UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report No.: 50-261/96-02

Licensee: Carolina Power & Light Company
P. O. Box 1551
Raleigh, NC 27602

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

Facility Name: H. B. Robinson Unit 2

Inspection Conducted: January 21 - March 2, 1996

Inspector: Jerry Wiseman 3/27/96
for W. T. Orders, Senior Resident Inspector Date Signed

J. Zeiler, Resident Inspector
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M. Janus, Resident Inspector, Brunswick Nuclear Plant
W. Kleinsorge, Region II Inspector, Paragraph 3.2 and 3.3
W. Rankin, Region II Inspector, Paragraph 5.2
H. Whitener, Region II Inspector, Paragraphs 2.5, 3.1.2, 3.2, and 5.4

Approved by: M B Shymlock 3-27-96
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Reactor Projects Branch 4
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SUMMARY

Scope:

Inspections were conducted by resident and regional inspectors in the areas of plant operations which included adverse trend of equipment mispositioning events, effectiveness of licensee control in identifying, resolving, and preventing problems, and close out of open issues; maintenance and surveillance which included on-line leak sealant repair of a main steam governor valve, Hagan room air conditioning maintenance, maintenance procedure review, retest of service water valve FCV-1608A, dedicated shutdown diesel generator testing, and close out of open issues; engineering which included accumulator level calibration error and close out of open issues; and plant support which included physical security program, radiation control organization, training, and staffing; Environmental and Radiation Control audits and self assessments; internal and external exposure controls; control of radioactive material and contamination; and program for maintaining

ENCLOSURE

exposures as low as reasonably achievable, fire protection program, and close out of open issues.

Results:

Plant Operations

Several valve mispositioning incidents occurred indicating an adverse trend in inattention to detail and lack of good self-checking. Management actions to address these incidents were comprehensive, however, continued management emphasis toward arresting this adverse trend in human performance problems was warranted (paragraph 2.3). A Non-Cited Violation (50-261/96-02-01) was identified for an operator's failure to follow a tagout procedure which resulted in a portion of both heat trace channels for the Boric Acid Transfer System flowpath to the reactor coolant system being deengerized (paragraph 2.3).

Maintenance

The maintenance procedure upgrade program appeared to be comprehensive. The procedures reviewed were technically correct, with the exception of a procedure for performing freeze seals. There were a number of human factors errors and inconsistencies which were identified. The procedure change backlog was well managed and consistently met established goals. The licensee was actively pursuing an improved contractor control program which addresses previous weaknesses related to inadequately qualified, trained, and managed contractors (paragraph 3.2).

Engineering

A Non-Cited Violation (50-261/96-02-02) was identified for inadequate design control involving the failure of the Safety Injection accumulator transmitter scaling calculation to compensate for the weight of pressurized nitrogen. The calculation error caused accumulator level indication to be lower than actual level, and resulted in accumulator level being greater than Technical Specification limits (paragraph 4.1). Similarly, a Non-Cited Violation (50-261/96-02-03) was identified for inadequate design control involving an inadequate engineering evaluation which allowed the Engineered Safety Features High Steam Flow setpoint to be changed in a non-conservative direction (paragraph 4.2).

Plant Support

In the Security area, three unresolved items (URIs) were identified. The first URI (50-261/96-02-04) involved questions on the adequacy of post-modification and periodic testing of the recently implemented hand geometry access control system (paragraph 5.1.1). The second URI (50-261/96-02-05) involved the safeguards classification of documents found in an unlocked declassified storage location (paragraph 5.1.2). The third URI (50-261/96-02-06) involved the apparent lack of adequate corrective actions for degraded protected area mast light equipment (paragraph 5.1.3).

The radiation control program was determined to be well implemented. An effective ALARA program continues to reduce cumulative dose to record lows for the site. The Environmental and Radiation Control organization continues to be organized effectively and staffed sufficiently to support radiation protection program requirements. Licensee audits and self assessments effectively identified and corrected adverse conditions. Self assessment was considered a radiation protection program strength. Internal and external exposure controls were effective in limiting maximum individual exposures to a small fraction of regulatory limits (paragraph 5.2).

