U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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License No:	DPR-23
Report No:	50-261/99-01
Licensee:	Carolina Power & Light (CP&L)
Facility:	H. B. Robinson Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	January 17 - February 27, 1999
Inspectors:	B. Desai, Senior Resident Inspector A. Hutto, Resident Inspector W. Bearden, Reactor Inspector (Sections M2.1, M2.2,M2.3) M. Ernstes, Reactor Engineer (Sections O5.1, O5.2, O5.3, O5.4)
Approved by:	Brian R. Bonser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

H. B. Robinson Power Plant, Unit 2
NRC Integrated Inspection Report 50-261/99-01

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a six-week period of resident inspection. In addition, it includes the results of inspections by a region based reactor inspector and an operator license examiner.

Operations

- The conduct of operations was professional, risk informed, and safety-conscious (Section O1.1).

- A system walkdown found that the auxiliary feedwater system was appropriately aligned, component labeling and housekeeping were adequate (Section O2.1).

- A clearance associated with emergency diesel generator maintenance provided adequate isolation conditions for personnel safety and protection of plant equipment. The clearance was implemented and restored in accordance with the licensee's procedures (Section O2.2).

- The conduct and performance of the simulator examinations were satisfactory. The facility evaluators were thorough in noting individual performance discrepancies and the scenarios observed were effective in determining areas in need of retraining (Section O5.1).

- Job performance measures adequately tested operators ability to perform tasks using the licensee procedures (Section O5.2).

- The majority of the biennial written examination questions met the guidelines of NUREG-1021, Examiner Standards, and facility training procedures, however some questions had flaws which diminished their effectiveness in evaluating operator knowledge. Overall, the exam was considered valid (Section O5.3).

- The licensee conducted remedial training and evaluations as required by 10CFR 55.59 and facility training procedures. Operators that had failed requalification tests and quizzes were removed from shift until remediation was complete (Section O5.4).

- The onsite review functions of the Plant Nuclear Safety Committee (PNSC) were conducted in accordance with Technical Specifications. During the PNSC meetings topics were thoroughly discussed and evaluated (Section O7.1).

Maintenance

- Maintenance activities were conducted in accordance with applicable work documents and procedures. Personnel were properly trained and knowledgeable of their assignments (Section M1.1).

- Completed surveillance test documentation reviewed demonstrated acceptable test results (Section M2.1).

- The program for maintenance and testing of pressure isolation valves (PIVs) satisfied Technical Specification requirements. Leakage testing of two PIVs was not required and not included in the licensee's in-service testing program. There were no examples of inadequate maintenance or examples that would indicate an adverse trend or degradation in the material condition of reactor coolant system PIVs. Review of leakage testing data indicated good material condition of these isolation boundaries (Section M2.2).

- The program for testing of ASME Section XI Class 2 and 3 relief valves met requirements. The initiatives for increased testing frequency of certain relief valves demonstrated a positive safety culture (Section M2.3).

- On line emergency diesel generator maintenance observed was conducted in accordance with procedures.

Plant Support

- Radiological controls and security practices were properly conducted. Areas observed in the radiological control area were appropriately posted and secured. The security plan was effectively implemented and compensatory actions were initiated when required (Section R1.1, S1.1).

- Human error during a resin sluice caused displacement of resin into the auxiliary building drains and sump. The cleanup and resin recovery efforts resulted in personnel dose of approximately 325 millirem. All contaminated floor space was promptly decontaminated (Section R1.2).

- Emergency diesel generator maintenance observed was conducted in accordance with procedures (Section M2.4).

Report Details

Summary of Plant Status

Robinson Unit 2 operated at 100 percent power for the entire report period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors conducted frequent control room tours to verify proper staffing, operator attentiveness and communications, and adherence to approved procedures. The inspectors routinely attended operations turnover meetings, management review meetings, and plan-of-the-day meetings to maintain awareness of overall plant operations. Operator logs, Condition Reports (CR), and instrumentation were routinely reviewed. Plant tours were conducted to verify operational safety and compliance with Technical Specifications (TS), as well as to assess plant housekeeping. In general, the inspectors concluded that the conduct of operations was risk informed, professional, and safety-conscious.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdown

a. Inspection Scope (71707)

The inspectors conducted a system walkdown of the steam driven auxiliary feedwater (SDAFW) system.

b. Observations and Findings

The inspectors conducted a system walkdown of the SDAFW system to assess the general condition of system components, including labeling, to verify that system valve positions matched the system drawings and station operating procedures, and to assess plant housekeeping around system components. The inspectors also reviewed the system notebook maintained by the SDAFW system engineer. This included a review of maintenance rule performance indicator data trending.

