U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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License No:	DPR-23
Report No:	50-261/98-09
Licensee:	Carolina Power & Light (CP&L)
Facility:	H. B. Robinson Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	September 13 - October 24, 1998
Inspectors:	B. Desai, Senior Resident Inspector A. Hutto, Resident Inspector M. King, Resident Inspector (Section O5.1) J. Kreh, Radiation Specialist (Sections P1.1, P8.1) J. Lenahan, Reactor Engineer (Sections E8.1-E8.11, E8.14) G. McDonald, Senior Project Engineer (Sections M8.1, E8.12, E8.13)
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Enclosure

EXECUTIVE SUMMARY

H. B. Robinson Power Plant, Unit 2
NRC Integrated Inspection Report 50-261/98-09

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 region based inspectors.

Operations

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

- Operator response to a turbine trip and automatic reactor trip was appropriate and in accordance with procedures. The licensee's event review was thorough and the findings identified the root causes of the event. The licensee appropriately generated Condition Reports to track planned corrective actions (Section O2.1).

- Two examples of operator inattention to detail were identified involving operators not recognizing a failed Emergency Response Facility Information System printer and the subsequent missed review of plant parameters; and documentation completeness during an emergency diesel generator surveillance (Section O4.1).

- Simulator examinations observed demonstrated the effectiveness of licensed operator requalification training. Simulator examination scenario guides were well developed and detailed. Operator performance was effectively assessed by the training department (Section O5.1).

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

Maintenance

- Maintenance observed on a service water pump discharge cross-connect valve breaker was completed satisfactorily and the acceptance criteria were met (Section M1.1).

- A Non-Cited Violation was identified involving two inadequate TS required surveillance procedures involving valve position verification. The affected valves were verified to be in their required position. Licensee corrective actions were appropriate (Section M2.2).

- A Maintenance and Test Equipment (M&TE) program review concluded that the program met the requirements of the Updated Final Safety Analysis Report (UFSAR) and was implemented in accordance with procedural requirements. Out-of-tolerance M&TE was identified and applicable procedure steps were reperformed with appropriately calibrated test equipment. The M&TE calibration laboratory personnel were knowledgeable of program requirements. M&TE calibration history, calibration

requirements and frequency, and the basis for the program were well maintained (Section M8.1).

Engineering

- Engineering reviews associated with resolution of north service water header leaks were found to be appropriate. Operability determinations were consistent with regulatory guidance and calculations were conservative (Section E2.1).

Plant Support

- The licensee's emergency program was being maintained in a state of full operational readiness. Changes to the program since the last inspection were consistent with commitments and NRC requirements, and did not decrease the licensee's overall state of preparedness. (Section P1.1)

- Radiation control and security practices were properly conducted. Inspected areas in the Radiation Control Area (RCA) were noted to be appropriately posted and secured as necessary. The security plan was effectively implemented and compensatory actions were initiated when required. (Section R1.1, S1.1)

Report Details

Summary of Plant Status

Robinson Unit 2 operated at 100 percent power through October 17. On October 17, the unit experienced a turbine runback and a subsequent automatic reactor trip from approximately 56 percent power. The unit was returned to power on October 18 and operated at full power through the end of the 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 turnovers, management reviews, 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). 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 Reactor Trip and Startup

a. Inspection Scope (71707)

Robinson Unit 2 experienced an automatic reactor trip from 56 percent power at approximately 2:10 a.m. on October 17. The inspectors responded to the site to assess plant response and licensee actions related to the trip.

b. Inspection Findings

Prior to the reactor trip, the unit was at 100 percent power and in a steady state condition. Operations Surveillance Test (OST)-005, "Nuclear Instrumentation (N) Power Range," Revision 18 was in progress on power range instrument N-44. As per the OST, the control rods were placed in manual and the rod drop mode was placed in bypass to preclude an inadvertent turbine runback. At 2:07 a.m., a failure of the 25 volt power supply in N-44 caused a control power fuse to blow. This caused the rod drop bistable to de-energize, resulting in a turbine runback.

During the turbine runback, the condenser steam dumps (selected to the T-average (Tavg) control mode) did not open as designed, causing Tavg to be greater than T-reference (Tref). This caused the primary system pressure to increase and pressurizer

Power Operated Relief Valves (PORVs) to lift and reseat several times. Upon noting the blown fuse in the N-44 drawer, the operator performing the OST reported the condition to the control room Reactor Operator (RO). The RO noted the ongoing run back and the difference between Tavg and Tref. He also observed that the condenser steam dumps had not opened as anticipated. The RO responded to the run back by placing the control rods in automatic and by selecting the steam pressure mode of operation for the condenser steam dumps. Control rods started stepping in and the condenser steam dump valves opened.

When the steam dump valves opened, the "C" steam generator level "swelled" to 75 percent causing a turbine trip. The turbine trip caused the reactor trip from 56 percent reactor power. The high steam generator level also caused both running main feedwater pumps to trip, which resulted in the two motor driven Auxiliary Feedwater (AFW) pumps and the steam driven AFW pump to start.

All control rods inserted following the reactor trip and electrical power transferred to the startup source. However, Source Range (SR) instrument N-32 read abnormally low (approximately 20 counts per second (cps) instead of 3000 cps). Operator response during the transient was appropriate.

A post trip review performed by the licensee concluded that the failure of the condenser steam dump valves to open on demand concurrent with a slower than designed response of the "C" feedwater regulation valve caused the unit to trip following the turbine runback. The condenser steam dump valves failed to open due to a misadjusted bias potentiometer. The response of the "C" feedwater regulation valve was slower than expected due to the valves' positioner being out of calibration.

In addition to the post trip review, the licensee also initiated an event review team to review the failures associated with the trip. The charter of the event review team was to perform a detailed evaluation of all the contributing factors, determine root cause(s), and recommend long term corrective actions.

Licensee corrective action included replacement of the power supply on N-44, replacement of the condenser steam dump comparator and control module, and recalibration of the "C" feedwater regulating valve positioner. N-32 was also repaired by replacing the power supply and signal connectors. Following completion of the corrective actions, the Plant Nuclear Safety Committee (PNSC) reviewed and approved the post trip review and authorized unit startup. The unit was taken critical at 12:49 p.m. on October 18 and placed on line at 5:30 p.m.

