-NGGC-1305, Operability Determinations. The inspectors compared the justifications provided in the determinations to the requirements from the TS, the UFSAR, associated design-basis documents, to verify that operability was properly justified and the subject components or systems remained available, such that no unrecognized increase in risk occurred:

- 410777, Steam Driven Auxiliary Feedwater Pump governor excessively hunting
- 411758, PCV-456, Pressurizer Pressure Relief Valve, Leak by Resulting in Improved Technical Specification Entry
- 391995, CCW Thermal Barrier Isolation, FCV-626 Closed Automatically on Auto Start of the CCW Pumps
- 392245, Reactor Coolant System Cooldown Greater Than 100 degrees Fahrenheit (F)

Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

For the seven maintenance activities 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.

- WO 1782148, Perform SP-1551 on 4KV Buses 4 & 5
- WO 1753110, Cable Replacement: Tray R40 Phase 1
- WO 1709634, “A” EDG Test of DA-23-A using OP-604, OST-401-1, or OST-409-1
- WO 1811689, During MST-021, Relay SRB-2(B) was not energized as required
- WO 01598839, R-11/12 Containment Radiation Monitor sample pump replacement
- WO 01745874, NS-52 Nitrogen Supply to Reactor Coolant Drain Tank seal leak
- AR 390095, Root Cause for the March 28, 2010 event, URI 05000261/2010009-10, Failure of Charging Pump Suction Valves to Automatically Transfer Due to Errors in Implementing an Instrumentation Component Upgrade

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

- 419481 Breaker 52/17B, Failed to Close in the Test Position
- 417562 During MST-021, Relay SRB-2(B) was not Properly Energized as Required

b. Findings

(Closed) URI 05000261/2010009-10, Failure of Charging Pump Suction Valves to Automatically Transfer due to Errors in Implementing an Instrumentation Component Upgrade

Introduction: A self revealing Green finding was identified associated with the failure to have adequate work order instructions to properly configure and post maintenance test the volume control tank (VCT) level comparator module. The licensee’s procedure ADM-NGGC-0104, Work Implementation and Completion, required that work orders contain all work activities necessary to perform all related work activities including Post Maintenance Testing (PMT). These inadequate instructions resulted in the failure of the charging pump suction to automatically transfer suction from the volume control tank to the refueling water storage tank when the auto transfer VCT low level setpoint was reached.

<u>Description</u>: During an event that occurred on March 28, 2010, involving a cable failure and resulting fire and reactor trip, the charging pump suction from the VCT failed to automatically transfer to the RWST on low VCT level. The licensee issued significant adverse condition investigation report 390095 to document their investigation of the complete event. The investigation indicated that the charging pump suction failure to transfer went undetected by the operators for 49 minutes (19:00 - 19:49). In addition, the CCW thermal barrier return valve from the RCP seals went closed, due to repositioning after power was returned, isolating cooling to the RCP thermal barriers and was undetected by the operators for 39 minutes (18:52-19:31). These two conditions occurred during the same time period with an overlap of 31 minutes. The remaining charging pump was no longer providing seal cooling at 19:37 when the charging pump suction supply was depleted. The inspectors review of RCP Labyrinth seal dp for “B” RCP identified that flow went negative during a period when the thermal barrier isolation valve was shut indicating that seal injection flow was not going down the shaft into the RCS, but that RCS was flowing up the shaft through the thermal barrier heat exchanger (TBHX) which had no cooling water. This demonstrate that the RCP seal injection was not adequate coincident with a loss of cooling to the TBHX to the “B” RCP. Once cooling water was returned to the TBHX seal temperatures peaked and began reducing. RCP seal return temperature for RCP “B” peaked at 193.5 degrees F. The return of cooling to the seals prevented seal failure. The licensee has inspected the seals for “A” and “B” RCPs and found no damage.

The cause of the failure of the charging pump suction to auto transfer from the VCT to the RWST on low VCT level was the result of an improperly configured VCT level comparator module (LC-112B). The card had been installed and tested in 2008. The licensee found that the work order instructions for configuration of the level comparator module were not adequate to ensure the proper configuration of the module. The configuration instructions were contained in work order WO 1162348 on a comparator jumper configuration sheet. The configuration sheet had jumper W109 in the position “1-2” verses the correct “2-3” position. The jumper configuration sheet is generated during the work order process from the NUS module technical manual. In addition, the calibration instructions contained in work order WO 1064104 were not adequate to detect the incorrect module jumper configuration because the testing did not verify the configuration of the module by testing the entire control loop circuit. Procedure ADM-NGGC-0104, Work Implementation and Completion, Section 3.12, Full Scope Work Order (W/O), states that work orders normally contain all work activities necessary to perform all related work activities including Post Maintenance Testing (PMT). This condition is addressed in corrective actions for AR 390095 and corrective actions identified included reviewing the adequacy of other replacement NUS modules that have non-safety control functions and revising the site specific PMT procedures to provide more specific guidance on ensuring that the control loop circuit is adequately tested.

The RCP seal injection function is described in the design basis document (DBD) as a safety-related function. The charging pump suction auto-swap feature is designated as a non safety-related feature by its Quality (Q) class designation in licensee databases, while the manual swap feature using the main control board switches is safety-related by its Q class designation.

Analysis: The failure to have adequate work order instructions to properly configure and post maintenance test the volume control tank level comparator module is a performance deficiency. The issue was self revealing because it was identified by an event. This finding is greater than minor because the failure to auto transfer from the VCT to the RWST could cause a failure of the charging pump, resulting in the loss of seal injection which is a precursor to a seal LOCA. Using IMC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required since the finding could have likely affected other mitigation systems resulting in a total loss of their safety function. This issue was evaluated using IMC 0609, Appendix A (SDP Phase 2) as being potentially greater than green with loss of component cooling water (LOCCW) and loss of service water (LOSW) as the dominant sequences. A phase 3 SDP risk evaluation was performed by a regional senior reactor analyst in accordance with the guidance in IMC 0609 Appendix A utilizing the NRC's Robinson SPAR model. The top sequence was a Loss of Offsite Power (LOOP) sequence resulting in loss of RCP seal cooling (LOSC) and a seal LOCA with a failure to cooldown and implement high pressure recirculation. The top non-LOOP sequence was a LOCCW sequence. The LOSC involved loss of CCW seal cooling and loss of seal injection. The loss of CCW seal cooling was due to closure of the RCP thermal barrier outlet valve, FCV-626, on the LOOP signal and failure to reopen the valve. The loss of seal injection was caused by the unavailability of the charging pump suction sources and the failure of the charging pump suction auto swap over to occur due to the performance deficiency and failure of the operator to accomplish the transfer. The normal makeup source would not be available under LOOP conditions and a LOOP/ Safety Injection probability factor was developed to account for the availability of the normal letdown supply to the VCT under LOOP conditions. The resultant sequences were examined and large early release frequency sequences were not among the dominant sequences. External event risk included fire risk for several fire scenarios where the licensee procedures did not specify use of manual RWST supply valve, CV-358. The VCT level comparator module performance deficiency resulted in a core damage frequency increase of less than 1E-6, GREEN. The risk was mitigated by the availability of the letdown and normal makeup charging pump suction sources, which would be available under certain conditions reducing the likelihood of an auto swap demand. Another factor which mitigated the risk is that the fire shutdown procedures for must fire areas specify use of a manual RWST supply valve. The performance deficiency is characterized as Green, a finding of very low safety significance. This issue has a cross-cutting aspect in the resources component of the human performance area because the licensee did not provide complete, accurate, and up-to-date work packages for the configuration and testing of the VCT comparator module (H.2.(c))

Enforcement: The inspectors determined that this finding did not involve a violation of NRC requirements and therefore is not subject to enforcement action. Because this performance deficiency is not a violation, it is characterized as a Finding (FIN). This performance deficiency is in the licensee's corrective action program as AR 390095 and is identified as FIN 05000261/2010004-01, Failure to have Adequate Work and Post Maintenance Testing Instructions for the Volume Control Tank Comparator Module.

URI 05000261/2010009-10, Failure of Charging Pump Suction Valves to Automatically Transfer due to Errors in Implementing an Instrumentation Component Upgrade, is closed.

1R20 Refueling and Outage Activities

A. Refueling Outage Activities

For the refueling outage that was in progress at the beginning of the inspection period and ended on July 20, the inspectors evaluated licensee outage activities as described below to verify that the 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.

.1 Licensee Control of Outage Activities

a. Inspection Scope

During the refueling 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)
- Spent Fuel Pool Cooling
- 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.

b. Findings

No findings were identified.

.2 Monitoring of Heat-up and Start-up Activities

a. Inspection Scope

Prior to mode changes and on a sampling basis, the inspectors reviewed system lineups and/or control board indications to verify that TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. Also, the inspectors periodically reviewed RCS boundary leakage data, and observed the setting of containment integrity to verify that the RCS and containment boundaries were in place and had integrity when necessary. Prior to reactor startup, the inspectors walked down containment to verify that debris had not been left which could affect performance of the containment sumps. The inspectors reviewed reactor physics testing results to verify that core operating limit parameters were consistent with the design.

b. Findings

No findings were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

Periodically, the inspectors reviewed the items that had been entered into the CAP to verify that the licensee had identified problems related to outage activities at an appropriate threshold and had entered them into the corrective action program. For the significant problems documented in the corrective action program and listed below, the inspectors reviewed the results of the investigations to verify that the licensee had determined the root cause and implemented appropriate corrective actions, as required by 10 CFR 50, Appendix B, Criterion XVI, Corrective Action.