The inspectors determined that the SDAFW system was properly aligned for accident conditions. Housekeeping and component labeling were adequate. A few minor discrepancies between system drawings and actual conditions were noted and brought to the attention of the licensee, for which CRs were promptly initiated by the licensee. The inspectors also reviewed the applicable sections of the Updated Final Safety Analysis Report (UFSAR) and identified no discrepancies.

c. Conclusions

A system walkdown found that the SDAFW system was appropriately aligned, component labeling and housekeeping were adequate.

O2.2 Clearance Walkdown

a. Inspection Scope (71707, 62707)

The inspectors performed a walkdown of a system clearance and reviewed other active clearances.

b. Observations and Findings

The inspectors verified proper implementation of clearance, 99-00067, during a walkdown on January 22. This clearance was established to support scheduled maintenance on the "A" emergency diesel generator (EDG). The inspectors verified that valves as well as electrical breakers were aligned appropriately to provide an adequate boundary for the scheduled maintenance activity. No discrepancies were identified during verification of the clearance. Upon removal of the clearance, the inspectors verified the restoration positions of the affected components with the required positions established by system restoration procedures and system drawings.

c. Conclusions

A clearance associated with EDG maintenance provided adequate isolation conditions for personnel safety and protection of plant equipment. The clearance was implemented and restored in accordance with the licensee's procedures.

O5 Licensed Operator Requalification Program Evaluation

The inspectors conducted a routine, announced inspection of the licensed operator requalification program during the period January 25-29. Specific areas of review included simulator examinations, job performance measures (JPMs), written examinations, and operator remedial training. The inspectors found the requalification program to be satisfactory.

O5.1 Simulator Examinations

a. Inspection Scope (71001)

The inspectors observed the licensee's conduct of annual simulator examinations on January 26, 1999. The licensee training department staff evaluated two crews of licensed operators. The inspection covered operator performance, and evaluated the licensee's effectiveness in conducting operator requalification evaluations in accordance with 10 CFR 55.59, "Requalification."

b. Observations and Findings

The inspectors observed the administration of Dynamic Simulator Scenario (DSS)-002 and DSS-04. Each scenario was administered to two separate crews of licensed operators. Both scenarios met the NRC requirements for an annual simulator evaluation of the licensed operators. The inspectors found that both scenarios were challenging and discriminating test tools that were appropriate for measuring the knowledge and skill of the operators.

Both crews adequately mitigated the events presented to them in the scenario using the appropriate plant procedures. The inspectors did not note any major competency weaknesses.

The inspectors observed the facility evaluators' debrief sessions and reviewed the evaluators' documentation of the crews' performance. The licensee's evaluators were critical of the operators performance and effectively identified areas for improvement. The evaluators' comments and findings were appropriate and agreed with NRC observations.

c. Conclusions

The conduct and performance of the simulator examinations were satisfactory. The scenarios observed were determined to be adequate evaluation tools. The facility evaluators were critical of the operators performance and thorough in noting individual operator performance discrepancies. Documentation of individual performance results was satisfactory. This portion of the licensed operator requalification program met the requirements of 10 CFR 55.59, "Requalification."

O5.2 Job Performance Measures (71001)

The inspectors reviewed the development and administration of the JPMs. The JPMs adequately tested the operators ability to perform tasks using the licensee procedures. The licensee evaluators satisfactorily administered and documented the JPMs. The inspectors concluded that this portion of the licensed operator requalification program met the requirements of 10 CFR 55.59, "Requalification."

O5.3 Biennial Requalification Written Examination

a. Inspection Scope (71001)

The inspectors reviewed the licensee's biennial requalification written examination administered on January 27, 1999, to 12 licensed operators to determine if it met the requirements of 10 CFR 55.59 and licensee procedure Training Administrative Procedure (TAP)-403, "Examination and Testing," Revision 7.

b. Observations and Findings

The reactor operator (RO) and senior reactor operator (SRO) examinations were comprised of thirty open reference questions. Twenty-four questions were common to both exams resulting in 36 different questions. The inspectors determined that six questions contained psychometric flaws which diminished their effectiveness in evaluating operator knowledge.