Following the reactor trip, the senior resident inspector was notified and a 10 CFR 50.72 notification was made. The inspectors responded to the site and conducted an event followup. This included verification of current plant status, understanding of the contributing causes, review of key plant component performance during the transient, discussion with the operators involved, and observation of post trip review meetings.

The inspectors also verified that the PORVs had reseated following the transient. Additionally, the inspectors reviewed the post trip review report and observed the recreation of the scenario on the plant simulator. The inspectors also reviewed the root cause analysis performed by the licensee which included recommendations for additional long term corrective actions.

c. Conclusions

Operator response to a turbine trip and automatic reactor trip was appropriate and in accordance with procedures. The licensee's event review was thorough and the findings identified the root causes of the event. The licensee appropriately generated CRs to track planned corrective actions.

O4 Operator Knowledge and Performance

O4.1 Operator Performance Issues

a. Inspection Scope (71707)

The inspectors reviewed two errors attributed to operator performance.

b. Observations and Findings

On September 15, hourly Emergency Response Facility Information System (ERFIS) plant parameter printouts for 4:00 a.m., 5:00 a.m., 6:00 a.m., and 7:00 a.m., were not printed in the control room due to a printer interface module problem. Operators are expected to review the printout hourly to verify plant parameters.

This printer failure was not recognized by the operating shift which caused the hourly reviews to be missed. At approximately 7:30 a.m., the oncoming shift identified the problem by noting the printout missing while attempting to verify a TS required parameter.

A significant CR was initiated as a result of the incident. Results of the CR were reviewed by the inspectors. The inspectors concluded that no TS required surveillances were missed as a result of the printer module failure. However, as corrective action, the licensee issued a night order to require the Control Room Shift Supervisor (CRSS) to review and initial the hourly ERFIS logs.

On October 5 the inspectors observed the performance of OST-401-1 "EDG 'A' Slow Speed Start," Revision 9. The inspectors observed a weakness in the proper documentation of prerequisite steps and coordination with the control room. When this was questioned by the inspectors it was corrected immediately.

During procedure step 7.2.25 the inspectors noted that the operator adjusted the speed control switch to increase load and adjusted the Generator AC Amp meter to 1000 to 1250 amperes instead of the required 1000 to 1250 kilowatts (kw) on the Generator KW meter. The operator requested a peer check by a second operator. The peer check identified that the wrong meter was being used and the load was then increased to the

required amount prior to proceeding to the next step. The use of a peer check by the operator prevented a mistake from occurring and was considered a good practice to help avoid errors. The inspectors reviewed all the data and the diesel test was completed satisfactorily with the acceptance criteria being met.

c. Conclusions

Two examples of operator inattention to detail were identified involving operators not recognizing a failed Emergency Response Facility Information System printer and the subsequent missed review of plant parameters; and documentation completeness during an emergency diesel generator surveillance.

O5 Operator Training and Qualification

O5.1 Licensed Operator Continued Training (LOCT)

a. Inspection Scope (71707)

The inspectors observed LOCT performed on the simulator and the related critiques and examinations.

b. Observations and Findings

On October 8, the inspectors observed LOCT on the simulator and portions of two LOCT simulator examinations. The training was organized, well conducted, and the critiques provided useful feedback to the LOCT students. The inspectors noted that the "As-Found" evaluations and "Time Out for Training Moment" as good practices. These training practices allowed the instructors and/or students to freeze the training scenario to discuss the occurring events. The inspectors also noted that the simulator examination scenario guides were well developed and detailed. The dynamic simulator facility operated properly with no deficiencies noted.

c. Conclusions

Simulator examinations observed demonstrated the effectiveness of licensed operator training. Simulator examination scenario guides were well developed and detailed. Operator performance was effectively assessed by the training department.

O7 Quality Assurance In Operations

O7.1 Plant Nuclear Safety Committee and Nuclear Assessment Section Oversight

a. Inspection Scope (71707)

PNSC and Nuclear Assessment Section (NAS) activities were reviewed to determine whether the onsite review functions were conducted in accordance with TS requirements.

b. Observations and Findings

The inspectors periodically attended 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 NAS audits and concluded that they were appropriately focused to identify and enhance safety.

c. Conclusions

The onsite review functions of the PNSC were conducted in accordance with TSs. During the PNSC meetings topics were thoroughly discussed and evaluated.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Service Water Valve Maintenance

a. Inspection Scope (61726)

The inspectors observed an inspection and thermal overload test on the Service Water (SW) pump discharge cross-connect valve, V6-12B.

b. Observations and Findings

The inspectors verified the test equipment used was within their calibration intervals. All work was performed with the work packages and procedures (PM-124 "Testing of Thermal Overload Relays for MCC-5," Revision 4, and PM-409 "Bridging and Insulation Resistance Testing of Electrical Equipment," Revision 6, present and in active use. The work was completed satisfactorily and the acceptance criteria were met.

c. Conclusions

Maintenance observed on a SW pump discharge cross-connect valve breaker was completed satisfactorily and the acceptance criteria were met.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Surveillance Observation (61726)

The inspectors observed all or portions of the following surveillance tests:

- OST-751, "Control Room HVAC R-1 Initiation and ERFIS Point Test," Revision 5

- OST-750-1, "Control Room Emergency Ventilation System- Train "A" (monthly)," Revision 8
- OST-51, "RCS Leakage Evaluation," Revision 24
- OST-908, "Component Cooling System Component Test," Revision 40
- OST-402-1, "EDG "A" Diesel Fuel Oil System Flow Test," Revision 9

The inspectors did not identify any deficiencies during the surveillance observations.

M2.2 Valve Position Verification

a. Inspection Scope (61726)

The inspectors reviewed two periodic valve position verification TS surveillance requirements that were not accomplished adequately. This issue was identified by the licensee, and the valves of concern were on the Component Cooling Water (CCW) system, and the Chemical and Volume Control System (CVCS).

b. Observation and Findings

On October 5, the licensee identified that the scope of surveillance procedure OST-942, "CCW To Safety Related Equipment Valve Position Verification," Revision 0 was limited to only those valves on the CCW main loop (i.e. CCW pump and heat exchanger suction and discharge valves and spent fuel pit heat exchanger suction and discharge valves). TS SR 3.7.6.1 requires, that every 31 days, correct position verification of each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position. Procedure OST-942 implemented the requirements of SR 3.7.6.1. The procedure did not include CCW valves supplying cooling water to the local bearing oil heat exchangers for the Safety Injection, Containment Spray, Residual Heat Removal, and Charging pumps.