In the fire protection area, an individual was assigned to the fire brigade who was not qualified due to lack of adequate training verification (paragraph 5.3).

REPORT DETAILS

Acronyms used in this report are defined in paragraph 8.

1.0 PERSONS CONTACTED

Licensee Employees:

*Clark, B., Manager, Maintenance
*Clements, J., Manager, Site Support Services
Crook, D., Senior Specialist, Licensing/Regulatory Compliance
*Gudger, D., Senior Specialist, Licensing/Regulatory Programs
*Hinnant C., Vice President, Robinson Nuclear Plant
*Keenan, J., Director, Site Operations
*Krich, R., Manager, Regulatory Affairs
*Meyer, B., Manager, Operations
*Miller, G., Manager, Robinson Engineering Support Services
*Moyer, J., Manager, Nuclear Assessment Section
Stoddard, D., Manager, Operating Experience Assessment
Warden, R., Superintendent, Plant Support Assessment
Wilkerson, T., Manager, Environmental Control
Young, D., Plant General Manager

Other licensee employees contacted included office, operations, engineering, maintenance, and chemistry/radiation personnel.

2.0 PLANT OPERATIONS (71707 and 92901)

2.1 Plant Status

The unit operated at or near full power for the entire report period with no major problems.

2.2 Plant Operations Observation Activities

The inspectors evaluated licensee activities to determine if the facility was being operated safely and in conformance with regulatory requirements. These activities were assessed through direct observation of ongoing activities, facility tours, control room observations, discussions with licensee personnel, evaluation of equipment status, and review of facility records. The inspectors evaluated the operating staff to determine if they were knowledgeable of plant conditions, responded properly to alarms, and adhered to procedures and applicable administrative controls. Selected shift changes were observed to determine that system status continuity was maintained and that proper control room staffing existed. Routine plant tours were conducted to evaluate equipment operability and to assess the general condition of plant equipment. No discrepancies were identified.

2.3 Adverse Trend of Equipment Mispositioning Incidents

Recently, there have been several valve mispositioning incidents which individually, had minimal safety significance, but raised concerns with the inspectors regarding the adverse trend of inattention to detail and lack of good self-checking. The inspectors reviewed these incidents, as well as

licensee corrective actions that had been initiated to date. These incidents are discussed below.

- Motor Driven Auxiliary Feedwater Pump Recirculation Valves

  The first major valve mispositioning incident was identified January 17, when two manual Motor Driven Auxiliary Feedwater pump recirculation valves were found approximately 5 turns from their required full open position. This incident was discussed in detail in NRC Inspection Report 50-261/96-01 and was the subject of Violation 50-261/96-01-01. The licensee had not been able to determine conclusively why the valves were not in their required position. The licensee believed that the valves may have vibrated 5 turns in the close direction when the pumps were operated during inservice testing. The inspectors were concerned that the change in valve configuration was not detected during subsequent valve position lineup checks. Several operators who had performed these checks indicated they may have used visual observation of the valve stem to verify the position of the valves. Visual observation of stem position was allowed by procedure, although the preferred method involved hands-on verification. In this case, visual observation may have contributed to the failure to identify that the two auxiliary feedwater valves were mispositioned.

  As a result of this incident, the licensee revised their procedures governing manual valve position verification methods. It is no longer acceptable to verify valve position using visual observation of valve stem position. Instead, hands-on position checks are required for all manual valve position verifications.

- Vacuum Pump B Valve Mispositioning

  On February 3, the control room operators observed that condenser vacuum was slowly decreasing. At this time, vacuum pump B was in service and vacuum pump A was out-of-service for replacement of a pump motor bearing. The operators entered the Abnormal Operating Procedure AOP-012, Partial Loss of Condenser Vacuum or Circulating Water Pump Trip, and appropriately performed actions to correct the condition. As part of these actions, the OAO was notified to check the condition of vacuum pump B. The OAO had just completed condenser inleakage testing in the vacuum pump room and had noticed that the vacuum pump B auto makeup bypass valve (CAR-21), was partially open. This valve was normally closed. The OAO alerted the control room operators of the earlier observation and was directed to close the valve and drain the Vacuum Pump B seal water tank. Condenser vacuum returned to normal within 20 minutes.