- AR 395921, Rubbing cable during upender operation
- AR 395788, Loose bolt on upender discovered during diving operations
- AR 410664, Tygon hose and oil residue identified by inspectors during containment closeout inspection

b. Findings

No findings were identified.

B. Forced Outage Activities

For the outage that began on September 9 and ended on September 14, the inspectors evaluated licensee outage activities as described below to verify that the 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.

.1 Licensee Control of Outage Activities

a. Inspection Scope

During the forced 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.

- 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.

b. Findings

No findings were identified.

.2 Monitoring of Heat-up and Start-up Activities

a. Inspection Scope

Prior to mode changes and on a sampling basis, the inspectors reviewed system lineups and/or control board indications to verify that TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. Also, the inspectors periodically reviewed RCS boundary leakage data.

b. Findings

No findings were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

Periodically, the inspectors reviewed the items that had been entered into the CAP to verify that the licensee had identified problems related to outage activities at an appropriate threshold and had entered them into the corrective action program. For the significant problems documented in the corrective action program and listed below, the inspectors reviewed the results of the investigations to verify that the licensee had determined the root cause and implemented appropriate corrective actions, as required by 10 CFR 50, Appendix B, Criterion XVI, Corrective Action.

- 421823, Pressure Controller PC-444J Saturated during the event response
- 421783, Automatic control of steam generator levels was not achieved until 25 percent reactor power

b. Findings

No findings 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.

- SP-1544, Dedicated Shutdown Diesel Generator (DSDG) Air Start Test
- OST-112-3, Reverse Flow Testing of CVC-298C and CVC-298F
- EST-050, Refueling Startup Procedure
- OST-409-1, EDG “A” Fast Speed Start

Inservice Testing Surveillance
- EST-010 Containment Personnel Airlock Leakage Test (Semiannual) Containment Isolation Valve Surveillance

Reactor Coolant System Leakage Surveillance
- OST-051, Reactor Coolant Leakage Evaluation (Every 72 Hours During Steady State Operation and Within 12 Hours of Reaching Steady State Operation)

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

- 422546 “A” EDG failed to reach rated Voltage during OST-409-1

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors evaluated the adequacy of licensee's methods for testing the Alert and Notification System (ANS) in accordance with NRC Inspection Procedure 71114, Attachment 02, "Alert and Notification System Evaluation". The applicable planning standard, 10 CFR Part 50.47(b)(5), and its related requirements, 10 CFR Part 50, Appendix E, Section IV.D, were used as reference criteria. The criteria contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, was also used as a reference.

The inspectors reviewed various documents which are listed in the Attachment. This inspection activity satisfied one inspection sample for the ANS on a biennial basis.

b. Findings

No findings were identified.

1EP3 Emergency Preparedness Organization Staffing and Augmentation System

a. Inspection Scope

The inspectors reviewed the licensee's Emergency Response Organization (ERO) augmentation staffing requirements and process for notifying the ERO to ensure the readiness of key staff for responding to an event and timely facility activation. The qualification records of key position ERO personnel were reviewed to ensure all ERO qualifications were current. A sample of problems identified from augmentation drills or system tests performed since the last inspection were reviewed to assess the effectiveness of corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 03, "Emergency Response Organization Staffing and Augmentation System." The applicable planning standard, 10 CFR 50.47(b)(2), and its related requirements, 10 CFR 50, Appendix E, were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment to this report. This inspection activity satisfied one inspection sample for the ERO staffing and augmentation system on a biennial basis.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, revisions 72, 73 and 74 of the Radiological Emergency Response Plan were implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The inspectors conducted a sampling review of the Plan changes and implementing procedure changes made between October 1, 2009, and August 31, 2010, to evaluate for potential decreases in effectiveness of the Plan. However, this review was not documented in a Safety Evaluation Report and does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 04, "Emergency Action Level and Emergency Plan Changes." The applicable planning standard, 10 CFR 50.47(b)(4), and its related requirements, 10 CFR 50, Appendix E, were used as reference criteria.

The inspectors reviewed various documents that are listed in the Attachment. This inspection activity satisfied one inspection sample for the emergency action level and emergency plan changes on an annual basis.

b. Findings

No findings were identified.

1EP5 Correction of Emergency Preparedness Weaknesses

a. Inspection Scope

The inspectors reviewed the corrective actions identified through the Emergency Preparedness program to determine the significance of the issues and to determine if repeat problems were occurring. The facility's self-assessments and audits were reviewed to assess the licensee's ability to be self-critical, thus avoiding complacency and degradation of their emergency preparedness program. In addition, the inspectors reviewed licensee self-assessments and audits to assess the completeness and effectiveness of all emergency preparedness related corrective actions.

The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 05, "Correction of Emergency Preparedness Weaknesses." The applicable planning standard, 10 CFR 50.47(b)(14), and its related requirements, 10 CFR 50, Appendix E, were used as reference criteria.

The inspectors reviewed various documents which are listed in the Attachment. This inspection activity satisfied one inspection sample for the correction of emergency preparedness weaknesses on a biennial basis.

b. Findings

No findings were identified.

1EP6 Drill Evaluation

a. Inspection Scope

On August 17, 2010 and September 7, 2010, the inspectors observed emergency preparedness drills to verify licensee self-assessment of classification, notification, and protective action recommendation development in accordance with 10 CFR 50, Appendix E. The inspectors also attended the post-drill critique to verify that the licensee properly identified failures in classification, notification and protective action recommendation development activities.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety, Public Radiation Safety

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

a. Inspection Scope

Waste Processing and Characterization. During inspector walk-downs, accessible sections of the liquid and solid radioactive waste (radwaste) processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included radwaste storage tanks; resin transfer piping, resin and filter packaging components; and abandoned evaporator equipment. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The 2009 Effluent Report and radionuclide characterizations from 2008 - 2010 for each major waste stream were reviewed and discussed with radwaste staff. For primary resin, reactor coolant system filters, and Dry Active Waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined quality assurance (QA) comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology for resins and filters was evaluated and discussed with radwaste staff. The inspectors also reviewed the licensee's procedural guidance for monitoring changes in waste stream isotopic mixtures.

Radwaste processing activities and equipment configuration were reviewed for compliance with the licensee's Process Control Program (PCP) and FSAR, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR 20, 10 CFR 61, and guidance provided in the Branch Technical Position on Waste Classification (1983). Reviewed documents are listed in the Attachment.

Radioactive Material Storage. During walk-downs of indoor and outdoor radioactive material storage areas, the inspectors observed the physical condition and labeling of storage containers and the posting of Radioactive Material Areas. The inspectors also reviewed licensee procedural guidance for storage and monitoring of radioactive material.

Radioactive material and waste storage activities were reviewed against the requirements of 10 CFR 20. Reviewed documents are listed in the Attachment.

Transportation. The inspectors directly observed preparation activities for the shipment of a contaminated underwater camera. The inspectors noted package markings and labeling, performed independent dose rate measurements, and interviewed shipping technicians regarding Department of Transportation (DOT) regulations.

Selected shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, waste classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for opening and closing Type A shipping containers were compared to manufacturer requirements. In addition, training records for selected individuals currently qualified to ship radioactive material were reviewed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR 20, 10 CFR 71 (which requires licensees to comply with DOT regulations in 49 CFR Parts 107, 171-180, and 390-397), as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR 172 Subpart H. Documents reviewed during the inspection are listed in the Attachment.

Problem Identification and Resolution. The inspectors reviewed NCRs in the area of radwaste/shipping. The inspectors evaluated the licensee's ability to identify and resolve the issues in accordance with procedure CAP-NGGC-0200, "Corrective Action Program", rev. 32. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Licensee CAP documents reviewed are listed in the Attachment.

The inspectors completed one sample as required by inspection procedure 71124.08.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors verified the PIs identified below. For each PI, the inspectors verified the accuracy of the PI data that had been previously reported to the NRC by comparing those data to the actual data, as described below. The inspectors also compared the licensee's basis in reporting each data element to the PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline." In addition, the inspectors interviewed licensee personnel associated with collecting, evaluating, and distributing these data.

Initiating Events Cornerstone

- Unplanned Scrams
- Unplanned Scrams with Complications
- Unplanned Power Changes

For the period from the first quarter of 2009 through the fourth quarter of 2009, the inspectors reviewed a selection of licensee event reports, operator log entries, daily reports (including the daily CR descriptions), monthly operating reports, and PI data sheets to verify that the licensee had accurately identified the number of scrams and unplanned power changes greater than 20 percent that occurred during the subject period. The inspectors compared those numbers to the numbers reported by the licensee for the PI. The inspectors also reviewed the accuracy of the number of critical hours reported, and the licensee's basis for crediting normal heat removal capability for each of the reported reactor scrams.

b. Findings

No findings were identified.

Emergency Preparedness Cornerstone

a. Inspection Scope

The inspector sampled licensee submittals relative to the Performance Indicators (PIs) listed below for the period October 1, 2009, and June 30, 2010. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," was used to confirm the reporting basis for each data element.