The licensee entered TS SR 3.0.3 and verified the position of the CCW valves that had not been included in OST-942. The licensee identified 35 CCW valves that were verified in their correct position within the required time frame. A followup investigation conducted by the licensee concluded that the procedure writer as well as the reviewer misinterpreted the surveillance requirement, limiting the scope of OST-942 during initial procedure development prior to implementation of the Improved TS (ITS). As corrective action, OST-942 was revised to include all CCW valves affecting safety related systems that are not locked, sealed or otherwise secured.

The licensee also identified that OST-944, "Manual Isolation Verification for Valves Outside Containment," Revision 0, had not included all required valves. TS SR 3.6.3.2 requires, every 31 days, correct position verification of each containment isolation manual valve and blind flange that is located outside containment and locked, sealed or otherwise secured, and is closed during accident conditions. Procedure OST-944 implemented this requirement. The licensee identified that the OST had not included two normally closed containment isolation valves, CVC-295 (seal injection filter bypass) and CVC-309A (HCV-121 bypass) in the required verification.

A licensee investigation of the circumstances revealed that OST-944 was developed based on existing Operating Procedure OP-923, and had not included the two containment isolation valves. The licensee could not identify any definitive cause for OP-923 to not include the two containment isolation valves. The licensee immediately verified that the two valves were in the required position. The licensee plans to revise OST-944 and OP-923. CRs 98-02196 and CRs 98-02197 were initiated as a result of this problem.

The inspectors reviewed the CRs, attended the PNSC review related to the problem, verified the revision to OST-942, verified that OST-944 was on hold, and independently verified the position of a sample of the valves. The inspectors determined that licensee investigation of the event was thorough and corrective actions appropriate. This non-repetitive, licensee-identified and corrected violation is being treated as a non-cited violation, consistent with section VII.B.1 of the NRC Enforcement Policy. This NCV will be documented as NCV 50-261/98-09-01, Missed Technical Specification Required Surveillance for Valve Position Verification.

c. Conclusions

An NCV was identified involving two inadequate TS required surveillance procedures involving valve position verification. The affected valves were verified to be in their required position. Licensee corrective actions were appropriate.

M8 Miscellaneous Maintenance Issues

M8.1 Measuring and Test Equipment Program

d. Inspection Scope (62707)

The inspectors reviewed the licensee's program for control of Measuring and Test Equipment (M&TE).

e. Observations and Findings

The licensee's calibration program was described in Updated Final Safety Analysis Report (UFSAR) sections 1.8 and 17.3. The licensee procedure for implementing the M&TE program was PLP-053, " Measuring and Test Equipment (M&TE) Calibration Program," Revision 7. The inspectors reviewed several M&TE components and verified that the equipment was calibrated to standards and that there was a basis for the calibration standard accuracy and the calibration frequency. M&TE records were reviewed and the inspectors determined that the M&TE database was used to control required accuracy, usage history, usage limitations, performance trending, manufacturer recommendations, and calibration frequency. Audited equipment was found to be properly labeled and controlled.

The inspectors reviewed four equipment Out-of-Tolerance (OOT) condition evaluations and determined that the dispositioning of the nonconformances was conservative and documented in accordance with PLP-053 requirements. M&TE OOT conditions were

documented on a Test Result Evaluation Form (TREF) which notified previous users of the OOT condition and the evaluation of the OOT condition was documented on a Calibration Nonconformance Action Form (CNAF). The evaluation of the OOT condition was performed by the organization which used the M&TE and reviewed by M&TE personnel.

The inspectors questioned whether the level of review was changed if the OOT condition involved an operability determination. Operability determinations are performed by personnel trained for these evaluations. The inspectors reviewed the handling of a recent M&TE OOT condition and determined that a CR was generated in addition to the CNAF. The CR process included an operability evaluation by trained personnel. The licensee initiated CR 98-02164 to determine if the calibration nonconformance process should be incorporated into the site Corrective Action program.

f. Conclusions

An M&TE program review concluded that the program met the requirements of the UFSAR and was implemented in accordance with procedural requirements. Out-of-tolerance M&TE was identified and applicable procedure steps were reperformed with appropriately calibrated test equipment. The M&TE calibration laboratory personnel were knowledgeable of program requirements. M&TE calibration history, calibration requirements and frequency, and the basis for the program were well maintained.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Service Water (SW) North Header Leaks

a. Inspection Scope (37551)

The inspectors monitored and assessed engineering activities associated with leaks in the north SW header piping.

b. Observations and Findings

On September 16, the north SW header piping was confirmed to be leaking in a section of the piping between the auxiliary building and radwaste building. The leak initially was manifested as a standing puddle of water between the buildings. Operations personnel monitored available installed flow and pressure indicators for the SW system and detected no discernable abnormalities. The leaks were a result of two 1/4 inch diameter through wall defects in the piping. A number of additional wall pitting defects were discovered. The failure mechanism was attributed to localized general and galvanic corrosion where the pipe coating had been damaged.

The licensee initiated an Operability Determination (OD) for the north SW header via an Engineering Service Request (ESR)-9800469. The engineering analysis concluded that no operability concerns existed and the header was determined operable on September 18. Concurrently, the licensee initiated code repairs to stop the leaks.

The inspectors reviewed the completed ESR package that included a 10 CFR 50.59 screening checklist. The inspectors also reviewed guidance in Generic Letter (GL) 91-18 and GL 90-05 "Guidance For Performing Temporary Non-code Repair of ASME Code Class 1, 2, and 3 Piping," and compared the guidance to the methodologies used in the OD. Calculations for through wall and wall thinning flaw evaluations were checked in the OD. A preliminary report analyzing the corrosion mechanism at work on the piping was also reviewed. The inspectors reviewed instructions for code repairs to the north SW header (ESR #9800475) and discussed the OD results with the responsible engineer and the responsible engineering manager. The inspectors also reviewed long term corrective action plans for the SW piping.