  The licensee determined that the decrease in condenser vacuum resulted from CAR-21 being partially open. This manual ball valve can be used to manually fill the vacuum pump B seal water tank. Normally this valve is closed and the seal water tank level is controlled by a float valve. Opening CAR-21 allows demineralized water to fill the seal water tank. If left open, the seal water tank will eventually overflow, with the

spillover directed to an overflow line routed to the floor drain. When the OAO performed condenser inleakage testing, the vacuum system was aligned such that the seal water tank spillover was diverted into the vacuum pump ejector inlet line. This caused the pump to partially flood out, reducing its effectiveness to maintain a constant condenser vacuum.

The licensee determined that, most likely, the CAR-21 operating handle was bumped by the OAO on the previous night shift while on routine rounds in the vacuum pump room.

- Steam Driven Auxiliary Feedwater Instrument Control Valve

On February 7, while troubleshooting a problem with the SDAFW pump, the licensee discovered that the isolation valve to pressure controller PIC-3120 was closed. The function of this controller is to regulate the speed of the SDAFW pump in order to maintain pump discharge pressure higher than Steam Generator pressure. The controller isolation valve was required to be open in accordance with operating procedure OP-402, the system lineup procedure for the SDAFW system. The valve mispositioning did not affect pump performance as it apparently leaked by its seat enough for the controller to detect changing system pressure and perform its function. The licensee initiated Significant CR 96-00287 to investigate how the valve was mispositioned. It was determined that, most likely, the valve was not realigned properly by I&C personnel following controller calibration activities in September 1995. Weakness were identified with the calibration procedure used (PIC-203), in that, it did not contain adequate controls to ensure proper isolation and restoration of the isolation valves. The procedure contained general steps requiring the controller to be isolated and unisolated, but, failed to describe the specific valves to be manipulated or provide signoffs documenting the completion of these activities.

To address the calibration procedure weaknesses, the licensee revised I&C procedure MMM-042, Documentation of Temporary Lead Lifts and Jumpers. The procedure revision required I&C personnel to document the manipulation of all valve, switches, etc., that are not specifically controlled and documented by existing procedures or work package instructions.

- Boric Acid Heat Trace Deenergized Due to Tagout Error

At 8:16 a.m. on February 20, while implementing LCTR 96-00177 for removing the BATS Pump B from service, operations personnel removed the incorrect heat trace circuit fuses for BATS Pump A. Instead of removing heat trace circuit fuses "1" Primary (1P) and "1" Secondary (1S), associated with BATS Pump B, the fuses for heat trace circuits "E" Primary (E-1P) and "E" Secondary (E-1S), associated with BATS Pump A, were removed. BATS Pump A was the dedicated boric acid flow path to the reactor coolant system. This error was identified when the operators who removed the incorrect fuses observed the temperature for heat trace circuits E-1, was unexpectedly decreasing. The BATS tank A was placed in recirculation in order to raise the temperature in the boric acid

flowpath. At 10:58 a.m., following discovery of the error, the fuses for heat trace circuits E-1P and E-1S were reinstalled and the correct fuses were removed.

The licensee initiated Significant CR 96-00438 to investigate this incident. The main cause of the error was attributed to an inadequate assumption made by the operators who manipulated and independently verified the LCTR alignment. The LCTR listed the fuses to be manipulated as "HT CKT 1P FUSE" and "HT CKT 1S FUSE(S)." The fuses that were incorrectly manipulated were labeled in the plant as "E1-P" and "E1-S." While the operators recognized that there was a labeling discrepancy between the LCTR and plant equipment. They discussed the differences among themselves, and incorrectly decided that the equipment was the same.

On February 21, Operations Night Order 96-012 was issued to reinforce the expectation that the Control Room or Shift Supervisor be contacted for LCTR and plant equipment labeling differences, as well as reinforce the responsibilities of the independent verifier to reach a conclusion based on independent means from the person performing the task. Planned licensee corrective actions included: 1) sharing the lessons learned from the incident with operations personnel via the operator training program, and, enhancing the labeling of the boric acid heat trace fuse boxes. In addition, CR 96-00574 was initiated to evaluate the recent adverse trend in operations human performance related errors.

