U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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License No:	DPR-23
Report No:	50-261/98-05
Licensee:	Carolina Power & Light (CP&L)
Facility:	H. B. Robinson Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	March 29 - May 9, 1998
Inspectors:	B. Desai, Senior Resident Inspector A. Hutto, Resident Inspector F. Wright, Region II Inspector G. MacDonald, Region II Project Engineer
Approved by:	M. Ernstes, Acting Chief, Projects Branch 4 Division of Reactor Projects

Enclosure 2

EXECUTIVE SUMMARY

H. B. Robinson Power Plant, Unit 2
NRC Integrated Inspection Report 50-261/98-05

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 II based inspectors.

Operations

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

- A non-cited violation involving inadequate testing of refueling equipment was identified. There were no direct safety consequences of the fuel assembly drop incident. Had an irradiated assembly been subject to a similar drop, any release of radioactivity would have been contained, due to containment closure requirements during fuel movement. Licensee follow up to this event was appropriate (Section O1.2).

- The inspector observed portions of startup related activities and concluded that management decision to suspend startup to verify potential affects of the seismic event were conservative. The inspector did not note any affect to the plant as a result of the seismic event. Operator performance during startup was considered good (Section O1.3).

- Reactor operator response to the failed closed turbine control valve was incorrect. Licensee plans to review this event to enhance future training activities. Overall plant response was appropriate. Reactor startup activities following the trip were uneventful (Section O2.1).

- A violation for failure to perform a required surveillance test on the containment personnel air lock test was identified (Section O4.1).

- Nuclear Assessment Section and Plant Nuclear Safety Committee continued to provide strong oversight, including during refueling outage 18 (Section O7.1).

Maintenance

- Maintenance and surveillance activities were performed satisfactorily. The inspector noted good controls of housekeeping and good supervisor oversight of work activities (Section M1.1).

- The repairs on the service water (SW) header were appropriately completed. Management oversight of this problem was considered good (Section M2.1).

- The "A" motor driven auxiliary feedwater (MDAFW) pump was deadheaded during a test configuration. This did not cause damage to the pump. Licensee actions to determine the root cause, including formation of an event review team (ERT), were appropriate.

- Two orifices on the MDAFW recirculation lines that had been worked on during the outage were installed backwards due to inattention to detail on the part of the mechanic. This issue was identified as a non-cited violation. The incorrectly installed orifices did not contribute to the pump deadheading (Section M4.1).

Engineering

- Modification packages reviewed were acceptable. Modification 97-382 implemented corrective action to restore the damper control scheme for the Containment Recirculation Cooling System units (Section E1.1).

- The environmental qualification (EQ) components inside containment which were inspected were being maintained with qualified seals in accordance with EQ program requirements. The EQ data packages reviewed were being updated in accordance with procedure EGR-NGGC-156, Environmental Qualification Of Electrical Equipment Important To Safety. No backlog of unincorporated engineering service requests (ESRs) was noted (Section E2.1).

- The licensee was conservative in evaluating a Westinghouse BF relay failure during RFO-18 testing. The relay testing discrepancies were being dispositioned in accordance with corrective action program requirements (Section E2.2).

- In response to the licensee's letter of April 4, 1998, several examples of cauculational deficiencies identified in Notice of Violation dated March 4, 1998, have been withdrawn and will be tracked as an Unresolved Item (Section E8.1).

Plant Support

- Housekeeping and cleanliness within the radiation control area were acceptable (Section R1.1).

- Overall, the inspectors observed good radiological controls and radiation worker compliance throughout the inspection (Section R1.1).

- A Non-Cited Violation was identified for radiation worker's failure to comply with radiation protection procedures (Section R1.1).

- The effectiveness of the licensee's dose reduction efforts in non-outage periods during 1997 were very good and had resulted in the site's lowest annual collective dose. The 1997 collective dose was 13 person-rem (Section R1.2).

- Overall, as-low-as-reasonably-achievable (ALARA) planning efforts were appropriate and were effectively implemented for most outage work activities. Unanticipated problems and poor planning resulted in excess dose of 13 person-rem for three of the thirty-one planned projects (Section R1.2).

- Overall licensee contamination control measures were effective in containing radioactive byproduct contamination and minimizing radiation exposures to the contamination (Section R1.3).

- Personnel contaminations were down from previous outages. The licensee was evaluating the events to identify their causes and was taking corrective actions to reduce the number of personnel contaminations. The 14 Personnel Contamination Events (PCEs) documented in 1997 were the site's lowest (Section R1.3).

- Licensee use of self assessments in the radiation protection program area was good (Section R7.1).

Report Details

Summary of Plant Status

Robinson Unit 2 was in Refueling Outage (RFO) 18 at the beginning of the report period. Plant startup activities were initiated on April 10, 1998 and the unit entered Mode 1 on April 10. Full power was reached on April 18. On April 25, the unit experienced an automatic reactor trip from 100 percent power due to low Steam Generator (SG) level. The unit was started up on April 26 and returned to full power on April 27. The unit operated at 100 percent for the remaining portion of the report period.