The inspectors determined that the OD was consistent with GL 90-05 guidance. Values used for pipe stresses at the flaw were conservative. The inspectors found the conclusions of the OD to follow GL 91-18 guidance. The north header piping replacement was identified as the number one priority in the licensee's top ten equipment issues list.

c. Conclusions

Engineering reviews associated with resolution of north service water header leaks were found to be appropriate. Operability determinations were consistent with regulatory guidance and calculations were conservative.

E8 Miscellaneous Engineering Issues (37551, 92903)

E8.1 (Closed) Inspector Followup Item 50-261/97-201-01: Operating Event Review of IN. The NRC design team questioned the licensee's followup action for NRC Information Notice 91-38, Thermal Stratification in Feedwater Piping Systems. The licensee was unable to provide the results of their review of Information Notice (IN) 91-38 to the NRC design team. CR 97-02238 was initiated by the licensee to document and disposition this issue. Subsequent to the inspection, the licensee retained a contractor, Altran, to perform a review of the IN. The results of the review were documented in Altran Technical Report 98135-TR-01, Feedwater Line Stratification Evaluation at the H. B. Robinson Nuclear Station. The licensee initiated ESR 9800403, FW Thermal Stratification, Pipe Support Modification, to followup on the recommendations in the Altran report. These recommendations included verification of settings of spring can supports and verification of the as-constructed details for support number FW-6C-83. The inspectors reviewed the ESR, the Altran report, and examined pipe support number FW-6C-83 and concluded that the licensee had performed an adequate review of the IN. In addition, the licensee has performed a review of their program for followup on INs as part of the corrective actions to resolve CR 97-02238 and identified some other examples of failure to properly document review and closure of INs. No unresolved safety issues were identified as a result of this review.

E8.2 (Closed) Inspector Followup Item 50-261/97-201-13: RAB Flooding Due to SW System Passive Failure: The design team questioned the licensee regarding how a failure of the service water system piping in the Reactor Auxiliary Building (RAB) could be controlled to prevent flooding which could then potentially disable safety-related equipment in the RAB. The inspectors discussed this issue with licensee engineers. The discussions disclosed that if flooding occurred in the RAB, water would be distributed throughout the RAB and routed to the RAB floor drains. When floor drain capacity is exceeded, water would overflow door dikes and flow outside of the building through the doorway openings. Safety related equipment is elevated above the door dikes. The inspectors walked down the RAB and verified that safety-related equipment which could be affected by flooding was elevated above the door dikes. The inspectors also reviewed Abnormal Operating Procedure (AOP)-022, "Loss of Service Water," Revision 19, which addressed operator actions in the event of flooding in the RAB. This procedure directs the operators to open interior and exterior RAB doors to prevent flooding of safety-related equipment in the RAB. The licensee revised Generic Issues Document R87038/006, "Pipe Failures," to clarify the consequences of internal plant flooding. The inspectors reviewed Revision 2, dated June 16, 1998, of this document and verified that the discussion on internal plant flooding had been clarified to address this issue.

E8.3 (Closed) Inspector Followup Item 50-261/97-201-15: SI Pump Motor Load Evaluation. Inconsistencies between calculation numbers RNP-E-5.004 and RNP-E-8.016 resulted in the design team questioning whether the Safety Injection (SI) pumps could operate in excess of their rated 350 horsepower. The licensee initiated CR 97-01194 to document and disposition this issue. ESR 9700276 was completed to evaluate the horsepower ratings for the SI pumps. Corrective actions included performance of an operability evaluation which determined that affected components were operable. The calculations were also reviewed to address the inconsistencies. Some discrepancies were identified in calculation numbers RNP-E-5.004 and RNP-E-8.002. The inspectors reviewed the calculations and verified the calculation discrepancies had been corrected. These included correction of cable ampacities in calculation RNP-E-5.004, Revision 4, which were documented on Design Change Backup (DCBU) form RNP-E-5.004-0002, and corrections to electrical loads based on maximum brake horsepower peaks at accident conditions in calculation RNP-E-8.002, which were documented on DCBU form RNP-E-8.002-0014. The changes to the calculations did not affect the conclusions/output of the calculations.

E8.4 (Closed) Inspector Followup Item 50-261/97-201-19: Containment Water Level Setpoints and Instruments Used in EOPs and AOPs. The design team questioned licensee engineers regarding instrument uncertainty and accuracy for instruments used in Abnormal Operating Procedures (AOPs) and Emergency Operating Procedures (EOPs) which were not classified as Regulatory Guide (RG) 1.97 Category 1 instruments. The licensee initiated CR 97-01221 to document and disposition this issue. After further review, the licensee determined that they would continue to follow the Westinghouse Owners' Group (WOG) guidelines concerning accuracy of non Category 1 RG 1.97 instruments for use in AOPs/EOPs. The licensee determined that since they complied with the WOG guidelines and that they had made no additional regulatory

commitments regarding use of these instruments, no corrective actions were required to resolve this issue. The inspectors concurred with the licensee's conclusions.

E8.5 (Closed) Inspector Followup Item 50-261/97-201-20: CCW System Overpressurization. The design team questioned the implementation of recommendations contained in a July 26, 1984 Westinghouse letter, Subject: Component Cooling Water System Overpressurization Modification. The licensee initiated CR 97-01753 to address this issue. The licensee also completed ESR 9700014, CV Isolation Reclassification of the Unit 2 CCW system. The inspectors reviewed the ESR and concluded that the licensee's methods to address the concerns in the Westinghouse letter were adequate.

E8.6 (Closed) Inspector Followup Item 50-261/97-201-22: Ampacity Derating of Cables. During review of calculation number RNP-E-5.004, Ampacity Evaluation of Safety Related Power Cables on 480V and 208V AC Motor Control Centers (MCCs) and Buses, Revision 4, the design team questioned the basis for not derating cables routed in fire stops, seals, and/or covered with fire wrap. The licensee initiated CR numbers 97-01085, 97-01155, and 97-01219 to document and disposition this issue. Corrective actions included revision of calculation numbers RNP-E-5.001, RNP-E-5.004, RNP-E-5.018, RNP-E-5.019, and RNP-E-5.038 to address derating of cables due to fire stops. The inspectors reviewed the calculations and verified that they had been revised to clarify cable derating. The inspectors noted that the conclusions/results of the calculations were not changed or affected by these revisions. The inspectors also reviewed CP&L procedure EGR-NGGC-0103, Revision 2 which incorporated additional guidance regarding cable derating due to fire stops.