- Emergency Response Organization Drill/Exercise Performance (DEP)
- Emergency Response Organization Readiness (ERO)
- Alert and Notification System Reliability (ANS)

The inspection was conducted in accordance with NRC IP 71151, "Performance Indicator Verification." For the specified review period, the inspector examined data reported to the NRC, procedural guidance for reporting PI information, and records used by the licensee to identify potential PI occurrences. The inspector verified the accuracy of the PI for ERO drill and exercise performance through review of a sample of drill and event records. The inspector reviewed selected training records to verify the accuracy of the PI for ERO drill participation for personnel assigned to key positions in the ERO. The inspector verified the accuracy of the PI for alert and notification system reliability through review of a sample of the licensee's records of periodic system tests. The inspector also interviewed the licensee personnel who were responsible for collecting and evaluating the PI data. Licensee procedures, records, and other documents reviewed within this inspection area are listed in the Attachment.

This inspection activity satisfied one inspection sample each for the Drill/Exercise Performance, ERO Drill Participation, and Alert and Notification System as defined in IP 71151-05.

b. Findings

No findings were identified.

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 follow up, the inspectors performed frequent screenings of items entered into the CAP. The review was accomplished by reviewing daily AR reports.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected the following two ARs for detailed review.

- 401309, Unplanned Safety Injection Signal Received While Restoring Safeguards
- 406852, Boric Acid Storage Tank Bubbler Lines Cause Frequent Indicator Problems. This AR relates to Operator Workarounds, in that it identifies a condition which requires manual operator action to maintain boric acid storage tank level indication.

The inspectors reviewed these reports 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 these ARs 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 were identified.

4OA3 Event Follow-up

.1 Reactor Trip Due To Turbine Control Valve Closure

a. Inspection Scope

Following the reactor trip that occurred on September 9, 2010, the inspectors reviewed the status of mitigating systems and fission product barriers, equipment and personnel performance, and related plant management decisions to assist NRC management in making an informed evaluation of plant conditions. The inspectors also reviewed post-trip activities to verify that the licensee identified and resolved event-related issues prior to restarting the plant. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Emergency Operating Procedures

(Closed) URI 05000261/2010009-06, Adequacy of Emergency Operating Procedure Background Document

a. Inspection Scope

During the week of September 20-24, 2010, the inspectors interviewed members of the control room staff on duty during the March 28, 2010 event. The inspectors reviewed the licensee emergency procedures that were implemented during the event, as well as their background documents.

b. Findings

Introduction: The NRC has identified an apparent violation (AV) of Technical Specifications (TS) 5.4.1.a, “Procedures” for the licensee’s failure to establish and maintain an adequate emergency procedure that ensured reactor coolant pump (RCP) seal cooling was adequately maintained following a reactor trip and/or safety injection.

Description: The licensee's Path-1 emergency operating procedure, is a flow path compilation of Westinghouse Owners Group (WOG) procedures E-0, "Reactor Trip or Safety Injection," and E-1, "Loss of Reactor or Secondary Coolant." The first two columns of the flow path generally align with E-0, and the last two columns generally align with E-1. The WOG Background Document, Low Pressure, revision 2 step 19 states: "Check RCP Seal Cooling." The purpose for this step is to maintain seal cooling to the RCPs. Path-1 directs operators to secure RCPs if seal cooling is not maintained. In the Path-1 procedure, operators are directed to check the RCP Thermal Barrier Cooling Water Hi or Lo flow annunciator illuminated. If the annunciator is illuminated, thermal barrier cooling is considered not available and the operator is directed to verify that a charging pump is running. If a charging pump is running, Path-1 directs the operator to proceed to the next step without securing RCPs. If a charging pump is not running, Path-1 directs operators to secure RCPs.

During the March 28, 2010, event, FCV-626 thermal barrier heat exchanger outlet isolation flow control valve, had failed closed due to a temporary loss of power that resulted in thermal barrier cooling being lost for approximately 39 minutes. An incorrectly installed modification resulted in the failure of an auto-swap feature that was supposed to automatically transfer the charging pump suction from the Volume Control Tank (VCT) to the RWST on low level in the VCT. Also, operators failed to recognize that the lowering VCT level, in conjunction with valve CVC-310A, Charging Flow to Loop 1, opening on a loss of instrument air, which resulted in no longer providing adequate RCP seal injection coincident with the loss of thermal barrier cooling. Indications reviewed after the event indicated that seal leakoff temperatures on all three reactor coolant pumps began to increase toward RCS temperatures, which indicated that there was inadequate seal cooling.

Operators who were on duty during the event incorrectly performed the verification of RCP seal injection prior to opening FCV-626. Operators did not verify that adequate seal injection existed by review of diverse indications. The operators indicated during interviews that they verified that a charging pump was running prior to re-opening FCV-626. They specifically indicated that thermally shocking the RCP seals was not a concern with a charging pump running. At the time that FCV-626 was re-opened, CCW had not been flowing to the thermal barrier heat exchanger for 39 minutes and seal injection had been inadequate for 10 to 15 minutes coincident with no thermal barrier cooling. Inspectors considered this information when performing the review of licensee procedures that were used during the event. URIs 05000261/2010009-01, Monitoring of Plant Parameters and Alarms, and 05000261/2010009-03, Utilization of Operators During Events Requiring Use of Concurrent Procedures will inspect the operator performance aspects discussed above.

Inspectors subsequently identified that Path-1 only explicitly required the operators to verify that a charging pump was running for making the determination that adequate seal injection existed. Inspectors also identified that the Path-1 background document supported the incorrect assumption that a charging pump running was satisfactory indication that adequate seal injection flow existed. Operators, during the March 28, 2010, event, literally complied with the procedure step for verifying that a charging pump was operating when making the determination that they had adequate seal injection, but they did not comply with the intent of the step in that they did not verify that adequate seal cooling had been maintained.

Analysis: The failure to establish and maintain an emergency procedure that would ensure adequate reactor coolant pump seal cooling, preventing seal degradation and a possible seal LOCA was a performance deficiency. The finding was more-than minor because it is associated with the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations, specifically a loss of seal cooling to prevent the initiation of an RCP seal loss of coolant accident (LOCA). Using Manual Chapter Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the inspectors determined the finding required a Phase 2 analysis because the finding could result in RCS leakage exceeding Technical Specification limits. The Phase 2 analysis determined that this finding was potentially greater than green; therefore, a Phase 3 analysis is required by a regional senior reactor analyst due to an increase in the likelihood of an RCP seal LOCA. The significance of this finding is designated as To Be Determined (TBD) until the safety characterization has been completed. The cause of this finding had a cross-cutting aspect of Documentation, Procedures, and Component Labeling, in the Resources component of the cross-cutting area of Human Performance, in that the licensee failed to ensure procedures for emergency operations were adequate to assure nuclear safety. (H.2(c)).

Enforcement: Technical Specification 5.4.1, “Procedures,” requires in part that procedures shall be established, implemented, and maintained covering the activities in Regulatory Guide (RG) 1.33, Rev. 2, “Quality Assurance Program Requirements.” Item 6 of RG 1.33, Appendix A, states, in part, that typical safety-related activities such as combating emergencies and other significant events including reactor trip, shall be covered by written procedures. The licensee’s PATH-1 procedure is the implementing procedure for operator response to a reactor trip. Contrary to the above, the licensee did not adequately establish and maintain procedures to ensure that seal cooling was adequately maintained to the RCPs following a reactor trip. Specifically, the licensee’s PATH-1 procedure and associated Background Document incorrectly informed operators that verification of an operating charging pump was adequate to determine that RCP seal injection existed. Not maintaining adequate seal cooling to the RCPs affects the likelihood of a RCP seal-LOCA caused by thermally shocking RCP seals and seal failure. The licensee entered this finding into the corrective action program as NCR423147. This finding is identified as an Apparent Violation (AV) 0500261/2010004-04, Failure to Establish an Adequate PATH-1 Emergency Operating Procedure.

URI 05000261/2010009-06, Adequacy of Emergency Operating Procedure Background Documents, is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors observed Security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings were indentified.

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

a. Inspection Scope

The inspectors performed a walkdown of the two ISFSIs on site (reference dockets 72-3 and 72-60). The inspectors also reviewed changes made to programs and procedures and their associated 10 CFR 72.48 screens and/or evaluations to verify that changes made were consistent with the license or Certificate of Compliance; reviewed records to verify that the licensee has recorded and maintained the location of each fuel assembly placed in the ISFSIs; and reviewed surveillance records to verify that daily surveillance requirements were performed as required by technical specifications. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.3 Post Fire Event Pre-Startup Inspection

a. Inspection Scope

On March 28, 2010, Robinson Unit-2 experienced two fire events in 4160 volt switchgear, a reactor tip and safety injection. The inspectors performed the following inspections of issues and corrective actions prior to the licensee performing a reactor startup following that event.

- Two separate inspectors observed the simulator training of three different crews to verify adequate command and control, board awareness and control of critical plant parameters.