The inspectors determined that two questions were "direct look ups" as defined in NUREG-1021, Examiner Standard (ES) - 602, Attachment 1, B.2.e. "Direct look ups" only require that the operator can locate the information and do not evaluate an operator's understanding of the topic. Three questions had implausible distractors. Distractors are the incorrect responses in a multiple choice test. These distractors could be easily eliminated as a possible answer with no knowledge of the subject matter. One question had two correct answers. In addition to the six flawed questions, five questions had grammar or sentence structure problems which made the questions difficult to understand.

c. Conclusions

The biennial requalification written examinations were adequate. The majority of the questions met the guidelines of NUREG-1021 and TAP-403, however some questions had flaws which diminished their effectiveness in evaluating operator knowledge. Overall, the exam was considered valid.

O5.4 Remedial Training and Testing

a. Inspection Scope (71001)

The inspectors reviewed the licensed operator requalification training records and associated procedures to ensure that an appropriate remedial training program was developed, implemented, and documented as required by 10 CFR 55.59 and TAP-402 , "Student Performance Review and Remedial Training," Revision 1.

b. Observations and Findings

The inspectors reviewed the documentation associated with one operator who passed but required remediation on the 1998 annual requalification examination, and one operator who failed an evaluation during a weekly training cycle examination. The documentation included a training plan which adequately addressed the areas identified in need of retraining and a re-evaluation exam.

c. Conclusions

The licensee conducted remedial training and evaluations as required by 10 CFR 55.59 and facility training procedures. Operators that had failed requalification tests and quizzes were removed from shift until remediation was complete.

O7 Quality Assurance In Operations

O7.1 Plant Nuclear Safety Committee and Nuclear Assessment Section Oversight (71707)

The inspectors periodically attended Plant Nuclear Safety Committee (PNSC) meetings during the inspection report period. The presentations to the committee were thorough and the presenters readily responded to all questions. The committee members asked probing questions and were well prepared. The committee members displayed an understanding of the issues. The inspectors also reviewed Nuclear Assessment Section audits and concluded that they were appropriately focused to identify and enhance safety. The inspectors concluded that the onsite review functions of the PNSC were conducted in accordance with TSs. During the PNSC meetings topics were thoroughly discussed and evaluated.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) LER 50-261/98-05-00, 01, 02: Reactor and Turbine Trip Caused by Feedwater and Steam Dump Control Problems: The circumstances surrounding the reactor trip that occurred on October 17,1998, were discussed in NRC Inspection Report 50-261/98-09. The licensee conducted an investigation of the reactor trip in a Significant Adverse Condition Evaluation and documented the results in CR 98-2352. This investigation determined that the initiating event was attributed to a failed control power supply in nuclear instrument channel NI 44. The steam dump valve failure was attributed to a bias potentiometer being out-of-position. The slow response of the "C" feedwater regulating valve (FRV) was due to an improper gain setting. The reactor trip highlighted some weaknesses in the Instrumentation and Control(I&C) calibration processes. The scope of completed and planned corrective actions was broad in that they incorporated circumstances and components beyond those that were directly related to the reactor trip. The licensee also solicited good practices from other plants with regard to post trip reviews. These corrective actions are tracked through the CR system and included changes to the I&C calibration process. The inspectors verified and discussed the status of the corrective actions with the licensee and determined that most of them were complete and the pending corrective actions were adequately tracked through the CR system.

O8.2 (Closed) LER 50-261/98-03-00, 01: Reactor Trip Due to Inadvertent Closure of Turbine Governor Valves: The circumstances surrounding the reactor trip that occurred on April 25, 1998, were discussed in NRC Inspection Report 50-261/98-05. The licensee determined that the reactor trip was most likely caused by a pressure spike sensed in the impulse pressure control of the turbine control system, causing the turbine governor valves to close. During the investigation following the reactor trip that occurred on October 17 (Section 08.1), the licensee determined that the steam dump valves had also failed to open just prior to the April 25 reactor trip. The cause of the steam dump valve failure to open was a mispositioned bias potentiometer. This occurred during calibration activities conducted in the steam dump control cabinet during refueling outage 18. The mispositioned bias potentiometer remained unnoticed until analysis of the October 17 reactor trip. The steam dumps are not credited in accident analysis and are considered non-safety related. Licensee corrective actions associated with the

October 17 trip were also applicable for this trip. Additionally, the exact cause of the turbine governor valves going closed could not be determined during the followup testing and troubleshooting.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Observation of Maintenance Activities (62707)

The inspectors observed all or portions of the following work requests (WR):

- WR No. JO 99-AARH1, RCS "C" T Cold Protection Converter Alarms received
- WR No. 98-ADTD1 on PT-475, Loop Calibration Procedure, LP-357 Rev. 9.