I. Operations

01 Conduct of Operations

01.1 General Comments (71707)

The inspector conducted frequent control room tours to verify proper staffing, operator attentiveness and communications, and adherence to approved procedures. The inspectors attended daily operation turnovers, management reviews, and plan-of-the-day meetings to maintain awareness of overall plant operations. Operator logs were reviewed to verify operational safety and compliance with TSs. Instrumentation, computer indications, and safety system lineups were periodically reviewed from the Control Room to assess operability. Frequent plant tours were conducted to observe equipment status and housekeeping. Condition Reports (CRs) were routinely reviewed to assure that potential safety concerns and equipment problems were reported and resolved. Good plant equipment material conditions and housekeeping continued to be observed throughout the report period.

In general, the conduct of operations was risk informed, professional, and safety conscious.

01.2 Refueling Activities

a. Inspection Scope (71707)

The inspector monitored refueling activities during RFO-18, including those related to the failure of the reactor building side upender lifting cable.

b. Observations and Findings

Core reload activities were started on March 27, 1998, in accordance with procedure FHP-006, Fuel Assembly And Insert Handling During Core Loading, revision 8. Twenty-one assemblies were transferred to the reactor vessel from the spent fuel pool without incident. On March 28, at approximately 4:27 a.m., while upending new fuel assembly, AA16, with the reactor side upender, the lifting cable associated with the upender failed, while the upender was in the vertical position. The upender and the transfer basket (with the fuel assembly) pivoted back (gravity fall

with water resistance) to the horizontal position on to the fuel transfer cart. The licensee immediately stopped fuel movement activities. No change in radiological conditions was noted as monitored by the area monitors as well as in samples drawn from the cavity water.

The licensee formed an ERT and initiated condition report CR 98-00736. The investigation revealed that the shaft that coupled the hoist (motor) drive to the programmable limit switch (resolver) had sheared. The hoist/resolver shaft was a two-piece shaft pinned together. The failure caused the resolver to also fail, resulting in the upender "FRAME-UP" and "UP OVERTRAVEL" limits to not come in. This resulted in the hoist continuing to run, pulling the upender cable past its intended position. The tension on the cable increased as the hoist continued to run beyond the "UP OVERTRAVEL" limit, causing the #2 sheave/pulley, mounted on the refueling cavity wall, to be partially pulled off its support plate. This rotation of the sheave caused the cable to contact the keeper, which increased the stress on the cable, resulting in the snapping of the cable. The upender and the transfer cart (with the new fuel assembly) pivoted back to the horizontal position with 51 inches of the cable attached. The hoist continued to reel the cable onto the drum.

The ERT was not conclusively able to determine what stopped the motor. However, the ERT did conclude that a diverse means of stopping the hoist motor did not function as intended. This means was through a proximity switch which actuates when a ball mounted on the upender cable comes in proximity to the switch. This diverse means of stopping the hoist did not function as the ball was positioned such that it fell short in actuating the sensing proximity switch. A modification had been performed in December, 1993 which had replaced the ball and the associate proximity switch. Post modification testing from this modification did not adequately demonstrate that the proximity switch would limit the upper travel of the hoist. Additionally, Engineering Surveillance Test, EST-030, Fuel Handling Equipment Interlock and Operation Test, was performed prior to operation of the upender. This EST includes testing of protective features provided by the resolver and the proximity switch. However, the correct position of the ball relative to the proximity sensor was not verified, nor was a functional test performed for the ball limit stop for the upender. 10 CFR 50, Appendix B, Design Control requires that measures shall provide for verifying or checking the adequacy of design. Further, 10 CFR 50, Appendix B, Inspection, Procedures, and Drawings requires that procedures include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to these requirements, the post modification procedure as well as the surveillance procedure associated with Fuel Handling Equipment did not verify the correct position of the ball relative to the proximity sensor. This resulted in the proximity switch not appropriately stopping hoist movement following the failure of the limit switch. This self identified , corrected, and non-repetitive violation is treated as an NCV, consistent with Section VII.B.1 of the NRC enforcement policy. This issue is documented as NCV 50-261/98-05-01: Upender Failure during RFO 18.

Licensee corrective actions included replacement of the sheared shaft with a newer design, replacement of the #2 sheave/pulley and cable, inspection of the spent fuel side hoist/resolver shaft, and repositioning and functional testing of the proximity sensor to correct distance from the magnetic ball. The licensee also inspected portions of the refuel movement apparatus on the spent fuel pool side. The ball/limit on the spent fuel side did not need any adjustment. The licensee also has plans to further enhance EST-30 prior to the next planned use of fuel handling equipment.

Following completion of repair and testing, refueling activities were commenced and completed without any incidents. A new fuel assembly was procured as a replacement. Assembly AA16, was transferred back to the spent fuel pool and current plans are to send it back to the manufacturer (Siemens).

c. Conclusions

A NCV involving inadequate testing of refueling equipment was identified. The direct safety consequences of the fuel assembly drop were none. Had an irradiated assembly been subject to a similar drop, any release of radioactivity would have been contained, due to containment closure requirements during fuel movement. Licensee follow up to this event was appropriate.