- The inspectors reviewed the changes to AOP-041, Response to a Fire Event, Rev. 1 to verify the changes support the effective use of the balance of plant operator.
- The inspectors verified the dedicated shutdown diesel had adequate starting air pressure when credited for risk mitigation purposes in mode 3.
- The inspectors reviewed the extent of condition review performed by the licensee regarding the simulator fidelity in response to FCV-626 unexpectedly going close when the E-2 bus was re-energized.
- The inspectors reviewed the licensee's reactor pressure vessel analysis in response to the 106 degrees F reactor coolant system cooldown, which occurred in less than one hour following the fire event.
- The inspectors reviewed emergency operating procedure changes including Path-1 and foldout page guidance.
- The inspectors reviewed the changes to GPP-004, Post Trip Stabilization, Rev. 13 to verify adequate guidance was in place to reset the main generator lockout.
- The inspectors reviewed and walked down the repairs to 4kv Bus 4 and 5 and associated cabling.

b. Findings

No findings were identified.

.4 URI 05000261/2010009-01, Monitoring of Plant Parameters and Alarms

This URI is still under NRC review and will be dispositioned in IR 05000261/2010013.

.5 (Closed) URI 05000261/2010009-02, RCS Cooldown Rate Exceeds Technical Specifications 3.4.3 Limit

On March 28, 2010, Robinson Unit-2 experienced two fire events in 4160 volt switchgear, a reactor tip and safety injection. During the course of events several non-vital power supplies were de-energized as a result of the fire effects. The loss of power caused four moisture separator reheater steam supply isolation valves to remain open and the alternate drain valves for the moisture separator drain tanks failed open. These open valves provided a main steam path to the condenser which resulted in a cooldown of the RCS. Technical Specification Limiting Condition For Operation 3.4.3 requires a cooldown rate not to exceed 100 degrees F in any one hour period. From the time period of 18:53 until 19:23 RCS cold leg temperature was reduced from 549.9 degrees F to 449.45 degrees F, which represented a cooldown greater than 100 degrees F. The cooldown continued until 19:26 with an RCS temperature of 443.5 degrees F. This resulted in a total RCS cooldown of 106.4 degrees F in 33 minutes. The Required Action Statement 3.4.3 A directs the licensee to restore the affected parameters to within limits in 30 minutes (3.4.3 A.1) and determine the RCS is acceptable for continued operation within 72 hours (3.4.3 A.2). RCS temperature was 450.6 degrees F and increasing at 19:37, this met the completion time for Required Action 3.4.3 A.1. However, the required evaluation actions of 3.4.3 A.2 were not completed within 72 hours. Technical Specification 3.4.3 B requires the unit to be placed in Mode 3 within 6 hours (3.4.3 B.1) and be in Mode 5 with RCS pressure less than 400 pounds per square inch gauge (psig) within 36 hours (3.4.3 B.2) if the required actions of Condition A are not met. The unit was in mode 3 as a result of the event at 1853 and subsequently placed in mode 5 with pressure less than 400 psig on 3/31 at 01:27. Therefore compliance with the requirements of Technical Specification 3.4.3 was satisfied. An evaluation demonstrating the RCS is acceptable for continued operation is documented in EC 76814. Based on the inspectors' review of plant data and documents, no performance deficiency was identified in the greater than 100 degrees F RCS cooldown event with respect to Technical Specification compliance. URI 05000261/2010009-01, Monitoring of Plant Parameters and Alarms, will inspect the operator performance aspects of the RCS cooldown.

URI 05000261/2010009-02, RCS Cooldown Rate Exceeds Technical Specifications 3.4.3 Limit, is closed.

.6 URI 05000261/2010009-03, Utilization of Operators During Events Requiring Use of Concurrent Procedures

This URI is still under NRC review and will be dispositioned in IR 05000261/2010013.

.7 (Closed) URI 05000261/2010009-04, Fidelity of Plant-Referenced Simulator.

The inspectors reviewed simulator performance and reference plant event data in order to compare the response of the simulator to the reference plant. The simulator performance was inspected using the criteria listed in Inspection Procedure 71111.11, Licensed Operator Requalification Program. Results of this inspection are detailed in Section 1R11.2.

.8 (Closed) URI 05000261/2010009-05, Corrective Action for Operating Crew Performance Issues.

During the week of September 20-24, 2010, the inspectors reviewed training records from 2007 through 2010 associated with the operating crew on watch the evening of March 28, 2010, during a plant fire, reactor trip, safety injection, among other complications. Results of this inspection are detailed in Section 1R11.3.

.9 (Closed) URI 05000261/2010009-06, Adequacy of Emergency Operating Procedure Background Documents.

During the week of September 20-24, 2010, the inspectors interviewed members of the control room staff on duty during the March 28, 2010, event. The inspectors reviewed the licensee emergency procedures that were implemented during the event, as well as their background documents. Results of this inspection are detailed in Section 4OA3.2.

.10 (Closed) URI 05000261/2010009-07, Loss of Seal Water Results in Failure of the "A" Main Condenser Vacuum Pump

On March 28, 2010, Robinson Unit-2 experienced two fire events in 4160 volt switchgear, a reactor tip and safety injection. During the course of events an additional fire was reported in the "A" Main Condenser Vacuum Pump Motor. The fire was extinguished without additional impact on the plant. The preliminary cause was attributed to a vacuum pump seizure due to loss of seal water resulting from the loss of electrical power to the seal water makeup source. Further inspection by the licensee determined no damage to the vacuum pump had occurred. The pump satisfactorily rotated by hand and a visual inspection showed no damage. The motor was inspected and determined to have a phase B short to ground. The cause of the shorted motor is most likely attributed to the two separate severe under voltage transients the motor was subjected to during the two fire events. The motor was replaced and adequate post maintenance testing performed to demonstrate availability. Based on the inspectors' review of licensee actions no performance deficiency was identified.

URI 05000261/2010009-07, Loss of Seal Water Results in Failure of the "A" Main Condenser Vacuum Pump, is closed.

.11 (Closed) URI 05000251/2010009-08, Deficiencies in Non Safety-Related Cable Installation

a. Inspection Scope

On March 28, 2010, Robinson Unit-2 experienced two fires events in 4160 volt switchgear, a reactor trip, and a safety injection. An URI was identified regarding a plant modification implemented in 1986 to expand the existing 4kV Bus 4 by installing 4kV Bus 5 and its associated components. The inspectors completed a review of the circumstances surrounding the modification including a qualitative evaluation of the risk significance. Documents reviewed by the inspectors are listed in the Attachment.

b. Findings

Introduction: A self-revealing Green finding was identified for the licensee not adequately following guidance in a design change package for the installation of non safety-related 4kV cables. Specifically, the licensee failed to follow the cable vendor recommendations and a self-imposed administrative requirement/standard for cable installation contained in cable specification L2-E-035, "Specification for 5,000 Volt Power Cable". This resulted in cables with design features inappropriate for the application being installed and eventually leading to a fire and a reactor trip.

Description: On March 28, 2010, an electrical fault occurred in a 4kV feeder cable from Bus 4 to Bus 5. The fault caused Bus 4 voltage to lower, which decreased the speed of the RCP that was powered from Bus 4. This caused the flow in RCS Loop B to decrease and initiated an automatic reactor trip. The failure of the output breaker in Bus 4 that feeds Bus 5 (breaker 52/24) to open exacerbated the event resulting in an electrical fire. The licensee investigated the fire event and identified cable failures inside a 4-inch conduit at the entrance of 4kV Bus 5 propagated to a second conduit at 90 degrees bend above 4kV Bus 5 in the Turbine Building. The analysis revealed that the cables likely failed because of being installed contrary to acceptable installation described in the manufacture's datasheet for the cables and cable specification requirements in L2-E-035, "Specification for 5,000 Volt Power Cable." It was determined that the guidance for the cable installation was not adequately followed in 1986 when an engineering modification (MOD 851) was implemented to expand the existing switchgear (Bus 4) by installing new switchgear (Bus 5).

The inspectors identified the following discrepancies with the cable installation at Robinson:

Modification specification L2-E-035 and Rome cable manufacture datasheet states:	**What Robinson Nuclear Plant installed:**
L2-E-035: 2.1.1& 2.1.2; Coated copper conductors	Uncoated copper conductors
L2-E-035: 2.3.2; 133% insulation and insulation shielding	100% insulation without insulation shielding
L2-E-035: 2.3.3; All cables be provided with an outer jacket	The installed cables did not have an outer jacket
L2-E-035: 2.1.3 & 2.3.3; Cable Insulation and jacketing that was self-extinguishing and non-propagating with regards to fire as described in IEEE 383-1974, Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations.	The event demonstrated that the cable lacked fire propagation properties because • the cable did not self extinguish after the fault was de-energized • flame was propagated along the cable • Rome cable manufacturer datasheet did not specify IEEE-383 rating
Rome cable manufacturer datasheet: Install cables in a non-magnetic conduit	Cables were installed in rigid steel conduit, which is magnetic

Additionally, the licensee concluded that there was insulation damage along several locations of the cable length and was most likely due to the design change modification package not including precautions for cable pulling tension limits or pulling instructions. Progress Energy's Significant Adverse Condition Investigation Report identified the specification (section 2.0, Design Criteria) and design change modification package required deviations waived by the Engineer be specified in the accompanying Purchase Order to justify the change. Contrary to the design change package, the original Purchase Order or technical justification documenting why the installed cable for the Bus 5 modification was different than the standard Specification L2-E-035 was not located with the copy of the MOD retrieved from records. The inspectors determined it was reasonable for the station to provide correct guidance to the field installers in 1986 because the design change modification package required specific instructions to be provided and the design change notice (DCN) package was reviewed by the maintenance department and senior engineers who were cognizant of the standards for the installation of 5kV power cables.