The inspectors determined that the maintenance observed was properly approved and was included on the plan of the day. The inspectors found that the work observed was thorough, and performed with the work package present and in use. Accompanying documents such as procedures and supplemental work instructions were properly followed. Personnel were properly trained and knowledgeable of their assignments. The inspectors noted that supervisors and system engineers monitored the jobs on a frequent basis.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Review of Completed Surveillance Test Packages (61726)

a. Inspection Scope

The inspectors reviewed selected completed periodic test packages to verify that the documentation satisfied the referenced TS Surveillance Requirements (SRs).

b. Observations and Findings

The inspectors reviewed test package documentation for the following recently completed surveillance tests:

- EST-096, "Safety Injection Swing Check Valve (SI-875A, B, C, D, E and F) Inspection (Refueling Shutdown)"
- EST 111, "Relief Valve Testing Type JRAK-BS/JMAK-BS (Refueling Shutdown and As Needed After Maintenance)"
- EST 112, "Relief Valve Testing Type JB (Refueling Shutdown and As Needed After Maintenance)"

- EST 113, "Relief Valve Testing Type JO (Refueling Shutdown and As Needed After Maintenance)"
- EST 115, "Relief Valve Testing Type JMAK-SP (Refueling Shutdown and As Needed After Maintenance)"
- EST 126, "Relief Valve Testing Anderson, Greenwood and Company Model 83F (Refueling Shutdown and As Needed After Maintenance)"
- EST-140, "Leak Test SI-864A & B and SI-856A & B (Refueling)"
- OST-160, "Pressure Isolation Valve Back Leakage Test (Refueling and Cold Shutdown Interval of Greater than 48 Hours Unless Performed in Preceding 9 Months)"
- OST-255 "RHR and SI System Check Valve Test (Refueling Interval)"

No problems were identified. The TS SR had been satisfied. Completed surveillance test packages demonstrated acceptable test results.

c. Conclusions

A review of nine completed surveillance test packages demonstrated acceptable test results.

M2.2 Maintenance/Material Condition of RCS Pressure Isolation Valves

a. Inspection Scope (62707)

The inspectors reviewed the licensee's program for maintenance and testing of reactor coolant system (RCS) pressure isolation valves (PIVs).

b. Observations and Findings

The inspectors reviewed machinery history and leak testing data for selected RCS PIVs to evaluate the adequacy of the program for maintaining the integrity of those RCS isolation boundaries and to verify that TS 3.4.14.1 requirements had been satisfied. Valves selected for review consisted of isolation valves, including check valves, which if failed could result in an interfacing system loss of coolant accident (IS-LOCA). The inspectors reviewed the surveillance procedures for periodic leak rate testing of PIVs and as-found leakage test data for selected valves from testing performed during the RO17 and RO18 refueling outages. Specific leakage test packages reviewed are listed in Section M2.1. The inspectors also reviewed selected maintenance procedures used by the licensee for disassembly and inspection of check valves as required by the inservice testing (IST) program.

The inspectors noted that each of the leakage testing procedures required that a corrected value for valve leakage be calculated for the RCS at 2235 psig. This corrected leakage value was required to be used rather than the actual observed leakage values anytime testing was performed at a lower test pressure.

The inspectors verified that the program for maintenance and testing of PIVs had satisfied the TS requirements. The inspectors determined that no as-found leakage testing failures had occurred during the previous two refueling outages. No examples of inadequate maintenance were identified during the review. No problems were identified during the review of machinery history which would indicate adverse trends or degradation of material condition of any RCS PIVs.