01.3 Plant Startup From Refueling

a. Inspection Scope (71707)

The inspector monitored startup activities for RFO-18.

b. Observations and Findings

On April 13, plant startup activities were in progress in accordance with procedure GP-003, Normal Plant Startup form Hot Shutdown to Critical. At approximately 5:56 a.m., tremors were felt in the plant. A 3.9 (Richter Scale) seismic event was recorded at a location approximately 25 miles from the plant. The site seismic recorder did not register any seismic activity as it was below the trigger value. The licensee entered Abnormal Operating Procedure 21, Seismic Disturbances, as a result of the tremor. Startup activities were suspended and a general inspection of the plant was conducted to ascertain any consequences. After confirming no effect on the plant, startup activities were recommenced. The plant reached Mode 2 at 2:10 pm and Mode 1 at 6:31 pm on April 13. The unit reached 100 percent power on April 18.

c. Conclusion

The inspector concluded that the management decision to suspend startup to verify potential effects of the seismic event were conservative. The

inspector did not note any affect to the plant as a result of the seismic event. Operator performance during startup was considered good.

02 Operational Status of Facilities and Equipment

02.1 Reactor Trip and Subsequent Unit Startup

a. Inspection Scope (71707)

Robinson Unit 2 experienced an automatic reactor trip from 100% power on April 25, 1998 at approximately 1:34 p.m. The resident inspector responded to the site.

b. Observations and Findings

The first out annunciator was low-low (16%) "A" Steam Generator (SG) level. A review of Emergency Response Facility Information System (ERFIS) data archived after the trip indicated that the initiating event was the closing of the turbine governor valves. This led to a shrink in all the SGs caused by reduced steam flow and resultant higher steam pressure. The shrink was enough to reduce the SG levels to below the low-low SG level reactor trip set point.

Control room operators initially noticed SG level deviation alarms, concurrent with control rods inserting rapidly. Reactor Coolant System (RCS) Tref. as well as first stage impulse pressure were also noted to be dropping rapidly. The operators' initial diagnosis was that a first stage pressure channel had failed low. Thus, they reacted by placing control rods in manual. All four governor valves were then noted shut and the control room shift supervisor (CRSS) directed manually tripping the reactor. Just prior to manually tripping the reactor, the reactor automatically tripped on low-low SG level.

An Event Review Team (ERT) was formed to investigate the reactor trip. Following extensive troubleshooting, the licensee was not able to positively identify the root cause. However, during troubleshooting of the Electro Hydraulic Control (EHC) system, it was revealed that a 35 psi change in impulse pressure output (PT-1359), when the governor valves are open 90%, will cause the governor valves to close. The licensee did not identify a cause for the 35 psi change in impulse pressure nor were they able to confirm that any change in impulse pressure or PT-1359 had actually occurred. The above scenario was thought of as the most likely scenario for the unexpected closing of the governor valves. A Westinghouse representative assisted the licensee in the troubleshooting process.

Post trip plant response was as expected. A Power Operated Relief Valve (PORV) opened momentarily to relieve RCS pressure. Reactor operator response to the initiating event was incorrect. The RO reacted thinking the turbine first stage pressure channel had failed low. He did not incorporate the sharp decline in the generated megawatts in his

decision. This led him to place the rods in manual, thus stopping inward rod motion. This incorrect response did not have any overall impact on the plant as the reactor tripped shortly thereafter. However, the licensee plans to include this scenario during future operator training activities.

As corrective action, the licensee replaced impulse pressure transmitter PT-1359. Further, the licensee started and operated the plant in the "impulse pressure out" Mode. The EHC system was instrumented to capture any additional anomalies. No anomalies in EHC operation were noted during start-up and for the remainder of this inspection period.

c. Conclusion

Reactor operator response to the failed closed turbine overnor valves was incorrect. Licensee plans to review this event to enhance future training activities. Overall plant response was appropriate. Reactor startup activities following the trip were uneventful.

04 Operator Knowledge and Performance

04.1 Missed Technical Specification Surveillance (61726, 71707)

a. Inspection Scope

The inspector reviewed circumstances related to a missed TS surveillance. The missed surveillance, OST-014, Local Leak Rate Testing (LLRT) of Personnel Air Lock Door Seals, placed the unit in TS 3.0.3. This condition was identified by the control room shift supervisor (CRSS).

b. Observations and Findings

OST-14 was successfully performed on April 10, 1998, as a prerequisite for entering Mode 4. OST-14 ensures containment air lock operability per TS 3.6.2. Further, TS 5.5.16, Containment Leakage Testing Program, requires implementation of a containment leak rate testing program in accordance with 10 CFR 50, Appendix J. 10 CFR 50, Appendix J, Section D.2.ii and iii requires that air locks opened during periods when containment integrity is required by TS shall be tested within three days after being opened; and for air lock doors opened more frequently than once every three days, the air lock shall be tested at least once every three days during the period of frequent openings.

The unit entered Mode 4 at approximately 8:16 p.m. on April 10, at which time the operability requirement, for the containment air lock became effective. Further with the air lock door utilized for personnel entry/exit, the three day testing requirements were effective in accordance with 10 CFR 50 Appendix J. The next performance was due to be performed by 4:45 a.m. on April 13. This was written and tracked on the white board above the CRSS desk in the control room.

On April 13 at approximately 9:32 a.m., upon noticing the note on the white board, a CRSS questioned it. At this time it was recognized that OST-14 had not been performed as required between April 10 and April 13. Upon identification, the plant entered TS 3.0.3 and immediately requested performance of OST-14, which was successfully completed at 11:15 a.m. on April 13.

The licensee initiated CR 98-00890 which concluded that the primary cause for the missed surveillance was inadequate administrative controls to ensure event based surveillance requirements. The term "event based" implies surveillances that are not scheduled through the Surveillance Tracking System.