TSs 3.2.2.d and 3.2.2.e respectively require one flow path from the boric acid storage tanks to the reactor coolant system, and two channels of heat tracing be operable. TS 3.2.3.d only allows one channel of heat tracing to be inoperable and for an allowed outage time of 24 hours. Therefore, during the period that heat trace circuits E-1 were removed and inoperable, the licensee had inadvertently entered the eight hour action statement to place the reactor in hot shutdown in accordance with TS 3.0.

The heat tracing associated with the BATS is designed to maintain temperature above 145°F in order to prevent boric acid precipitation. The licensee determined that the lowest temperature reached was 149°F. Based on this, the safety significance of this event was minimal since the ability to add boric acid to the reactor coolant system was not affected.

Failure to follow LCTR 96-00177 was identified as a violation of TS 6.5.1.1, Procedures, Tests, and Experiments, which requires that written procedures be established, implemented, and maintained, covering the activities recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, 1978, including procedures for controlling the tagout of safety related systems. This licensee identified and corrected violation is being treated as a non-cited violation, consistent with Section VII of the NRC Enforcement Policy. This Non-Cited Violation is identified as NCV 50-261/96-02-01: Failure to Follow LCTR Resulting in Deenergization of Boric Acid Transfer Pump Heat Tracing.

Due to the adverse number of component mispositioning events discussed above, the licensee initiated a program to re-verify manual valve lineups of safety-related and safety-significant systems. Existing system lineup checklists were utilized to perform this re-verification. In addition, plant management conducted a "Stand Down" with each department. The stand down re-emphasized the need for good self-checking techniques, as well as provided the following specific instructions to address these incidents: 1) prior to re-positioning any plant equipment, it must be clearly authorized by the operations department, 2) component re-positioning must be documented in logs or work packages if not specifically addressed by procedures, logs, etc., and, 3) in planning and personnel performing work must be sensitive to adjacent equipment to ensure that the activity being performed does not inadvertently reposition adjacent components.

The inspectors determined that the licensee had adequately identified the causes of these incidents. The licensee's corrective actions that have been implemented or are planned, were considered adequate. However, as a result of the subsequent human performance errors which occurred following the plant standdown, continued management attention was warranted to address the continuing adverse trend in human performance errors. Licensee management demonstrated sensitivity and concern regarding these incidents as evidenced by the initiation of CRs to address the recent adverse trend. The inspectors will continue to monitor the licensee's effort to arrest this situation.

2.4 Effectiveness of Licensee Control in Identifying, Resolving, and Preventing Problems

The inspectors evaluated certain activities of the Plant Nuclear Safety Committee to determine whether the onsite review functions were conducted in accordance with TS and other regulatory requirements. In particular, the inspectors attended the January 24, 1996, meeting. It was ascertained that provisions of the TS dealing with membership, review process, frequency, and qualifications were satisfied. The minutes from these meetings were reviewed to confirm that decisions and recommendations were accurately reflected.

2.5 Close Out Items

(Closed) LER 94-019-00: TS Violation Due to Exceeding Pressurizer Cooldown Rate

This issue involved a condition prohibited by the plant TS. Specifically, that the TS 3.1.2.3. limit for pressurizer heatup and cooldown had been exceeded. Notice Of Violation 50-261/94-23-02 was issued for this event on November 28, 1994. The inspectors reviewed the licensee's response to the violation dated December 27, 1994, and LER 94-019, Revision 1.

The corrective action for Violation 94-23-02, specified in the licensee's response to the NOV, had been reviewed by the NRC and found to be adequate and properly implemented. The Violation was closed out in NRC Report 50-261/95-30.

The inspectors verified that the corrective action specified in the response to the NOV was the same as that specified in LER 94-019. Consequently, LER 94-019-00 is closed out based on the previous review of the implementation of corrective action.