E8.7 (Closed) Inspector Followup Item 50-261/97-201-23: Agastat Relay Lifetime. During review of various test reports, the design team identified a discrepancy between test reports for the service life for Agastat E7000 series relays. The manufacturer's recommended replacement schedule for these relays is ten years. The licensee stated that the relays have a qualified life of greater than 50 years based on testing documented in Acton Laboratories Test Report 15761. The inspectors reviewed the test report and noted that environmental qualification testing did indicate that the relays had a service life of 51.9 years. However, in a June 10, 1994, letter, the manufacturer informed the licensee that these relays were only qualified for 10 years from the date of manufacture, or 25000 operations, whichever occurs first. The inspectors questioned licensee engineers regarding the use of the Agastat relays in safety related systems and their scheduled replacement dates. These discussions disclosed that the licensee did not have a preventive maintenance program to identify when the relays required replacement. However review of work requests and purchase documentation showed that none of the relays currently installed in safety related system exceeded the ten year services life. Specific record reviewed included those for relays installed in the auxiliary feedwater system. These relays had

been installed in 1989. The inspectors identified the lack of a preventive maintenance program which would identify Agastat series E7000 relays which exceeded their service life as an unresolved item, pending further review by NRC. This issue is identified as URI 50-261/98-09-02, Agastat E7000 Series Relay Replacement Schedule.

E8.8 (Closed) Inspector Followup Item 50-261-97-201-26: Station Battery Test Control Deficiencies and Test Procedure Revisions. The design team questioned test acceptance criteria specified in two procedures for testing of station batteries. The questions concerned the minimum terminal voltage battery capacity specified in CP&L procedure MST-920, "Station Battery Performance Capacity Test (Five Year Interval)," and failure to include test acceptance criteria in CP&L procedure MST-921, "Station Battery Service Test." The licensee initiated CR numbers 97-01046 to address issues concerning procedure MST-920, and CR 97-01138 for MST-921 issues. The inspectors reviewed the licensee's corrective actions. These included revising procedure MST-920 to require that terminal capacity is based on terminal voltage of 105 V DC or greater. The inspectors reviewed the current revision, Revision 16, of MST-920, and verified that the minimum terminal voltage of 105 V DC was specified. The acceptance criteria for procedure MST-921 had been previously contained in various calculations. The licensee revised procedure MST-921 to include the acceptance criteria within the procedure. The inspectors reviewed the current revision, Revision 14, of MST-921 and verified that the procedures included the acceptance criteria for discharge current values. No discrepancies had been identified in actual performance of the battery surveillance testing prior to revision of MST-920 and MST-921.

E8.9 (Closed) VIO 50-261/98-01-03: Failure to Update UFSAR. The licensee responded to this violation in a letter dated April 10, 1998, Subject: NRC Inspection Report 50-261/98-01, Notice of Violation. The Notice of Violation documented three examples of failure to revise the Updated Final Safety Analysis Report (UFSAR). The licensee's corrective actions for this violation included revision of the UFSAR to correct the specific violation examples, and performance of additional reviews of the UFSAR to identify other errors. The licensee submitted the corrections to the UFSAR to NRC in Amendment 15 to the UFSAR on October 14, 1998. The inspectors reviewed the following sections in Amendment 15 and verified that the licensee had corrected the UFSAR to address the violation examples: Section 6.3.2.2.3, Table 6.3.2-5, Figures 6.3.2-4a, -4b, and -4c, Section 10.4.8.2, and Table 10.4.8-1. The licensee also provided training for site personnel on the design and licensing basis of the plant to emphasize the importance of maintenance of the design and licensing basis and the maintenance of applicable documents, including the UFSAR.

E8.10 (Closed) EEI 50-261/98-03-01: Inadequate Design Control. This EEI identified twenty potential examples of inadequate design control activities and failure to implement the requirements of 10 CFR 50, Appendix B, Criterion III. On March 4, 1998, a Notice of Violation was issued to the licensee which cited the 20 examples as Violation Items 50-261/98-03 EA 98-043 and EA 98-050, Item B (14 examples), Item C (4 examples), and Item D (2 examples).

The licensee responded to the Notice of Violation in a letter dated April 3, 1998, Subject: Notice of Violation, NRC Inspection Report No. 50-261/98-03, EA 98-043, and 98-050. In their response, the licensee stated that Examples 4, 6, 12, 13, and 14 of Violation B and Example 1 of Violation D were not violations of NRC requirements. Resolution of these violation examples was addressed under Unresolved Item number 50-261/98-05-05, discussed in paragraph E8.14, below. The inspectors reviewed the licensee's corrective actions for the violation examples documented in their April 3, 1998, response. The results of the inspectors' review are summarized below:

Violation B

Examples 1, 2, 3, 5, and 7 through 11 identified errors in various calculations. The inspectors reviewed calculation numbers RNP-I/INST-1023, RNP-I/INST-1109, RNP-I/INST-1058, RNP-M/MUCH-1620, RNP-E-6.020, RNP-E-6.021, RNP-E-6.23, RNP-E-6.004, RNP-E-6.018, and RNP-E-8.016 and verified the calculations had been revised to correct the errors identified in the violation examples. The licensee performed a review of calculation RNP-E-5.018 to determine the effect on cable Ampacity evaluations of the 60° Centigrade (C) cable which was found to be installed in the plant. The licensee reviewed a random sample of 71 other cables and determined that no others had a temperature rating less than the 75° C value assumed in calculation RNP-E-5.018. The licensee replaced the 60° C rated cable with a new cable during the Spring, 1998 refueling outage under a modification installed by ESR 9800024. The inspectors reviewed the completed ESR package and verified the modification had been implemented. Additional corrective actions to correct the violation and avoid further violations in this area was training of engineering personnel in the design and licensing basis, design review training, and review technology training. The purpose of this training is to reduce errors in performance of design verification. The inspectors discussed the training with engineering personnel, and reviewed randomly selected training records.