As corrective action, the licensee plans to revise OMM-007 to track similar surveillances in the Equipment Inoperable Record (EIR) logs. This would allow a positive tracking control, rather than a note on the board. The inspector will review operator performance in this area to determine similarity and effectiveness of corrective actions. NCV 97-12-01: Failure to Log TS Surveillance Completion in accordance with OST-20 was reviewed. This NCV was attributed to inattention to detail and log keeping accuracy on the part of the operator when the control rod insertion limit monitor (RILM) was inoperable. NCV 97-12-02: Failure to Verify Dose Equivalent I-131 in accordance with GP-005, involved a missed TS surveillance, also due to lack of operator attention to detail and inadequate supervisory oversight during periods of high control room activities. The inspector determined that the recent failure to perform OST-14 was partially attributable to inattention to detail on the part of the operating crew. This failure to perform the required surveillance in accordance with 10 CFR 50, Appendix J is identified as violation 50-261/98-05-02: Failure to Perform Personnel Air Lock Test.

c. Conclusion

A violation for failure to perform a required surveillance test on the containment personnel air lock was identified.

07 Quality Assurance In Operations

07.1 Plant Nuclear Safety Committee and Nuclear Assessment Section Oversight

a. Inspection Scope (40500)

The inspector evaluated certain activities of the Plant Nuclear Safety Committee (PNSC) and Nuclear Assessment Section (NAS) to determine whether the onsite review functions were conducted in accordance with TS and other regulatory requirements.

b. Observations and Findings

The inspector periodically attended PNSC meetings during the report period. The presentations were thorough and the presenters readily responded to all questions. The committee members asked probing questions and were well prepared. The committee members displayed understanding of the issues and potential risks. Further, the inspector reviewed NAS audits and concluded that they were appropriately focused to identify and enhance safety.

c. Conclusions

The inspector concluded that the onsite review functions of the PNSC were conducted in accordance with TSs. The PNSC meetings attended by the inspector were well coordinated and meetings topics were thoroughly discussed and evaluated. NAS continued to provide strong oversight of licensee activities.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments (62707,61726)

The inspector reviewed/observed all or portions of the following maintenance and/or surveillances and reviewed the associated documentation:

- OST 401-2, Emergency Diesel Generator Slow Start
- OST 402-2, Emergency Diesel Fuel Oil Flow Test
- OST 302-1, Service Water System Component Test
- Service Water Leak Repair Activities

Maintenance and surveillance activities observed were performed satisfactorily. The inspector noted good controls of housekeeping and good supervisor oversight of work activities.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Service Water System Leak Repairs (62707)

a. Inspection Scope

The inspector reviewed and observed licensee actions related to a leak in the service water (SW) system. The SW system is a safety related system. The discharge flow to the plant from the four SW pumps is through two headers, the North and South header. These two headers constitute the two redundant trains. This leak was in the North header and was approximated to be 10 gpm.

b. Observations and Findings

A leak in the SW system was suspected when a water puddle near the vicinity of the radwaste system was noted on March 27, 1998. A portion of the SW header is located underground in the vicinity of the radwaste Building. The licensee confirmed that the leak was in the SW system by chlorinating the SW headers and sampling the leaked water. Following extensive excavation, the licensee located the leak.

The leak was repaired by a welding a "cap assembly" over the section of the small hole. The welding repair activities were conducted through ESR 98-00202. The ESR documented the 10 CFR 50.59 evaluation associated with the repair. The repair was successfully completed in accordance with ANSI B31.1-1980. The root cause of the leak was determined to be general corrosion, not particularly attributed to any single mechanism.

c. Conclusions

The repairs on the SW header were appropriately completed. Management oversight of this problem was considered good.

M4 Maintenance Staff Knowledge and Performance

M4.1 Motor Driven Auxiliary Feedwater Pump 'A' Trip During Testing (61726, 62707)

a. Inspection Scope

The inspector reviewed circumstances related to the tripping of the "A" Motor Driven Auxiliary Feedwater Pump (MDAFW) during performance of Operations Surveillance Test (OST)-163, Safety Injection Test and Emergency Diesel Generator Autostart on Loss of Power and Safety Injection.

b. Observations and Findings

During performance of OST-163, both the MDAFW pumps auto started as designed. The valve configuration during the test precluded flow to the SG. For the purpose of the test, the manual valves on the discharge lines were maintained CLOSED.

Approximately 11 minutes into the OST, the "A" MDAFW pump tripped on low discharge pressure. Subsequently, the "A" MDAFW pump was restarted. Four minutes following the restart, the "A" MDAFW pump tripped again on low discharge pressure. OST-163 was continued by adding a component cooling water pump to provide the compensating electrical load to complete the OST.

Following completion of the OST, the licensee formed an Event Review Team (ERT) to determine cause of the "A" MDAFW pump trip. The ERT noted that the "A" MDAFW pump casing temperature was approximately 189 degrees F one hour after the trip, indicative of liquid flashing to steam within

the pump. During an initial system walkdown, the system engineer determined by visual inspection that the recirculation line flow restriction orifices (RO-1400A and RO-1400B) on the "A" and "B" MDAFW pumps were installed backwards. Although the orifice plates were reversed, the system engineer did not expect that this was the cause of the trip, since the flow through the orifice was not changed significantly by the reverse installation. The insignificant change in flow was based on the flow orifice design.