The licensee's Event Review Team Report stated that all warehoused cable were inspected and a search conducted, using catalog identification numbers, across their fleet to identify these types of cables or cables with similar construction and no deficiencies were noted.

Analysis: The failure to follow the guidance in the design change package to install non safety-related cables between Bus 4 and Bus 5 in accordance with the design change program and vendor and cable installation specifications was a performance deficiency. This finding was determined to be more than minor because it affected the Initiating

Events Cornerstone objective of limiting events that upset plant stability, and was related to the attribute of Design Control (i.e., Plant Modifications). Specifically, the inadequate cable modification was determined to be the root cause of the reactor trip that occurred on March 28, 2010. This deficiency also paralleled Inspection Manual Chapter 0612, Appendix E, Example 2.e, as the licensee did not follow their own administrative requirements and vendor recommendations for cable installation. The performance deficiency was screened using Phase 1 of Inspection Manual Chapter 0609, Significance Determination Process. The finding was assigned to the Initiating Events Cornerstone and was determined to be a fire initiator contributor that increased the likelihood of a fire and required a phase 3 SDP analysis utilizing IMC 0609 Appendix F. The phase 3 analysis was performed by a regional SRA utilizing the NRC's Robinson SPAR model. The analysis assumed that a high energy arcing fault (HEAF) could occur in the non-IEEE qualified thermoset feeder cables from 4KV Bus 4 to 4KV Bus 5. For approximately 2 feet of the 25 foot cable run, a HEAF in the feeder cable could damage the condensate pump cables located in nearby cable tray resulting in a reactor trip transient. The weighting factor for this damage yielded a probability of 8E-2. The dominant transient sequences were an anticipated transient without scram sequence with reactor coolant system pressure limited and a transient sequence with failures of main and auxiliary feedwater and a failure to implement feed and bleed leading to core damage. The core damage frequency increase due to the performance deficiency was less than 1E-6 resulting in characterizing the finding as Green, a finding of very low risk significance. The inspectors determined that there was no cross-cutting aspect associated with the finding because the performance deficiency occurred greater than 20 years ago and does not reflect current licensee performance.

Enforcement: The inspectors determined that this finding did not involve a violation of NRC requirements and therefore is not subject to enforcement action. Because this performance deficiency is not a violation, it is characterized as a Finding (FIN). The licensee entered this issue into the CAP as NCR 390095. The corrective actions taken by the licensee included replacement of the cable, conduit and other damaged equipment, and evaluation of damage to cables in overhead, and the feeder cables to station service transformer (SST) 2E and 4kv bus 5. Therefore this finding is identified as FIN 05000261/2010004-03, Deficiencies in Non Safety-Related Cable Installation Result in Electrical Fire and Reactor Trip.

URI 05000251/2010009-008, Deficiencies in Non Safety-Related Cable Installation, is closed.

.12 (Closed) URI 05000261/2010009-09, Failure to Repair Circuit Breaker 52/24 Resulting in Breaker Being Unable to Operate

This URI was dispositioned in the Problem Identification and Resolution Report, IR 05000261/2010006. A finding was issued during the closure of this URI. This finding was identified as: FIN 05000261/2010006-01, Failure to Correct a Control Power Fuse Defect in 4kV Breaker 52/24.

.13 <u>(Closed) URI 05000261/2010009-10, Failure of Charging Pump Suction Valves to Automatically Transfer Due to Errors in Implementing an Instrumentation Component Upgrade</u>.

The inspectors reviewed the licensee's root cause evaluation contained in AR 390095 and reviewed various associated documents listed in the attachment to determine the cause of the failure of the charging pump suction to auto-swap to the RWST. Results of this inspection are detailed in Section 1R19.

.14 <u>(Closed) URI 05000261/2010009-11, FCV-626 RCP Thermal Barrier Outlet Isolation CCW Valve, Unexpected Closure</u>

An unresolved item was identified regarding the closure of reactor coolant pump thermal barrier outlet isolation CCW valve, FCV-626 when power to safety-related 480 volt Bus E-2 was transferred to the emergency diesel generator during the March 28 events. The valve remained closed for approximately 39 minutes before the operators recognized the condition, reopened FCV-626, and restored CCW cooling to the RCP thermal barrier heat exchangers. Plant staff knew that FCV-626, a motor operated valve, was powered from Bus E-2 via MCC 6. However, plant staff, including operators, was unaware that FCV-626 would close on a momentary loss of power. Additionally, the simulator was modeled such that FCV-626 remained open when power to Bus E-2 was momentarily interrupted. The AIT opened the URI because additional NRC review was required to determine whether the design of FCV-626 represents a performance deficiency.

The inspectors reviewed the operability determination associated FCV-626, the reactor coolant pump thermal barrier outlet isolation valve, unexpectedly closing during the fire event and plant shut down that occurred on March 28, 2010. The inspectors assessed the accuracy of the evaluation, the use and control of any necessary compensatory measures, and compliance with the TS. The inspectors verified that the operability determination was made as specified by Procedure OPS-NGGC-1305, Operability Determinations. The inspectors compared the justifications provided in the determination to the requirements from the TS, the UFSAR, associated design-basis documents, and conducted interviews with a number of licensee personnel. The licensee determined the closure of FCV-626 during a loss of offsite power was an unintended response. The inspectors also reviewed history of loss of offsite power to determine past operability issues and found no prior occurrence. The licensee has resolved the unexpected valve performance with a permanent plant modification, which was inspected and documented in IR 05000261/2010003 Section 1R18. Based on the actions taken by the licensee and the completion of the NRC inspection, no performance deficiency was identified.

URI 05000261/2010009-11, FCV-626 RCP Thermal Barrier Outlet Isolation CCW Valve, Unexpected Closure, is closed.

.15 (Closed) URI 05000261/2010009-12, NUREG 0737 Response from Licensee to the NRC Describing the Behavior of RCP Seal Cooling Following a Loss of Offsite Power Event

An unresolved item was identified regarding the response from the licensee to the NRC of NUREG 0737, Clarification of Three Mile Island (TMI) Action Plan Requirements, Item II.K.3.25, “Power on Pump Seals”. This TMI item required the licensee to determine the consequences of a loss of RCP cooling due to a loss of offsite power event. In a letter to the NRC dated May 31, 1983, the licensee stated that no modifications were necessary because the CCW system is still operable during a loss of offsite power (powered from the emergency buses) and provides flow to the RCP thermal barrier heat exchangers. They also stated that the “B” and “C” CCW pumps are automatically (requiring no operator action) started by a station blackout signal during a loss of offsite power event. The AIT opened the URI because additional NRC review was required to determine if the behavior of RCP seal cooling following a loss of offsite power event is consistent with the description provided by the licensee in NUREG 0737 correspondence and if any differences represent a violation.

On March 28, 2010, Robinson Unit-2 experienced two fires events in 4160 volt switchgear, a reactor trip, and a safety injection. During the course of events E-2 bus was lost which caused a loss of power to FCV-626, CCW from RCP thermal barrier isolation. FCV-626 remained as is, i.e. stayed full open. Upon re-energization of the E-2 bus from the “B” EDG, FCV-626 unexpectedly closed due to the motor operator re-energizing shortly before the control power circuit. This is a result of the motor operator being powered directly from MCC-6 and the control power being powered via an inverter to instrument bus 4. Therefore, it was concluded that the automatic closure of FCV-626 due to the conditions associated with the loss of power and restoration of power to E-2 was an unintentional outcome of the way the system was designed. This condition was entered in the licensee’s corrective action program as NCR 391995 and it was corrected prior to the unit restart from the outage. The licensee modified the control power to ensure FCV-626 remains open following a loss of offsite power and subsequent re-energization.

The licensee’s failure to provide an accurate response to the NUREG 0737 on correspondence dated May 31, 1983, regarding the CCW system being operable in an offsite power event without requiring any operator action was a performance deficiency. A minor violation of 10 CFR 50.9(a) was identified for failure to provide accurate information in the licensee’s response to the NRC of NUREG 0737. In that, the licensee stated that the CCW system will automatically function during a loss of offsite power event (with no operator action). This issue was dispositioned as traditional enforcement, instead of the Significance Determination Process, because it had the potential for impacting the NRC’s ability to perform its regulatory function. Enforcement Policy, Rev. 6, Section IX, “Inaccurate and Incomplete Information”, states in part, if the initial submittal was accurate when made but later turns out to be erroneous because of newly discovered information, a citation normally would not be appropriate if, when the new information became available, the initial submittal was corrected. This violation was determined to be minor because the licensee’s initial submittal was reasonably accurate based on the information and understanding of how the system works that was available at the time. Once the new information was discovered, the licensee took prompt actions to modify the control circuitry of FCV-626 to restore its intended function. After review of this issue and the Enforcement Policy, NRC management determined that this was a minor violation and it will not be subject of formal enforcement action.