Violation C

This violation identified four examples of failure to translate design input/output into drawings or surveillance procedures. The inspectors reviewed the corrective actions to resolve the discrepancies. These included revision of in-service testing procedures to require quarterly testing of valve numbers SW-906, SW-907, CC-927, and CC-928. The inspectors reviewed the results of surveillance testing (opening and closing the manual valves) on SW-906 in October, 1997, and January, May, and July, 1998; and on SW-907 in December, 1997, and March, May, and August, 1998. The inspectors also reviewed revisions to station battery testing procedures, MST-920 and MST-921 and verified the issues identified during by the NRC design team had been corrected.

Violation D

Example 2 of Violation D identified three calculations which had not been voided or superseded by other calculations. The licensee voided the three calculations in 1997, after completion of the design inspection. The inspectors reviewed the calculations and verified that they had been voided. Additional corrective actions included performance of a self assessment of the service water system by September 30, 1998. The inspectors reviewed a draft report of the self-assessment results. No additional calculations were identified as superseded or voided.

E8.11 (Closed) EEI 50-261/98-03-04: SI Pumps Inoperable Due To Inadequate NPSH. This EEI identified potential examples of inadequate Net Positive Suction Head (NPSH) for the B and C Safety Injection (SI) pumps resulting from an incorrect design input in a design modification. On March 4, 1998, a Notice of Violation was issued to the licensee as Violation 50-261/98-03, EA 98-043, and EA 98-050, Item A, which cited the inadequate NPSH for SI pumps B and C as a violation of Technical Specification 3.3.1.1.c. The licensee responded to this violation in a letter dated April 3, 1998, Subject: Notice of Violation, NRC Inspection Report No. 50-261/98-03, EA 98-043, and EA 98-050. The cause of the violation was attributed to failure of engineering personnel to adequately verify design inputs. Immediate corrective actions were development of a system flow model which identified the NPSH available and required for each SI pump, and implementation of a modification under ESR 9700336 which raised the water level in the Refueling Water Storage Tank (RWST). The inspectors reviewed the records documenting implementation of the ESR and also changes to RWST instrumentation Setpoints implemented by ESR 9700307. Additional margin in NPSH was provided for the B and C SI pumps in a modification implemented under ESR 9700366. The inspectors examined the ESR package and the suction piping for the B and C SI pumps and verified that the piping was modified in accordance with the design requirements. The inspectors examined the quality records associated with the piping modifications implemented under WR/JO 97-AEXC1. Records examined included: the bill of materials, weld data reports for 22 welds, results of visual inspections and nondestructive liquid penetrans tests performed on the welds, records of weld repairs, and results of hydrostatic testing performed on the modified piping.

The licensee also committed to perform an A/E type inspection on the Component Cooling Water (CCW) system during 1998 as additional corrective actions for this violation. This inspection was performed by a five man team from Sargent and Lundy (S&L) between September 20 and October 23, 1998. The inspectors reviewed the preliminary findings of the S&L team. They concluded that overall the CCW system was capable of performing the design-basis safety functions and adhered to its design and licensing bases. Several potential weaknesses were identified which were incorporated into the licensee's corrective action program for followup and resolution.

8.12 (Closed) Violation 50-261/98-03-05: Failure To Comply with 10CFR50.59 For Modifications To Two Valves Under ESR 9600012. The violation was issued for failure to perform a safety evaluation which provided the basis for a determination that a stroke time increase for Residual Heat Removal (RHR) valves RHR-744 A and B did not involve an unreviewed safety question. Replacement of motor pinions and worm shaft gears resulted in an increase in the motor operator stroke time beyond the ten seconds specified in the UFSAR. Licensee corrective action included counseling the individuals involved with the modification, performing a new 10CFR50.59 safety evaluation and revising the UFSAR to reflect the new stroke time. Corrective action to prevent recurrence included a memorandum to Qualified Safety Reviewers (QSRs) regarding use of UFSAR electronic searches, and training for Robinson engineering and QSR personnel. The corrective actions were further detailed in CR 9800251. The inspectors reviewed the CR and verified the corrective actions.

E8.13 (Closed) Violation 50-261/98-03-06: Failure To Provide Accurate Information To NRC On Cable Separation For SI Pumps. The violation was issued for inadequate cable separation for the "C" SI pump control cables. The immediate corrective action was to declare "C" SI pump inoperable and to place the "B" SI pump in service. The licensee completed modification ESR 9700274 which rerouted the control cables for "C" SI pump to meet the separation requirements, four feet vertically and eighteen inches horizontally. Corrective action to prevent recurrence included revision of procedure PLP-057, "Self Assessment," to include a requirement for engineering to perform an annual system assessment which would focus on translation of design basis information including cable separation. Corrective action was further detailed in CR 9701177. The inspectors reviewed the modification package, performed walkdowns and verified that the installation met modification package requirements for cable separation.

E8.14 (Closed) Unresolved Item 50-261/98-05-05: Questions on Design Calculations The licensee's April 3, 1998 response to the March 4, 1998, Notice of Violation (NOV) for EEI 50-261/98-03-01 stated that examples 4, 6, 12, 13, and 14 of Violation B and example 1 of Violation D were not violations of NRC requirements. The inspectors reviewed the calculations concerning these violation examples and determined that examples 12, 13, and 14 of Violation B were not violation examples. However, the inspectors determined that examples 4 and 6 of Violation B and example 1 of Violation D were violations of NRC requirements. These three issues will be tracked as examples of the violations as stated in the NOV referenced above. The reasons for the inspectors' conclusions are discussed below:

Violation B

Example 4

This violation example concerned failure to include seismic uncertainty factors in calculation numbers RNP-I/INST-1040 and RNP-I/INST-1043 for main steam line flow and pressure instrumentation. The licensee's response stated that seismic uncertainty factors were not required to be included in these calculations. However since these instruments are required to be operable before and after a

seismic event, the seismic uncertainty factors were required to be included in the calculations. Discussions with licensee engineers disclosed that the calculations had been revised to included seismic uncertainty factors after the NRC design team identified this issue in April - May, 1997. The inspectors reviewed Revision 1 of calculation RNP-I/INST-1040, dated June 2, 1997, and Revision 2 of calculation RNP-I/INST-1043 and verified that the seismic uncertainty factors were included in the calculations. The inspectors determined that failure to include the seismic uncertainty factors had been a violation of 10 CFR 50, Appendix B, Criterion III, as stated in the NOV. Since the licensee has taken adequate corrective actions to correct the violation example, no additional response is necessary.