The ERT subsequently developed an action plan to determine the nature of the event, a problem definition, and an investigation process to gather any other data from the event. During this investigation, the ERT determined that the during the OST, the "B" MDAFW pump started first as designed, since the E-1 bus had to be energized by EDG-A before the A train sequencer could sequence loads on. The E-2 bus remained on off-site power for this part of the test. The "B" MDAFW pump developed full pressure in a few seconds. Since the cross connects were open and the isolations valves to the steam generators were closed, this pressure transmitted to the discharge side of the "A" AFW pump check valve. This caused the check valve to not open or to partially come off the seat, thus reducing flow from the "A" MDAFW pump. This was further substantiated by ERFIS data. This data indicated the presence of flow in the pump discharge headers downstream of the recirculation lines. This was consistent with flow from the "B" AFW pump, to the cross connect header, to the "A" AFW pump recirculation line. This "deadheaded" the "A" AFW pump resulting in insufficient flow from the pump.

The ERT concluded that the AFW pump had insufficient flow resulting in increased temperatures until the liquid in the "A" MDAFW pump casing flashed. This resulted in heating of the casing and reduction in pressure until the pump tripped. The cause of the loss of flow was concluded to be the effect of deadheading the "A" MDAFW pump by "B" AFW pump pressure, preventing opening of the discharge check valve. The ERT also concluded that the system was not susceptible to a similar problem during normal operation and when the MDAFW pumps are required to be operable because of a different valve configuration. The "A" MDAFW pump was externally inspected for any possible damage and none was found. Further, the vendor was contacted with the data gathered. Per the vendor recommendations, the pump was successfully started and no changes in vibration and temperatures were noted. The pump was subsequently declared operable.

With regard to the restricting orifices installed backwards, the licensee determined that during RFO-18, the two orifices were disassembled to perform an inspection. Following the inspection, the orifices were re-installed backwards. The inspection and installation activities were conducted under a work request(WR). The WR did not specify any particular orifice orientation, however, the orifices were clearly stamped identifying the inlet orientation. This job was performed by a shared resources maintenance mechanic from the Brunswick plant. The work was considered within the skill of the craft.

Notwithstanding, procedure MMM-001, Maintenance Administration Program, section 3.1.1 required the craftsmen to restore an area to its design condition following completion of maintenance activities. Further, section 2.4.1 of MMM-001 required the mechanic to "THINK" and "APPLY" a healthy skepticism to review each step of the job before doing to prevent errors. Contrary to the requirements of MMM-001, absent clear written instructions, the mechanic did not appropriately conduct the skill of the craft maintenance activities involving orifice installation. This resulted in the backward installation of two orifices on the auxiliary feedwater system. The backward installed orifices did not negatively impact system performance and thus the overall significance of the condition was minor. The inspector determined that failure to follow MMM-001 was a violation. This licensee identified, corrected, and non-repetitive violation is treated as an NCV, consistent with Section VII.B.1 of the NRC enforcement policy. This issue is documented as NCV 50-261/98-05-03: Backward Orifice Installation on MDAFW system.

c. Conclusions

The "A" MDAFW pump was deadheaded during a test configuration. This did not cause damage to the pump. Licensee actions to determine root cause, including formation of an ERT, were appropriate. Two orifices on the MDAFW pump recirculation lines that had been worked on during the outage were installed backwards due to inattention to detail on part of the mechanic. This did not contribute to the pump deadheading.

III. Engineering

E1 Conduct of Engineering

E1.1 Review of Modifications

a. Inspection Scope (37551)

The inspectors reviewed portions of modifications 97-469 and 97-382.

b. Observations and Findings

During Refueling Outage 17 (RFO-17) the Containment Air Recirculation Cooling system (HVH) unit damper control scheme was changed to have the normal damper closed and the emergency damper open with no safeguards signal damper repositioning required post accident. This configuration reduced the performance of the containment coolers. NRC violation 50-261/97-12-04 was issued regarding the post maintenance testing of this modification. The licensee's January 21, 1998, violation response indicated that the pre-RFO-17 configuration would be restored for the HVH unit dampers. Modification 97-382 was implemented to complete the restoration. This modification changed the HVH unit damper configuration to normal damper open and emergency damper open with the

normal damper closing on safeguards signal. The inspector noted that the modification did not address the impact of the new HVH unit damper configuration scheme on the HVH unit fan motor current.

Modification 97-469 was an EQ related modification which replaced the cable to solenoid valve SVA33 for the normal damper of HVH unit HVH-1 with an environmentally qualified cable. The inspectors reviewed the modification package and verified the package adequately addressed technical and environmental qualification requirements for the replacement cable. The inspectors performed a walkdown and verified that hardware and equipment were installed in accordance with modification requirements.

c. Conclusions

Modification packages reviewed were acceptable. Modification 97-382 implemented corrective action to restore the damper control scheme for the Containment Recirculation Cooling system units.