Although the IST program, as described in Robinson Technical Management Procedure, TMM-004, "Inservice Inspection Testing," included leakage testing of those RCS PIVs listed in the TS, the TS did not require leak testing PIVs residual heat removal (RHR)-750 and 751, RHR hot leg suction motor operated valves. No specific leakage criteria existed for those PIVs and the existing RHR design did not provide for a means of performing leakage testing. The inspectors determined that the licensee had relied on the RHR system high pressure alarm in the main control room and routine performance of an overall RCS water inventory balance in accordance with TS 3.4.13.1 to alert operators of excessive leakage. These valves, installed in series, were only opened at reduced RCS pressure to allow use of RHR for shutdown cooling. The valves were interlocked to prevent inadvertent opening during normal operation. Power was also normally removed from one of the valves.

Licensee PRA efforts quantified the IS-LOCA associated with the excessive reactor coolant system leakage past the seats of the RHR suction valves as of very low probability but, of high consequence. This conclusion is consistent with industry results of similar analysis. Excessive leakage past the RHR valves would cause an IS-LOCA, possibly disabling the emergency core cooling system designed to mitigate LOCAs and disabling the confining properties of the primary containment by creating a direct radiological release path to the environment. The licensee's Individual Plant Examination (IPE) submittal of 1992 indicated that failure of these valves was the dominate IS-LOCA contributor. The excessive leakage would flash to steam with two possible consequences - a pipe break adversely affecting refueling water storage tank inventory (the suction source for the emergency core cooling system (ECCS)) or steam binding all ECCS suction piping and pumps. The IPE submittal also reflected the containment bypass aspects of an IS-LOCA. More recent PRA analysis quantified the large early release frequency (an airborne fission product release to the environment prior to the implementation of protective actions under the emergency plan) for the Robinson facility at 8E-6/year with the IS-LOCA comprising 51% of this condition. The inspectors noted that licensee planned to perform a PRA model upgrade during the spring of 1999. That upgrade is scheduled to include an IS-LOCA re-analysis presenting an opportunity for the licensee to reevaluate this condition.

c. Conclusions

The program for maintenance and testing of PIVs satisfied TS requirements. Leakage testing of two PIVs was not required and not included in the licensee's IST program. There were no examples of inadequate maintenance or examples that would indicate an adverse trend or degradation in the material condition of RCS PIVs. Review of leakage testing data indicated good material condition of those RCS isolation boundaries.

M2.3 Testing of ASME Section XI Class 2 and 3 Relief Valves

a. Inspection Scope (62700)

The inspectors reviewed the program for testing ASME Section XI Class 2 and 3 relief valves to verify that the program satisfied requirements of ASME/ANSI OM-1987, "Operation and Maintenance of Nuclear Power Plants." Verification of correct lift setpoints for these relief valves was necessary to insure proper operation of emergency core cooling systems (ECCS) and because of the potential impact of improper lift setpoints on a postulated IS-LOCA event.

b. Observations and Findings

ASME Section XI Class 2 and 3 relief valves included a number of smaller relief valves in various systems such as safety injection (SI), RHR, and other systems. The inspectors reviewed documentation for selected ASME Class 2 and 3 relief valves in the chemical volume control system (CVCS), SI and RHR systems that had been tested during the RO17 and RO18 refueling outages. Specific relief valve test packages reviewed were documented in Section M2.1. Some as-found lift set point failures have occurred for relief valves but lift set points were readjusted whenever the as-found set point exceeded +/- 3% of nominal as required by ASME/ANSI OM-1987. The inspectors reviewed documentation for resetting lift setpoints for selected relief valves which had as-found lift setpoint failures. Additionally, the inspectors reviewed maintenance work packages and post maintenance test documentation for completed work on selected relief valves. No problems were identified during this review. The inspectors determined that the licensee had checked a sufficient number of relief valves to satisfy sampling requirements from ASME/ANSI OM-1987, Part 1, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." The inspectors also noted that the program for testing some relief valves involved more frequent testing than required by the Code. ASME/ANSI OM-1987 required testing of Class 2 and 3 relief valves such that at least 20% of each group (by vendor, valve model, and application) are tested during a 48 month period and all tested within 10 years. The program required testing most ASME Class 2 and 3 relief valves within the 10 year interval. However, certain relief valves had been tested more frequently based on failure history. Additionally, the licensee was testing certain valves every outage and some others were tested every other outage due to a higher safety significance given to the potential failure of those relief valves.