URI 05000/261/2010009-12, NUREG 0737 Response from Licensee to the NRC Describing the Behavior of RCP Seal Cooling Following a Loss of Offsite Power Event, is closed.

.16 <u>(Closed) URI 05000261/2010009-13, Dedicated Shutdown Diesel Generator Failed to Start Due to Low Starting Air Pressure</u>

As part of the Augmented Inspection of the events of March 28, the Inspectors identified the Failure of the Designated Shutdown Diesel Generator (DSDG) to start as Unresolved Item (URI) 05000261/2010009-13. On March 28, 2010, the DS bus was automatically de-energized, as designed, due to undervoltage on 4kV Bus 3. As a result, the DSDG support equipment, such as the starting air system compressor and battery charger, lost power. Based in part of adequate starting air pressure, the licensee considered the DSDG available for the purpose of assessing on-line risk. The log reading normal minimum value for starting air pressure is 165 psig and operators were monitoring this parameter twice per day. On March 31, 2010, the licensee attempted to start the DSDG and re-energize the DS bus to maintain adequate DSDG support parameters such as starting air pressure and battery voltage. Starting air pressure had decreased to 100 psig and the DSDG did not start. On April 1, 2010, the licensee successfully started the DSDG by pressurizing the DSDG starting air receiving tank using high pressure air bottles. The AIT opened the URI because additional NRC review was required to determine if the DSDG was available when credited in the licensee's risk assessment during the plant cooldown to Mode 4.

On July 11, 2010, the inspectors observed the implementation of special procedure, SP-1544, DSDG Air Start Test, which determined the minimum starting air system pressure needed to start the DSDG. The licensee used this procedure to show that the DSDG will start at a starting air system pressure of less than 130 psi as measured on PI-73, which is the starting air pressure gauge used by operators during log checks. Results of the test were that the DSDG was able to start at an air pressure of 120 psi. The inspectors noted that for the time period the DSDG was credited in the licensee's risk assessment, the pressure in the air start system was logged as 130 psi or higher, thus the DSDG was available as required. Based on the actions taken by the licensee and the results of the special procedure, no performance deficiency was identified.

URI 05000261/2010009-13, Dedicated Shutdown Diesel Generator Failed to Start Due to Low Starting Air Pressure, is closed.

.17 <u>(Closed) URI 05000261/2010009-14, Unexpected Loss of Instrument Bus 3 for Two Minutes</u>

As a result of the NRC's Augmented Inspection Team review of the causes, safety implications, and the licensee's actions for an event that occurred on March 28, 2010, under inspection procedure Inspection Procedure 93800 (NRC Inspection Report 05000261/2010008), the inspectors opened an Unresolved Item to assess the adequacy of the licensee's troubleshooting efforts following the unexpected loss of Instrument Bus 3 for two minutes, and to determine whether any deficiencies exist. The team interviewed the Robinson staff and reviewed the licensee's investigation performed under AR 390070 to determine whether the cause investigation was properly performed and comprehensive. In addition, the team reviewed the licensee's cause investigation performed under AR 004 06834-01 to investigate the June 4, 2010 failure of the Inverter-B, to determine whether there were any common failure mechanisms with the March 28, 2010 loss of Instrument Bus 3. The team concluded that the licensee's investigation of the event was adequate and that no deficiencies existed. The licensee also performed an overhaul of the inverter and replaced all major components.

URI 05000261/2010009-14, Unexpected Loss of Instrument Bus 3 for Two Minutes, is closed.

.18 Periodic Resident Inspector Reviews of INPO Evaluations

The Inspectors and Branch Chief reviewed the Institute of Nuclear Power Operations (INPO) evaluation report dated February 2010. The report was reviewed to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

4OA6 Meetings, Including Exit

On July 15, 2010, the inspectors discussed the results of the Occupation Radiation inspection with Mr. Scott Sanders, Plant General Manager, and other responsible staff.

An exit meeting was conducted on September 23, 2010 to discuss the findings of the Operator Licensing inspection. The inspectors confirmed that no proprietary information was reviewed during this inspection.

On November 12**,** 2010, the resident inspectors presented the inspection results to Mr. R. Duncan and other members of his staff. The inspectors confirmed that proprietary information which was examined during the inspection was returned.

Attachment: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

C. Castell, Licensing Supervisor
J. Cole, Manager – Shift Operations
D. Corlett, Licensing
R. Duncan, Vice President
W. Farmer, Engineering Manager
W. Gurganious, Director – Training
B. Houston, Radiation Protection Superintendent
S. Howard, Operations Manager
J. Lucas, Nuclear Assurance Manager
E. McCartney, Vice President
J. McCrory, Risk Analyst
K. Moore, Lead Engineer
C. Morris, Maintenance Manager
J. Pierce, Fleet Area Manager – Operations Training
E. Roberts, Superintendent Operations Training
S. Saunders, Plant General Manager
K. Smith, Training Manager
D. Sunthankar, Simulator Support Lead
S. Wheeler, Outage & Scheduling Manager
B. White, Manager, Support Services – Nuclear

NRC personnel

R. Musser, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000261/2010004-04	AV	Failure to Establish an Adequate PATH-1 Emergency Operating Procedure (Section 4OA3.2)
05000261/2010004-05	AV	Failure to Correctly Implement a Systems Approach to Training for the Licensed Operator Requalification Program (Section 1R11.3)

Closed

05000261/2010009-02	URI	RCS Cooldown Rate Exceeds Technical Specifications 3.4.3 Limit (Section 4OA5.5)
05000261/2010009-04	URI	Fidelity of Plant-Referenced Simulator for Conduct of Component Cooling Malfunctions (Section 1R11.2)
05000261/2010009-05	URI	Corrective Action for Operating Crew Performance Issues (Section 1R11.3)
05000261/2010009-06	URI	Adequacy of Emergency Operating Procedure Background Documents (Section 4OA3.2)
05000261/2010009-07	URI	Loss of Seal Water Results in Failure of the “A” Main Condenser Vacuum Pump (Section 4OA5.10)
05000261/2010009-08	URI	Deficiencies in Non Safety-Related Cable Installation (4OA5.11)
05000261/2010009-10	URI	Failure of Charging Pump Suction Valves to Automatically Transfer Due to Errors in Implementing an Instrumentation Component Upgrade (Section 1R19)
05000261/2010009-11	URI	FCV 626, RCP Thermal Barrier Outlet Isolation CCW Valve, Unexpected Closure. (Section 4OA5.14)
05000261/2010009-12	URI	NUREG 0737 Response from Licensee to the NRC Describing the Behavior of RCP Seal Cooling Following a Loss of Offsite Power Event (Section 4OA5.15)
05000261/2010009-13	URI	Dedicated Shutdown Diesel Generator Failed to Start Due to Low Starting Air Pressure. (Section 4OA5.16)
05000261/2010009-14	URI	Unexpected Loss of Instrument Bus 3 for Two Minutes (4OA5.17)

Opened & Closed

05000261/2010004-01	FIN	Failure to Have Adequate Work and Post Maintenance Testing Instructions for the Volume Control Tank Comparator Module (Section 1R19)
05000261/2010004-02	FIN	Failure to Design and Implement a Simulator Model that Demonstrated Reference Plant Response (Section 1R11.2)
05000261/2010004-03	NCV	Deficiencies in Non Safety-Related Cable Installation Result in Fire and Reactor Trip. (Section 4OA5.11)

Discussed

05000261/2010009-01	URI	Monitoring of Plant Parameters and Alarms. (Section 4OA5.4)
05000261/2010009-03	URI	Utilization of Operators During Events Requiring Use of Concurrent Procedures (Section 4OA5.6)
05000261/2010009-09	URI	Failure to Repair Circuit Breaker 52/24 Resulting in Breaker Being Unable to Operate (Section 4OA5.12)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Partial System Walkdown

Procedures
OP-201, Residual Heat Removal System, Rev. 62
OP-604, Diesel Generators “A” and “B”, Rev. 82
OP-909, Fuel Oil System, Rev. 42
OP-306, Component Cooling Water System, Rev. 63

Complete System Walkdown

Procedures
OP-604, Diesel Generators “A” and “B”, Rev. 79
OP-306, Component Cooling Water System Rev. 62
AOP-014, Component Cooling Water System Malfunction, Rev. 26

Other documents
HB Robinson Unit 2 Technical Specifications
G-190204-A, Emergency Diesel Generator System Flow Diagram, sheet 1 Rev. 32
Sheet 2 Rev. 18, Sheet 3 Rev. 19
5379-376, Component Cooling Water System Flow Diagram, sheets 1-4, Rev. 39
G-190199 Flow Diagram Service and Cooling Water, Rev. 19
G-190204-D Fuel Oil System Flow Diagram, sheet 2, Rev. 23
SD-005 Emergency Diesel Generators, Rev. 15
SD-043 Diesel Generator CO2 Fire Suppression System, Rev. 1