E2 Engineering Support of Facilities and Equipment

E2.1 Review of Environmental Qualification Program

a. Inspection Scope (37551)

The inspector performed a walkdown of selected inside containment Environmental Qualification (EQ) Program components and reviewed several of the associated EQ Data Packages (EQDPs).

b. Observations and Findings

Selected EQ components were inspected to verify their EQ sealing requirements were met. Twenty eight transmitters with Patel conduit seals and twenty one solenoid valves with Patel conduit seals were inspected. The inspectors determined that the equipment sealing for the components inspected met the requirements of TMM-019, List of Environmentally Qualified Electric Equipment, revision 29, CM-310, Installation of Patel Conduit Seals, revision 10, and EQDP 21.0, Patel Conduit Seals. Additionally, eight limit Switches with Namco EC210 conduit seals were inspected. The inspectors determined that the equipment sealing for the components inspected met the requirements of TMM-019, List of Environmentally Qualified Electric Equipment, TMM-036, Environmentally Qualified Electric Equipment Required Maintenance, and EQDP 25.0, Namco Limit Switches.

The following EQDPs were reviewed:

EQDP 3.0 - Rockbestos Cabling
EQDP 21.0 - Patel Conduit Seals
EQDP 11.2 - Kerite Cabling
EQDP 28.0 - Brand Rex Cabling
EQDP 25.0 - Namco Limit Switches
EQDP 40.0 - Ram-Q Cable Connector Assemblies

The inspectors reviewed the Nuclear Records Control System (NRCS) database to determine all ESRs which were posted against the above listed EQDPs. All ESRs posted against the listed EQDPs had been incorporated into the EQDPs except for current RFO-18 open outage ESRs. No backlog of unincorporated ESRs was noted.

c. Conclusions

The EQ components inside containment which were inspected were being maintained with qualified seals in accordance with EQ program requirements. The EQDPs reviewed were being updated in accordance with procedure EGR-NGGC-156, Environmental Qualification Of Electrical Equipment Important To Safety. No backlog of unincorporated ESRs was noted.

E2.2 Westinghouse BF Relays

a. Inspection Scope (37551)

The inspectors reviewed the licensee's activities related to resolution of Westinghouse BF relay performance discrepancies.

b. Observations and Findings

In January 1997, BF relay 412C1 failed during surveillance testing. The relay actuated correctly when denergized but experienced a delay during re-energization. The relay was a normally energized relay used in a Reactor Protection System (RPS) Overpower/Overtemperature (OP/OT) Delta T application. Significant Condition Report (CR) 9700092 was initiated for resolution and root cause evaluation. The relay was sent to the Harris Energy and Environmental Center (HEEC) for failure analysis. The analysis concluded that the energization delay was due to the relay armature pin contacting the internal surface of the relay casing.

A 1979 vendor technical bulletin documented a problem with BFD (DC) relays experiencing armature pin binding. No problem was identified with BF (AC) relays. The vendor evaluation indicated that this was an isolated case but instituted a relay improvement to epoxy the armature pin to prevent movement.

Relay 412C1 was one of 46 relays replaced in 1988 due to relay contact deterioration. The replacement relays did not have epoxy on the

armature pin to restrict pin movement. The corrective actions developed in CR 9700092 consisted of pro-active measures to replace the relays. During RFO-18 twenty percent of the BF relays in RPS and Safeguards System were scheduled to be replaced with nuclear qualified relays. Additionally the licensee will analyze and evaluate the BF relays removed in RFO-18 and determine if additional BF relay replacements are necessary.

During BF relay testing during RFO-18, relay TC-432B1-XB failed to go to the energized position. The relay would denergize and drop out but was erratic in its pull in times when energized. Upon disassembly, the armature pin was found binding on the side of the relay internal surface. This relay was also an OP/OT Delta T relay which had been replaced in 1988. A significant CR (98-751) was initiated for resolution. The common factors between the failures in January 1997, and March 1998, were that both relays were BF relays from the same 1988 procurement and both relays were used in RPS OP/OT Delta T input relay applications which received more frequent cycling due to more frequent testing than the other RPS and Safeguards BF relays.

The inspectors reviewed procurement records and verified that the failed relays were from the same procurement purchase and installed on the same work orders in 1988. The inspectors reviewed the procurement issue history and determined that the licensee had relays during 1988 without epoxy on the armature pin. The licensee developed an action plan for relay replacement to address relays from the same procurement batch and those with higher cycling rate and those which were energized to complete the safety function. Forty one BF relays were scheduled to be replaced in the RPS and 15 relays in Safeguards System during RFO-18 to implement the corrective actions of CR 9700092. Based on the second failure of a BF relay, an additional group of relays was replaced to address the relays from the 1988 procurement activity. The new replacement relays were purchased nuclear grade. Samples of the new relays were inspected and none were noted without the epoxy on the armature pin.

c. Conclusions

The licensee was conservative in evaluating a BF relay failure during RFO-18 testing and accelerating relay replacements. The relay testing discrepancies were being dispositioned in accordance with corrective action program requirements.

E7.1 Special UFSAR Review (37551)

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR related to the areas inspected. The inspector verified that for the select portions of the UFSAR reviewed, the UFSAR wording

was consistent with the observed plant practices, procedures and/or parameters.