Example 6

This violation example concerned failure to include rupture of non-seismic piping that supply instrument and station air compressors in Revision 1 of calculation RNP-M/MUCH-1362, Service Water Wash Piping Flow Analysis. In their response, the licensee stated that this piping had been included in an Engineering Evaluation, (EE) 89-108 which was referenced in the calculation. The inspectors reviewed EE 89-108, Analysis for SW System Function Post-Seismic, and determined that this EE only addressed the non-seismic service water piping to the diesel air dryers. However, in response to the NRC design team's questions, the licensee initiated CR 97-00993. The corrective actions to resolve the issues included a review of Calculation RNP-M/MUCH-1362 which resulted in the determination that failure of the piping to the diesel air dryers had been bounded in the calculation. Additional corrective actions included revision of the SW Design Basis Document and UFSAR to clarify the design criteria for the SW piping to the diesel air dryers. The inspectors determined that failure to consider rupture of the non-seismic service water piping to diesel air dryers in calculation RNP-M/MUCH-1362 was a violation of 10 CFR 50, Appendix B, Criterion III, as stated in the NOV. However, since the licensee's corrective actions have been completed, no additional response is necessary.

Example 12

This violation example concerned use of a temperature value of 100° Fahrenheit (F) in calculation number RNP-M/MUCH-1460 for determination of the Steam Driven Auxiliary Feedwater (SDAFW) pump NPSH instead of the value of 115° F listed in the plant parameter document for Cycle 18. The inspectors reviewed the licensee's response, discussed this issue with licensee engineers, and reviewed the calculation. The inspectors concurred with the licensee that this issue was not a safety concern or an example of failure to verify adequacy of design since the use of 100° F in the SDAFW pump NPSH design analysis was conservative since the maximum temperature for SDAFW pump operability is 89° F. The inspector reviewed CP&L Operating Procedure OP-501, "Condensate System," Revision 38.

The condensate storage tank water temperature is measured on a daily basis per steps 6.1.2.16.c and 6.1.2.17 of OP-501. If the Condensate Storage Tank (CST) temperature exceeds 89° F, the SDAFW pump is declared inoperable.

Example 13

This violation example concerned use of a value of 1.00 for specific gravity in calculation RNP-M/MUCH-1394. The maximum specific gravity for water is 1.0088 at 40° F within the CST operating limits. The inspectors concurred with the licensee's conclusions that use of 1.0088 versus 1.00 for the specific gravity value in the calculation would not change the output or conclusions of the calculation, and therefore was not a violation of NRC requirements.

Example 14

This violation example concerned calculation inconsistencies in the CST levels at which a change in AFW suction supply occurs. Calculation RNP-I/INST-1015 used a 10 percent level whereas calculation 84065-M-06-F used a 15 percent level. The inspectors reviewed the calculations and discussed this issue with licensee engineers. Review of the calculations showed that calculation 84065-M-06-F determined the water level in the CST when the plant operators have 20 minutes to switch the AFW suction source to an alternative supply. This level was calculated at 14.116 percent which corresponds to 31230 gallons. The actual alarm setpoint for this condition (low level alarm) is set higher, at 15 percent CST level. The CST level determined in calculation RNP-I/INST-1015 is the low, low level alarm setpoint, which is 6.873 percent CST level, at which AFW suction must be terminated from the CST. After including values for instrument errors and uncertainties, the CST level was determined to be 9.90 percent for the low- low level alarm setpoint. This value was set at 10 percent CST level. The inspectors concurred with the licensee that this example was not a violation of NRC requirements.

Violation D

Example 1

This violation example concerned a calculation in ESR 96-00474, Revision 0, which had not been design verified. The licensee, in their response, stated that since the ESR was an Engineering Disposition (ED) ESR, in accordance with CP&L procedure NGGC-EGR-0005, Engineering Service Requests, design verification was not required. The inspectors reviewed the ESR and the attached calculation and concluded that design verification of the calculation was required to comply with the requirements of 10 CFR 50, Appendix B, Criterion III. After further discussions, the licensee concurred with the inspectors and initiated CR 98-02230 to disposition this problem. A design verification was performed on the calculation. In addition the licensee reviewed more than 500 ED type ESRs which had been issued since January 1, 1996, and identified three additional ED ESRs which should have been design verified. The design verification for the four ESRs was completed during the inspections. The inspectors reviewed the

ESRs and verified that the design verification had been completed. No errors had been identified in the ESRs or calculations during the design verification process. An interim step has been added to the ESR process which requires supervisor approval of ED ESRs to verify the ED content complies with the procedure NGGC-EGR-0005. The inspectors determined that failure to perform design verification of the calculation in ESR 96-00474 was a violation of 10 CFR 50, Appendix B, Criterion III. However, since the licensee's corrective actions have been completed, no additional response is necessary.

IV. Plant Support

P1 Conduct of Emergency Preparedness (EP) Activities

P1.1 Program Review

a. Inspection Scope (82701)

The inspectors reviewed EP program activities at the H. B. Robinson Plant to determine whether the licensee's emergency response capability was maintained in a state of operational readiness, and to determine whether changes to the program since the last such inspection (in November 1996) met commitments, NRC requirements, and affected the licensee's overall state of preparedness.

b. Observations and Findings

The inspectors reviewed Emergency Plan Revisions 34, 35, 36, 37, 38, 39, and 40, issued between May 1996 and June 1998. These changes to the Plan were submitted in accordance with regulatory requirements and did not adversely affect the licensee's level of emergency preparedness. The inspectors noted that Revision 40 did not include markings to show where changes had been made, as specified in Section 5.6.2.1 of the Plan. The licensee initiated a CR 98-02142 to investigate the cause and track corrective actions. No emergency declarations were made since the last inspection.

Emergency facilities, equipment, instrumentation, and supplies were inspected and found, almost without exception, to be well maintained. An exception was the stock of silver zeolite cartridges, contained in several emergency kits. These cartridges are used in conjunction with an air sampler as a medium for detecting radioiodine in the atmosphere in the event of a radiological release. The cartridges were supplied by the vendor in packages of 10, sealed in heavy-duty 10-mil plastic sleeves to maximize shelf life (rated at 10 years). However, none of the cartridges carried any indication of manufacturing date or shelf-life expiration date, nor was the licensee able to definitively determine this information from its purchasing records. The inspectors questioned whether any of the licensee's silver zeolite cartridges were within their rated 10-year shelf life. The licensee initiated CR 98-02139 to ensure follow-up and corrective action as necessary. In addition, CR 98-02137 was initiated to investigate potential shelf-life issues with other categories of supplies. The only significant change to emergency facilities and equipment since the last inspection was the implementation of an

electronic information-feedback system for the public-notification sirens, which provided essentially continuous information on the status of each siren's operability.