Section 1R05: Fire Protection

Procedures
OMM-003, Fire Protection Pre-Plans/unit No.2, Rev. 56

Drawings
HBR2-11937 Rev.0 Sheet 36, Fire Pre-plan Control Room
HBR2-11937 Rev 4, Sheet 46, Fire Pre-plan Turbine Building/Ground Level
HBR2-11937 Rev 1, Sheet 58, Fire Pre-plan Turbine Building/Mezzanine Level
HBR2-11937 Rev 2, Sheet 10, Fire Pre-plan “A” Diesel Generator Room
HBR2-11937 Rev 1, Sheet 8, Fire Pre-plan, Component Cooling Pump Room

Section 1R06: Flood Protection Measures

Procedures
AOP-022, Loss of Service Water, Rev. 34
AOP-014, Component Cooling Water System Malfunction, Rev. 28

Other documents
RNP-F/PSA-0009, Assessment of Internally Initiated Flood Events, Rev. 1

Section 1R11: Licensed Operator Requalification

OMM-001-1, Operations Unit Organization and Administration, Rev. 38
OMM-001-2, Shift Routines and Operating Practices, Rev. 65
OMM-001-5, Training and Qualification, Rev. 43
OMM-001-6, Operations Assessments, Rev. 27
OMM-001-19, Standards for Operations Department Continuous Improvement, Rev. 0
OPS-NGGC-1000, Fleet Conduct of Operations, Rev.3
Robinson Significant Adverse Condition Investigation Report, AR 390095, Rev. 11.
Robinson Adverse Condition Investigation – Equipment Report, Extent of Condition, AR 394584.
H. B. Robinson Steam Electric Plant – Augmented Inspection Team Report 05000261/2010009, July 2, 2010.
TPP-200, Licensed Operator/Shift Technical Advisor Continuing Training Program, Rev. 14
TPP-206, Simulator Program, Rev. 17.
TAP-409, Conduct of Simulator Training and Evaluation, Rev. 24.
TAP-413, Simulator Scenario Based Testing, Rev. 4.
TRN-NGGC-0002, Performance Review and Remedial Training, Rev. 0
Westinghouse Owner's Group Emergency Response Guidelines, Rev. 2.
NUREG-1021, "Operator License Examination Standards for Power Reactors, Rev. 9, Supplement 1.
Dynamic Simulator Scenario DSS-026, Rev. 6.
Nuclear Oversight Reports from 2007 through 2010.
Individual and Crew Training Records for Operators on Duty March 28, 2010.
PATH-1, Emergency Operating Procedure, Rev. 18
PATH-1, Introduction Powerpoint Presentation, Rev. 18.
PATH-1, Immediate Actions Powerpoint Presentation, Rev. 18.
PATH-1, LOCT Powerpoint Presentation, 03/20/2008.

Section 1R12: Maintenance Effectiveness

Procedures
PM-479, Motor Testing, Rev. 6
PDM-001, Equipment Lube Oil Sampling, Rev. 71
OP-202, Safety Injection and Containment Vessel Spray System, Rev. 85
OST-155, Safety Injection System Integrity Test, Rev. 31
OST-163, Safety Injection Test and Emergency Diesel Generator Auto Start on Loss of Power and Safety Injection, Rev. 56

Work Orders
1735213, Meggar and Bridge Condenser Vacuum Pump "A" Motor
1758168, "A" Main Condenser Vacuum Pump Inspection

Action Requests
3900027-28, Condenser Vacuum Pump "A" Motor Fire
404094, C Safety Injection Pump Appears Mechanically Bound
390072, Safety Injection Automatically Initiated
379420, OWP-016, Safety Injection System, Rev. 47

406514, OST-154, Safety Injection System High Head Check Valve Test
417136, OST-151-6, Comprehensive Flow Test Safety Injection Pump "C"
398259, Safety Injection Valves Needing Formal Abandonment or Test P
416030, OP-202, Safety Injection and Containment Vessel Spray System
416455, OST-163, Safety Injection Test Modification, Rev. 56

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Procedure OMM-048, Work Coordination and Risk Assessment, Rev. 44
ADM-NGGC-0006 Online EOOS Models for Risk Assessment, Rev. 7

Action Requests
422546, "A" EDG failed to reach rated Voltage during OST-409-1

Section 1R15: Operability Evaluations

Procedures
OST-163, Safety Injection Test and Emergency Diesel Generator Auto Start on Loss of Power and Safety Injection (Refueling), Rev. 48
GP-007, RCS and PZR Cooldown Data Table Attachment 10.1, Rev. 81

Action Requests
410777, SDAFW Pump Governor Hunting Excessively
411758, PCV-456, PZR PORV, Leak by Resulting in ITS Entry
391995, FCV-626 Closed Automatically on Auto Start of the CCW Pumps
395264, RCS Cooldown Exceeds Technical Specification Limit
392245-08, Review of MSR isolation valve control circuit

Other documents
NRC Information Notice 95.04, Supplement 1, Excessive Cooldown and Depressurization of the Reactor Coolant System Following Loss of Offsite Power

Section 1R19: Post Maintenance Testing

Procedures
SP-1551, Testing of 4KV Bus 5 to 4Kv Bus 4 Cubicle 24 and Cubicle 21 Pass through Circuitry, Rev 0
OST-401-1, EDG A Slow Speed Start, Rev. 44
OP-604, Diesel Generators "A" and "B", Rev. 82
CM-041, Conde Gastite Vacuum Pump Maintenance, Rev. 0
CM-608, Alignment and Adjustment of Belt Driven Equipment, Rev. 12
OST-021, Daily Surveillance, Rev. 29
OST-924-1, Process Radiation Monitoring System (Quarterly), Rev. 22
CM-750, Hand Operated ITT Grinnell Diaphragm Valve Procedure, Rev. 10

Work Orders
1782148, 1753110, 1811689, 1709634, 1709604

Action Requests
419481, Breaker 52/17B, Failed to Close in the Test Position
417562, During MST-021, Relay SRB-2(B) was not Properly Energized as Required

Other documents
EC 76842
Annunciator Panel Procedures APP-003, RCS & Makeup Systems, and APP-001, Miscellaneous NSSS
System Description SD-021, Chemical and Volume Control System
Design Basis Document, DBD/R87038/SD-21, Chemical and Volume Control System
UFSAR section 9.3.4, Chemical and Volume Control System
TS 3.4.17 and associated bases
NRC IR 05000261/2010009
Operator post event written statements
Action Request 390095
Drawing B-190628, Sheet 198, Control Wiring Diagram RWST to Charging Pump Suction Header Valve LCV-115B
Drawing B-190628, Sheet 160, Control Wiring Diagram LCV-115C Volume Control tank Discharge
License Amendment 176, Conversion to ITS
CR 98-01122
EE 92-144, Upgrade of Hagan Control and Protection System Modules
Action Requests generated from this inspection: 00420733, Inconsistencies between the CVCS DBD, system description, and UFSAR

Section 1R20: Refueling and Outage Activities

Procedures
GP-004, Post Trip Stabilization, Rev. 14
GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown, Rev. 82
GP-009-1, Filling the Refueling Cavity with Fuel in the Reactor Vessel, Rev. 15
GP-009-2, Filling the Refueling Cavity or Reactor Vessel with Reactor Defueled, Rev. 9
GP-009-5, Adjusting Reactor Vessel Level After Refueling Cavity Drain with Fuel in the Reactor, Rev. 3
GP-010, Refueling, Rev. 71
GP-002, Cold Shutdown to Hot Subcritical at No Load Tavg, Rev. 112
GP-001, Fill and Vent of the Reactor Coolant System, Rev. 52
OMP-001, Outage Scheduling, Rev. 15
POP-002, Planned Outage Implementation, Rev. 21
OMP-003, Shutdown Safety Function Guidelines, Rev. 43
OMA-NGGC-0203, Shutdown Risk Management, Rev. 0
SP-1535, Setup and Operation of the SFP Alternate Cooling System, Rev. 15

Other documents
Operating Logs
Shutdown Risk Profiles

Section 1R22: Surveillance Testing

Procedures
SP-1544, DSDG Air Start Test, Rev 0
EST-010, Containment Personnel Airlock Leakage Test (Semiannual), Rev. 30
EST-050, Refueling Startup Procedure, Rev. 48
OST-112-3, Reverse Flow Testing of CVC-298C and CVC-298F, Rev. 0
OST-051, Reactor Coolant Leakage Evaluation (Every 72 Hours During Steady State Operation and Within 12 Hours of Reaching Steady State Operation), Rev. 29
OST-409-1, EDG “A” Fast Speed Start, Rev. 45
EST-125, Emergency Diesel Generator Automatic Voltage Regulator Dynamic Response Test, Rev. 3

Work Orders
1068839, Calibrate the DSD Pressure Instruments
1825903, “A” EDG failed to reach rated Voltage during OST-409-1

Action Requests
422546, “A” EDG failed to reach rated Voltage during OST-409-1

Other documents
EC 78424

Section 1EP2: Alert and Notification System Testing

Procedures
EPPRO-07, Operation and Maintenance of the Alert and Notification System, Rev. 8
NGGM-IA-0038, Carolinas – Nuclear Generation Group Siren Maintenance, Rev. 1
WPS-2900 Series High Power Voice and Siren System, Operating and troubleshooting Manual

Records and Data
ANS Weekly Rotation Test Reports
ANS Quarterly Growl Test Reports
ANS Annual Full Volume Test Reports
FEMA Siren Approval Letter
Special Needs Consideration List