E8 Miscellaneous Engineering Issues

E8.1 (Open) Unresolved Item 50-261/98-05-05: Questions on Design Calculations

NRC Inspection Report 50-261/98-03, dated February 6, 1998, included an apparent violation for a number of calculational deficiencies identified in the NRC Design Inspection. A Notice of Violation was issued on March 4, 1998, for these calculational deficiencies. The licensee response of April 3, 1998, to the Notice of Violation, stated that examples 4,6,12,13, and 14 of violation B and example 1 of violation D were not violations of NRC requirements. These violation examples are withdrawn and will be tracked as Unresolved item 50-261/98-05-05, Questions on Design Calculations, pending further NRC review of the information provided in the April 3, 1998 violation response.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Conduct of Radiological Protection Controls (83750)

a. Inspection Scope

Radiological controls associated with RFO-18 were reviewed to verify that the licensee was effectively implementing the radiation protection program and meeting 10 CFR Part 20, Standards for Protection Against Radiation requirements. In particular, the inspectors reviewed and evaluated the adequacy of general housekeeping, radiological controls, area radiological postings, radiation worker compliance with radiation protection procedures, and controls of radioactive materials.

b. Observations and Findings

Overall, the inspector noted appropriate levels of housekeeping and cleanliness within the observed work areas and radioactive material storage areas. Housekeeping practices within the Containment Building (CB) were considered acceptable. However, the inspector did find loose debris and standing water on portions of the first floor of the CB.

The inspector made independent radiation surveys of areas, equipment, and containers within the Radiation Control Area (RCA). The licensee's radiation survey results compared well with the inspector's surveys. Licensee radiological area postings met posting requirements for the areas surveyed by the inspector and were consistent. Vacuum cleaners, portable air filtration systems, and containers of radioactive materials within the RCA were properly labeled. Observed radiological controls met licensee and NRC requirements and were good overall.

During the tours within the RCA, the inspector found all portable radiation survey meters and air sampling equipment in use possessed valid calibration stickers. Radiation and contamination monitoring equipment had recent response checks.

The inspector observed radiation workers performing various tasks in the Unit 2 CB, Fuel Building, Reactor Auxiliary Building and yard areas within the primary RCA. The inspector discussed radiological work controls with Health Physics (HP) staff. Radiological controls for the various tasks were appropriate for the radiological conditions. Observed radiation worker and HP interactions were good.

On March 14, 1998, an employee working in the CB failed to immediately exit a radiation area when his Electronic Personal Dosimeter (EPD) alarmed. The EPD was designed to give an audible alarm for the user when either a dose rate was too high or a predetermined dose limit was reached. The worker's dosimeter was set to alarm at a dose rate of 200 mrem/hr and a dose of 100 mrem. The alarm showed the worker had exceeded an authorized radiation dose for the job he was working. After the EPD alarmed, the worker remained in the radiation area approximately 10 additional minutes to complete his assigned task with his dosimetry in continuous alarm. The worker exceeded the authorized dose for the job by approximately ten mrem. While the employee's actions did not result in exceeding any regulatory limits, the employee failed to follow guidance provided in the licensee's radiation protection training and procedures. Specifically, the incident was a violation of paragraph 12 of Attachment 3 to licensee procedure DOS-NGGC-0016, Electronic Personal Dosimeter System Operation, Rev 2. As stated in the licensee's procedures, upon activation of the dose alarm workers should exit the area immediately and report to radiation control.

The licensee documented the event as a significant condition in CR 98-00572. The licensee investigated and took immediate corrective actions. Corrective actions to prevent recurrence were also made. The inspectors found the licensee's corrective measures were very good. This non-repetitive, licensee-identified, and corrected violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1 of the NRC Enforcement Policy. NCV 50-261/98-05-04, Radiation Worker's Failure to Comply with Radiation Protection Procedures

c. Conclusions

Housekeeping and cleanliness within the RCA were acceptable.

Overall, the inspector observed good radiological controls and radiation worker compliance with those controls throughout the inspection.

A Non-Cited Violation was identified for radiation worker's failure to comply with radiation protection procedures.

R1.2 As Low As Reasonability Achievable (ALARA)(83750)

a. Inspection Scope

The licensee's goals, plans, and implementation of the site ALARA program was reviewed.

b. Observations and Findings

The 1997 collective dose goal was established at 22 person-rem. That goal included a contingency exposure budget of 3.619 person-rem. However, the licensee did not need the contingency as the plant was operational for approximately 360 days in 1997. The licensee ended 1997 with an annual collective dose of approximately 13 person-rem. That was the lowest annual exposure ever at the H. B. Robinson site. The 1997 annual collective dose was also the lowest for any Carolina Power and Light (CP&L) nuclear facility. The licensee attributed the record to detailed planning, ALARA culture, excellent operational performance, and good plant chemistry.

Annual and RFO collective dose goals for 1998 were set at 165 and 155 person-rem respectively. When the inspection began, most of the outage work had been completed and the licensee was approximately 10 person-rem above the dose the licensee had projected for that day in the outage. The inspectors reviewed the status of jobs resulting in significant collective radiation exposures. Most tasks were being completed with collective doses well within budget. However, three tasks had significantly contributed to the increased collective dose. The licensee had not planned to repair a main flange gasket on the "C" reactor coolant pump that resulted in approximately eight person-rem. Actual person-hours for sludge lancing and Steam Generator (SG) eddy current activities were approximately three and two times projected hours for those projects respectively. As a result, sludge lancing exceeded the projected dose of eight person-rem by eight person-rem and the eddy current projected dose of 6.6 person-rem was exceeded by approximately 2.4 person-rem. A new vendor was used for the sludge lancing activities with unanticipated work and poor planning contributing to the dose over run. Steam generators were empty longer than expected and higher dose rates adversely affected the eddy current project. Through the end of inspection, the licensee had closed the gap between the actual and projected outage dose. Outage collective doses were approximately two person-rem above the projected dose.