A significant change to the organizational and management control of the EP program since the last inspection was the appointment of a new Emergency Preparedness Supervisor in September. This individual, interviewed at length by the inspectors, appeared committed to maintaining a high-quality EP program. Interviews with EP staff and review of program accomplishments and initiatives disclosed continuing strong management support for EP.

The inspectors reviewed the Emergency Response Organization (ERO) training program and exercise/drill schedule. Virtually all EP training drills were conducted in conjunction with LOCT, which meant that they were "driven" by the simulator and were performed much like the biennial, NRC-evaluated exercises. With the ERO divided into four teams, the licensee conducted 12 LOCT/ERO training drills in 1997 and seven in 1998 (including a planned November drill). All of these drills included actual notifications of State and county agencies, just as would occur in a real emergency. A detailed critique package was issued to all ERO personnel to convey "lessons learned" from each cycle of drills. The ERO drill schedule provided for annual participation by each of the five control room shifts and essentially all personnel assigned to the ERO, with many individuals participating in two or three drills each year. Review of a random sample of training records identified no deficiencies relative to training requirements.

The inspector reviewed Audit Reports R-EP-96-01 and R-EP-97-01, and concluded that the audits, conducted by the Nuclear Assessment Section, were comprehensive and met NRC requirements. Corrective actions taken in response to issues identified during audits were thorough and timely.

Licensee findings resulting from activities such as exercises, drills, self-assessments, and audits were tracked to ensure resolution in one of two systems: the plant-wide Corrective Action Program (CAP), which tracked CRs, or the EP Improvement Database. The more significant EP issues were included in the CAP, while minor items not captured by CRs were included in the EP Improvement Database. Statistics for the latter system showed that 55 items (12 of which originated in 1997) were currently open, and that 184 items had been closed since October 1, 1996. Management commitment and attention to timely corrective actions for identified problems in EP were evident from the nature of the measures taken to resolve problems.

c. Conclusion

The licensee's emergency program was being maintained in a state of full operational readiness. Changes to the program since the last inspection were consistent with commitments and NRC requirements (except as noted), and did not decrease the licensee's overall state of preparedness.

P8 Miscellaneous EP Issues (82701)

P8.1 (Closed) IFI 50-261/97-13-01: Exercise Weakness -- Untimely Declaration of a Notification of Unusual Event. The inspectors reviewed the licensee's February 12, 1998 response to this finding. To the extent possible, the inspectors independently verified the corrective actions delineated in this letter, as well as other improvements not listed in the letter. The licensee had aggressively pursued appropriate actions to prevent recurrence of the subject weakness. The inspectors reviewed the final package for CR 97-02306 on this issue, which was very detailed and thoroughly addressed the subject finding. Timely emergency declarations during all of the LOCT/ERO drills in 1998 to date were considered proof of performance. This item is closed.

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 inspector concluded that radiation control practices were being conducted in a accordance with procedures. The inspector 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.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

During the period, the inspector 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 inspector 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.

During a walkdown associated with a planned plant modification, the licensee identified several areas of potential security barrier vulnerabilities. As a result, the licensee promptly initiated compensatory actions while developing a permanent solution. The

permanent solution was implemented and the compensatory measures discontinued. The licensee notified the resident inspectors as well as discussed the circumstances with the NRC regional security inspector.

The resident inspectors reviewed the security plan and periodically verified compensatory measures. Additionally, the inspectors observed the installation of the permanent solution and concluded that the identification of this vulnerability by the licensee was an example of a good questioning attitude. The installation of a prompt permanent solution to resolve the vulnerability was a strength.

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 October 23, 1998. 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
R. Warden, Manager, Nuclear Assessment Section
T. Wilkerson, Manager, Regulatory Affairs
D. Young, Vice President, Robinson Nuclear Plant

NRC

B. Desai, Senior Resident Inspector
A. Hutto, Resident Inspector

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 82701	Operational Status of the Emergency Preparedness Program
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-261/98-09-01	NCV	Missed Technical Specification Required Surveillance For Valve Position Verification (Section M2.2)
50-261/98-09-02	URI	Agastat E7000 Series Relay Replacement Schedules

Closed

50-261/98-09-01	NCV	Missed Technical Specification Required Surveillance For Valve Position Verification (Section M2.2)
50-261/97-201-01	IFI	Operating Event Review of IN (Section E8.1)
50-261/97-201-13	IFI	RAB Flooding Due to SW System Passive Failure (Section E8.2)
50-261/97-201-15	IFI	SI Pump Motor Load Evaluation (Section E8.3)
50-261/97-201-19	IFI	Containment Water Level Setpoint and Instruments Used in EOPS and AOPs (Section E8.4).
50-261/97-201-20	IFI	CCW System Overpressurization (Section E8.5)
50-261/97-201-22	IFI	Ampacity Derating of Cables (Section E8.6)
50-261/97-201-23	IFI	Agastat Relay Lifetime (Section E8.7)
50-261/97-201-26	IFI	Station Battery Test Control Deficiencies and Test Procedure Revisions (Section E8.8)
50-261/98-01-03	VIO	Failure to Update UFSAR (Section E8.9)
50-261/98-03-01	EEI	Inadequate Design Control (Section E8.10)
50-261/98-03-04	EEI	SI Pumps Inoperable Due to Inadequate NPSH (Section E8.11)
50-261/98-03-05	VIO	Failure To Comply With 10CFR50.59 For Modifications to Two Valves Under ESR 9600012 (Section E8.12).

50-261/98-03-06	VIO	Failure to Provide Accurate Information to NRC On Cable Separation for SI Pumps (Section E8.13).
50-261/98-05-05	URI	Questions on Design Calculations (Section E8.14)
50-261/97-13-01	IFI	Exercise Weakness -- Untimely Declaration of a Notification of Unusual Event (Section P8.1)