Section 1EP3: Emergency Preparedness Organization Staffing and Augmentation System

Procedures
EPPRO-03, Training and Qualification, Rev. 30
EPPRO-05, Scenario Development and Drill Control Guidelines, Rev. 19
EPEOF-00, Activation and Operation of the Emergency Operations Facility, Rev. 16
EPTSC-00, Activation and Operation of the Technical Support Center, Rev. 13
EPOSC-00, Activation and Operation of the Operational Support Center, Rev. 20

EMG-NGGC-0004, Maintenance of the Emergency Response Organization Notification System, Rev. 0
EMG-NGGC-0005, Activation of the Emergency Response Organization Notification System, Rev. 1

Records and Data
08/12/2010 Augmentation Drill
06/22/2010 ERO Communication Drill
03/23/2010 ERO Communication Drill
12/09/2009 ERO Communication Drill
09/08/2009 ERO Communication Drill

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Procedures
REG-NGGC-0010, 10 CFR 50.59 and Selected Regulatory Reviews, Rev. 14

Records and Change Packages
PLP-007, Robinson Emergency Plan, Rev. 72
PLP-007, Robinson Emergency Plan, Rev. 73
PLP-007, Robinson Emergency Plan, Rev. 74
EPNOT-01, CR/EOF, Emergency Communicator, Rev. 32
EPNOT-01, CR/EOF, Emergency Communicator, Rev. 33
EPNOT-01, CR/EOF, Emergency Communicator, Rev. 34
EPNOT-01, CR/EOF, Emergency Communicator, Rev. 35
EPCLA-01, Emergency Control, Rev. 29
EPCLA-01, Emergency Control, Rev. 30
EPRAD-03, Dose Projections, Rev. 24
EPRAD-03, Dose Projections, Rev. 25
EPRAD-03, Dose Projections, Rev. 26
EMG-NGGC-0002, Off-Site Dose Assessment, Rev. 1

Condition Reports (CRs)
NCR 424006, implementing procedure evaluation and forwarding requirements

Section 1EP5: Correction of Emergency Preparedness Weaknesses

Procedures
CAP-NGGC-0200, Condition Identification and Screening Process, Rev. 33
CAP-NGGC-0201, Self-Assessment/Benchmark Programs, Rev. 13

Audits and Self-Assessments
R-EP-10-01, 2010 Assessment of Emergency Preparedness
R-EP-09-01, 2009 Assessment of Emergency Preparedness
02/16/2010 Drill Rollup
11/18/2009 Drill Rollup
09/14/2009 Drill Rollup
06/17/2009 Drill Rollup

05/19/2009 Drill Rollup
04/14/2009 Drill Rollup
08/12/2010 Augmentation Drill Report
06/22/2010 ERO Communication Drill Report
03/23/2010 ERO Communication Drill Report
12/09/2009 ERO Communication Drill Report
09/08/2009 ERO Communication Drill Report
Self Assessment 293345, EPLAN and Implementing Procedures

Condition Reports (CRs)
NCR 377160, ERO change management and communication
NCR 377158, ERO drill reports and critiques not sufficient
NCR 315466, ERO training and qualification
NCR 315471, corrective action program rigor and corrective action timeliness
NCR 315475, ERO drill/exercise reports do not support performance improvement
NCR 315477, inconsistent documentation of emergency preparedness activities
NCR 315479, change management lacking or not implemented correctly
NCR 315481, operating experience with potential impact not identified
NCR 315484, poor communications to ERO
NCR 420559, selective signaling phones
NCR 420455, ENMON vehicle not available
NCR 417013, ENMON air particulate monitor blocked
NCR 397045, late declaration
NCR 382880, notification process difficulties

Section 2RS8: Radioactive Material Processing and Transportation

Procedures, Manuals, and Guides
HPS-NGGC-0001, “Radioactive Material Receipt and Shipping Procedure”, Rev. 29
HPP-007, “Handling and Storage of Contaminated and Radioactive Materials”, Rev. 34
HPP-259, “Spent resin Transfer to Waste Processing Containers Using the Self-Engaging Dewatering System (SEDS)”, Rev. 1
HPP-260, “Dewatering Procedure for Energy Solutions 14-215 or Smaller Liners, Utilizing the Self-Engaging Dewatering System (SEDS)”, Rev. 0
OP-704, “Spent Resin Storage Tank”, Rev. 41
Technical 3002, “Cask Handling Procedure for US DOT Specification 7A, Type A Transportation Cask”, Rev. 5
Process Control Program (PCP), Rev. 6
CAP-NGGC-0200, “Corrective Action Program”, Rev. 32

Shipping Records and Radwaste Data
Shipment 10-0061, Type A, Airborne Shipment
Shipment 10-0011, Dewatered Resin, >Type A Low Specific Activity
Shipment 09-0013, DAW, LSA
Shipment 09-0010, Dewatered Resin, Low Specific Activity
Shipment 09-0057, Dewatered Filters, Low Specific Activity
Radiological Survey M060410-7, IF-300 Rail Car
Radiological Survey 121301-6, Boric Acid Evaporators

10 CFR 61 Analyses, DAW, 11/14/07 and 11/3/08
10 CFR 61 Analyses, RCS Filters, 5/23/07 and 1/5/09
10 CFR 61 Analyses, Spent Resin Storage Tank Resin, 4/22/08 and 11/17/09
2009 Annual Radioactive Effluent Release Report

CAP Documents
R-RP-08-01, Robinson Nuclear Plant Radiation Protection Assessment Report
R-RP-09-01, Assessment of Radiation Protection
AR 410266, Contamination found on rail car stored outside
AR 290313, Radioactive shipment sent with incorrect pages from the DOT Emergency Response Guidebook
AR 366690, Water spray from valve WD-3346 during resin transfer

Section 4OA1: Performance Indicator Verification

Procedures
REG-NGGC-0009, NRC Performance Indicators and Monthly Operating Report Data, Rev. 10
EPPRO-04, EP Performance Indicators, Rev. 18
EPNOT-01, CR/EOF Emergency Communicator, Rev. 35
EPCLA-04, Emergency Action Level Technical Bases Document, Rev. 2

Other documents
NEI 99-02, Regulatory Assessment Performance Indicator Guidance, Rev. 6

Records and Data
Documentation of Performance Indicator data April 1, 2009, to June 30, 2010, for DEP, ANS, and ERO

Section 4OA2: Identification and Resolution of Problems

Procedures
CAP-NGGC-0200, Corrective Action Program, Rev. 33
CAP-NGGC-0206, Corrective Action Program Trending and Analysis, Rev. 5
SP-1544, DSDG, Air Start Test
OPS-NGGC-1000, Fleet Conduct of Operations, Rev. 3
OMM-001-2, Shift Routines and Operating Practices, Rev. 65

Work Orders
1068839-01, Calibrate the DSD Pressure Instruments
1794155, HCV-1459 Has Air Issuing from the Positioner Vent Port

Action Requests
390954, Risk Assessment During Plant Transient
390958, Dedicated Shutdown Diesel Generator Failed to Start
401309, Unplanned SI Signal Received While Restoring Safeguards
406852, BAST Bubbler Lines Cause Frequent Indicator Problems
413124, HCV-1459 Positioner Air Leak

Other documents
Drawing B190628, Control Wire Diagram
Drawing DSDG-FIGURE-11, One-Line Diagram of “DS” Electrical System

Section 4OA3: Event Follow-up

Procedures
Path 1, Rev. 18
EPP-4, Reactor Trip Response, Rev. 25
GP-004, Post Trip Stabilization, Rev. 15

Action Requests
421107, ‘B’ Pressurizer Backup Heaters required reset to operate
421110, Trip of the ‘B’ Heater Drain Pump
421111, Trip of the ‘B’ Main Feed Pump
421149, Opening of Pressurizer PORV PCV-455C during reactor trip response

Section 4OA5: Other Activities

Procedures
OST-021, Daily Surveillance, Rev. 29
FMP-004, Special Nuclear Material (SNM) Inventory, Rev. 24
AOP-028, [Independent Spent Fuel Storage Installation] Abnormal Events, Rev. 7
FHP-003, Fuel Assembly Movement in the Spent Fuel Pit, Rev. 37

Licensing Bases Documents
H.B. Robinson, Unit 2 Updated FSAR, Rev. 15

Design and Engineering Documents
B/M 851-E-3001, Bill of Material for Design Change 851, dated 6/28/85
DCN No. 581-6, Switchgear Wiring Revisions, dated 2/14/86
L2-E-035, Specification for 5,000 Volt Power Cable, Rev.1
NGG-PMB-SWG-01, NGG Reliability Template Medium and Low Voltage Switchgear, Rev 0
EGR-NGGC-005, Engineering Change, Rev. 31
RNP-E-8.004, Attachment L, Page L1, Rev 0, Specification 7155 by Rome Cable Corporation, dated 1/1/91
Engineering Change 50856R2, Original Cable Specification, dated 3/21/68
Medium Cable Voltage Survey prepared ad part of plant life extension documentation
RNP2, 4160 Volt Cable Identification List, Rev 0

Event Investigation Documents
PDS-ER-02, Event Review Team Field Notes, dated 4/10

Condition Report Reviewed
AR 39005, Significant Adverse Condition investigation Report