The licensee continued to improve the process for accurately assigning doses to very specific tasks. The licensee began the process of assigning dose to specific work request and job orders in 1996 and was continuing to improve its use. The process had not been fully implemented and the implementation progress had been slow. The process when fully carried out could be a valuable tool for the ALARA and plant management staffs. With full implementation the licensee would have more accurate time and dose information for planning activities.

Management support of the ALARA program was evidenced in the increased staffing levels of the outage ALARA group, increased use of remote monitoring equipment, visibility of ALARA goals, application of shutdown chemistry controls, and activities of the Robinson ALARA Committee.

c. Conclusions

The effectiveness of the licensee's dose reduction efforts in non-outage periods during 1997 were very good and had resulted in the site's lowest annual collective dose. The 1997 collective dose was 13 person-rem.

Overall, ALARA planning efforts were appropriate and were effectively implemented for most outage work activities. Unanticipated problems and poor planning resulted in excess dose of 13 person-rem for three of the thirty-one planned projects.

R1.3 Personnel Contamination Controls (83750)

a. Inspection Scope

Work activities in the licensee contaminated areas were observed by the inspectors to verify the licensee was implementing appropriate personnel contamination controls. Personnel contamination reports were reviewed to determine the frequency of Personnel Contamination Events (PCEs) and the adequacy of the licensee's response to the events.

b. Observations and Findings

Good contamination control measures were observed during the inspection. Inspectors found the contamination controls appropriate for the contamination levels and the work being performed. Radiation workers were properly wearing anti-contamination clothing as specified on Radiation Work Permits (RWPs). Good use of containments and engineering controls to reduce transport of radioactive contamination were also observed.

The licensee documented PCEs for all contaminations having radioactivity greater than 100 corrected counts per minute. The number of PCEs documented by the licensee in 1994, 1995, 1996, and 1997 were 54, 129, 207, and 14 respectively. The frequency of PCEs increased during outage periods. The PCE goal for 1997 was 50. Plant operability was very good in 1997 and there were only a few outage days. The fourteen PCEs for were the lowest in the site's history. The annual and RFO goals for 1998 were 90 and 70 respectively. As of April 2, 1998, the numbers of PCEs documented were 60 and 53.

The inspectors found the Environmental and Radiation Control (E&RC) Manager was reviewing all PCEs and had sorted and characterized the PCEs looking for trends and causes. Inspectors also reviewed the PCE reports and found approximately half involved discrete radioactive particles. The inspectors determined the licensee had taken measures to control the dispersion of the particles. No other PCE cause categories were

distinguishing. The radioactivity of the particles were all low and only a few had resulted in skin doses.

c. Conclusions

Overall licensee contamination control measures were effective in containing radioactive byproduct contamination and minimizing radiation exposures to the contamination.

Personnel contaminations were down from previous outages. The licensee evaluated the events to identify their causes and was took corrective actions to reduce the number of personnel contaminations. The number of PCE's documented in 1997 were the site's lowest.

R7 Quality Assurance in RPC Activities

R7.1 Radiological Protection Program Self Assessments (83750)

a. Inspection Scope

Self Assessments of the radiation protection program were reviewed to verify the licensee was identifying and correcting radiation protection program problems.

b. Observations and Findings

The inspectors reviewed several self assessments of various radiation protection programs completed in 1997. The inspectors found the licensee's self assessments of radiation protection programs were critical and findings were being identified. All findings were documented in a corrective action program and the staff was doing a good job of tracking and trending problems identified in the process.

c. Conclusions

Licensee use of self assessments in the radiation protection program area was good.

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.

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 May 19, 1998. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Boska, Manager, Operations
H. Chernoff, Supervisor, Licensing/Regulatory Programs
T. Cleary, Manager, Maintenance
J. Clements, Manager, Site Support Services
J. Keenan, Vice President, Robinson Nuclear Plant
R. Duncan, Manager, Robinson Engineering Support Services
R. Moore, Manager, Outage Management
J. Moyer, Manager, Robinson Plant
D. Stoddard, Manager, Operating Experience Assessment
R. Warden, Manager, Nuclear Assessment Section
T. Wilkerson, Manager, Regulatory Affairs
D. Young, Director, Site Operations

NRC

B. Desai, Senior Resident Inspector
M. Ernstes, Acting Branch Chief, Region II

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 83750:	Occupational Radiation Exposure used

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
NCV	98-05-01	Open	Upender Failure during RFO 18 (Section O1.2).
VIO	98-05-02	Open	Failure to Perform Personnel Air Lock (Section O4.1).
NCV	98-05-03	Open	Backward Orifice Installation on MDAFW system (Section M4.1).
NCV	98-05-04	Open	Radiation Worker's Failure to Comply with Radiation Protection Procedures (Section R1.1)
URI	98-05-05	Open	Questions on Design Calculations (Section E8.1)

Closed

Type	Item Number	Status	Description and Reference
NCV	98-05-01	Closed	Upender Failure during RFO 18 (Section O1.2).
NCV	98-05-03	Closed	Backward Orifice Installation on MDAFW system (Section M4.1).
NCV	98-05-04	Closed	Radiation Worker's Failure to Comply with Radiation Protection Procedures (Section R1.1)