Based on reviewing the program for testing ASME Section XI Class 2 and 3 relief valves, the inspectors determined that the implementation of ASME/ANSI OM-1987 requirements was good. No problems were identified during this review.

c. Conclusions

The implementation of testing for ASME Section XI Class 2 and 3 relief valves met requirements. The initiatives for increased testing frequency of certain relief valves demonstrated a positive safety culture.

M2.4 "A" Emergency Diesel Generator (EDG) Maintenance Outage

a. Inspection Scope (62707, 37551)

The inspectors observed and assessed maintenance activities associated with the "A" EDG. The on-line maintenance included the semiannual, 18 month, and three year preventive maintenance procedures.

b. Observation and Findings

The licensee performed an on-line diesel maintenance outage for the "A" EDG during the week of January 25. This outage included the performance of preventive maintenance procedures PM-007, "EDG Semiannual Inspection," PM-008, EDG "Refueling Inspection," and PM-009, "EDG 3 Year Inspection."

The inspectors observed portions of the maintenance during the outage. The inspectors verified that the mechanics had the applicable work permits and procedures for the maintenance being performed. Calibration stickers on torque wrenches and multi-meters used during observed maintenance were checked to be current. The inspectors observed the performance of the jacket water hydrostatic test. Test procedures were performed correctly and inspection hold points observed. The acceptance criteria were documented with the appropriate quality control verification. The inspectors also inspected the general condition and configuration of the "B" EDG while the "A" EDG was declared out of service. No operability concerns were identified with the "B" EDG. The inspectors also observed portions of the emergent work and determined that the emergent issues were dealt with in accordance with plant procedures and efficiently implemented thus minimizing additional unavailability time for the diesel.

During the initial start up following the maintenance, a lube oil leak was observed on a coupling in the lube oil cooling piping. The coupling was tightened by a mechanic and the leak was stopped. However, on February 25, approximately two hours into the EDG run during performance of "A" EDG surveillance testing in accordance with OST-401-1, "EDG "A" Slow Speed Start," Revision 9, the leak occurred again and increased to a magnitude that a decision was made by the operators to shut-down the diesel.

Upon disassembling and inspecting the leaking coupling, the licensee determined that the gaskets were improperly positioned in the lip on the coupling, causing them to be pinched at the edges. This indicated that the gaskets were not properly seated, causing lube oil to leak. A significant CR (99-00447) was initiated to determine past operability

as well as to determine the root cause. The gaskets were replaced within the allowable TS action statement.

The inspectors questioned the past operability of the EDG considering the lube oil leak. At the end of the report period, the licensee had not completed the past operability determination or the root cause evaluation.

c. Conclusions

On line EDG maintenance observed was conducted in accordance with procedures.

M8 Miscellaneous Maintenance Issues (IP 92902)

M8.1 (Closed) IFI 50-261/97-14-01: Opening of RHR Valves FCV-605 or HCV-758 During Testing. During review of Post Maintenance Testing (PMT) activities following corrective maintenance on air operated valves HCV-758 and FCV-605, on the RHR system, the licensee had identified that PMT requirements involving valve stroking could potentially place the unit in TS 3.0.3.

These valves were required by TS 3.5.2.B to be maintained closed with the motive air isolated. Since the valves were not allowed to be opened in Modes 1, 2, and 3, the PMT involving valve stroking had been postponed until the next available opportunity (upcoming refueling outage). During the subsequent evaluation of this issue, the licensee identified that OST-251-1, "RHR Pump "A" and Components Test (Quarterly)," and OST-251-2, "RHR Pump "B" and Components Test (Quarterly)," required opening FCV-605 to accomplish RHR pump discharge check valve testing. The licensee had planned to evaluate the adequacy of these test instructions prior to the next required performance.

The inspector reviewed current versions of OST-251-1 and OST-251-2 and verified that the test instructions had been revised to address this issue. The inspectors determined that the licensee had revised those test procedures to delete opening FCV-605. Additionally, the inspector noted that these test instructions had not previously required opening HCV-758.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

The inspectors periodically toured the Radiological Control Area (RCA) during the inspection period. Radiological control practices were observed and discussed with radiological control personnel including RCA entry and exit controls, survey postings, locked high radiation area controls, and radiological area material condition. The inspectors concluded that radiation control practices were being conducted in accordance with procedures. The inspectors also toured the radwaste building and

found that radwaste storage containers and laundry bags were in good condition and appropriately labeled. In addition, outside radwaste storage areas and structures were properly posted and exhibited correct labeling and effective housekeeping. The inspectors found that housekeeping throughout the plant was effective in maintaining areas free of unnecessary equipment and debris. Relatively few contaminated areas were noted and posted locked high radiation areas were properly secured against unauthorized entry.

R1.2 Waste Water Demineralizer Systems (WWDS) Resin Spill

a. Inspection Scope (71750)

The inspectors reviewed and observed the circumstances that led to the WWDS resin spill, licensee efforts to cleanup and recover the resin, and licensee investigation and corrective actions related to the event.

b. Observations and Findings

On February 24, Pressure Vessel (PV) 4 associated with the WWDS system was sluiced to the Spent Media Storage Tank (SMST). The WWDS system is installed to process liquid drains within the auxiliary building. After the sluice was completed, the SMST was being dewatered in accordance with procedure CP-100, "Waste Water Demineralization System Operation and CVCS Hut Water Processing," Revision 22. During this time a radiological control technician noted resin coming out of the floor drains in the WWDS room and a security guard noted resin coming out of the auxiliary building floor drains. The dewatering was immediately stopped and health physics technicians began efforts to decontaminate the floor space. The licensee initiated CR 99-435, and assembled an event review team (ERT) to determine the root case. Additionally, the licensee formed a team to cleanup and recover resin from the drain system and auxiliary sump tank that was potentially transferred from the SMST.

The inspectors observed the decontamination efforts as well as the resin recovery efforts conducted by the licensee. All contaminated floor space was decontaminated by the licensee. Approximately 10 cubic feet of resin was recovered from the drain system and the sump. The cleanup recovery effort resulted in an exposure of approximately 325 millirem (mrem). There were no personal contamination events during the cleanup and recovery.

The event was attributed to human error. While the SMST was being dewatered, an adjacent valve on a hose connection was inadvertently bumped partially open. This caused the resin as well as the water to be pumped into the auxiliary building drains.

c. Conclusions

Human error during a resin sluice caused displacement of resin into the auxiliary building drains and sump. The cleanup and resin recovery effort resulted in personnel exposure of approximately 325 mrem. All contaminated floor space as a result of the event was promptly decontaminated.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

During the period, the inspectors toured the protected area and noted that the perimeter fence was intact and not compromised by erosion or disrepair. Isolation zones were maintained on both sides of the barrier and were free of objects which could shield or conceal an individual. The inspectors periodically observed personnel, packages, and vehicles entering the protected area and verified that necessary searches, visitor escorting, and special purpose detectors were used as applicable prior to entry. Lighting of the perimeter and of the protected area was acceptable and met illumination requirements.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on March 5, 1999. The licensee acknowledged the findings presented at the exit meeting. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Cleary, Manager, Operations
H. Chernoff, Supervisor, Licensing/Regulatory Programs
J. Clements, Manager, Site Support Services
R. Duncan, Manager, Robinson Engineering Support Services
J. Fletcher, Manager, Maintenance
J. Moyer, Director, Site Operations
R. Steele, Manager, Outage Management
T. Walt, Plant General Manager
R. Warden, Manager, Manager, Regulatory Affairs
A. Williams, Manager, Training
D. Young, Vice President, Robinson Nuclear Plant

NRC

B. Desai, Senior Resident Inspector
A. Hutto, Resident Inspector
M. Ernstes, Senior License Examiner
W. Bearden, Reactor Engineer

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71001:	Requalification Inspection
IP 71707	Plant Operations
IP 71750:	Plant Support Activities
IP 92901:	Followup Operations
IP 92902:	Followup Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-261/97-14-01	IFI	Opening of RHR Valves FCV-605 or HCV-758 During Testing (Section M8.1).
50-261/98-05-00	LER	Reactor and Turbine Trip Caused by Feedwater and Steam Dump Control Problems (Section O8.1).
50-261/98-05-01	LER	Reactor and Turbine Trip Caused by Feedwater and Steam Dump Control Problems (Section O8.1).
50-261/98-05-02	LER	Reactor and Turbine Trip Caused by Feedwater and Steam Dump Control Problems (Section O8.1).
50-261/98-03-00	LER	Reactor Trip Due to Inadvertent Closure of Turbine Governor Valves (Section O8.2).
50-261/98-03-01	LER	Reactor Trip Due to Inadvertent Closure of Turbine Governor Valves (Section O8.2).