3.0 MAINTENANCE (61726, 62703, and 92902)

3.1 Maintenance Observations

The inspectors observed safety-related maintenance activities on systems and components to determine if the activities were conducted in accordance with regulatory requirements, approved procedures, and appropriate industry codes and standards. The inspectors reviewed associated administrative, material, testing, and radiological control requirements to determine licensee compliance. The inspectors witnesses and/or reviewed portions of the following maintenance activities:

3.1.1 On-Line Leak Sealant Repair of Main Steam Governor Valve

On February 9, the inspectors witnessed maintenance activities associated with the on-line leak sealant repair of turbine governor valve MS-GV1. The valve had developed a body-to-bonnet steam leak. The licensee decided to inject the valve with Furmanite sealant material to stop the leak. The leak sealant activity was conducted under WR/JO 96-AAFE1 using Temporary Modification 96-34. The inspectors reviewed the temporary modification package. The package included an adequate safety evaluation of the proposed repair method, as well as adequate installation instructions. The installation instructions had been prepared by the contractor performing the repair, but, were reviewed and approved by the licensee. Calculations were performed for determining the maximum allowable injection pressure, volume of space to be injected, as well as the amount of leak sealant compound to be injected. The amount of the sealant made available to the contractor was limited to that amount calculated to fill the space. In order to inject the sealant material into the gasket area of the body-to-bonnet joint a seismically evaluated temporary clamp was installed to cover the joint area. The licensee planned to make permanent valve repairs during the next refueling outage. The leak repair activity was properly controlled and performed in accordance with the temporary modification procedure. No discrepancies were identified.

3.1.2 Hagan Room Air Conditioning Maintenance

The inspectors observed portions of the implementation of Work Request WR/JO 95-APLI1 which required maintenance to provide material and labor for relocation of the thermostat for air conditioning condensing units ACC 2A and ACC 2B. The WR/JO refers to Minor Modification ESR 95-01014, Revisions 0 and 1, for the detailed instructions for relocating the thermostat. Additionally, this modification added a relay in the thermostat control circuit for each ACC unit to bring the voltage drop in the control circuit within the vendors recommendation.

Although Units ACC 2A and ACC 2B are not classified as safety-related equipment, these units do provide cooling for the safety-related Hagan

instrumentation room and the Computer room (also referred to as the cable spread room and relay room). As such, the thermostat (TSA41) which is located in the Computer room does function to support the environmental control of safety-related equipment. Relocation of the thermostat from the Computer room to the Hagan room which has a much higher heat load will prevent the broad temperature swings in the Hagan room while having little effect on the larger Computer room. The licensee believed that these temperature swings were related to degradation/malfunction experienced with the Hagan protection modules. Temperature related issues with the Hagan room equipment has been tracked as the Number 1 priority item on the "Top Ten" Equipment Issues List. Installation of this modification completes a major milestone for licensee actions to address this problem.

The inspectors reviewed the minor modification, ESR 95-01014, Revisions 0 and 1, and concluded that the instructions for implementation were adequate. The modification referenced specific procedures, provided an impact review for Appendix R requirements, Station Blackout equipment, Seismic considerations, Heat Load considerations, and a 10 CFR 50 safety screening evaluation. Instructions for post-modification testing were also included.

The inspectors observed cable pulling, verified the proper routing was specified and followed, determined that fire penetrations were resealed, and observed installation of the new relays on the ACC units. When the cover was removed from ACC 2A to install the new relay the I&C technician identified an unlanded jumper on the unit control circuit which was not taped or tagged. A Condition Report (CR) was issued to identify the cause and appropriate action.

3.2 Maintenance Procedures

The NRC's Systematic Assessment of Licensee Performance for the period December 26, 1993 through June 17, 1995, indicated that inadequate maintenance procedures continued to contribute to performance problems, and deficiencies continued in the control of contractors. To evaluate the licensee's effectiveness toward improving maintenance procedures and their effectiveness, the inspectors reviewed procedures, interviewed licensee personnel and examined selected records as indicated below.

3.2.1 Procedure Upgrade Program

In late 1988 and early 1989 there were several indicators that maintenance procedures did not meet industry standards for technical content and human factors. There were self-identified Adverse Conditions where the cause was identified as inadequate procedures. An Institute of Nuclear Power Operation (INPO) review identified that maintenance procedures needed to be upgraded to improve content and human factors considerations. Also, NRC findings identified the need to improve maintenance procedures.

To control the upgrade process and to obtain consistent results, the licensee developed MI No. 018-1, Maintenance Procedure Upgrade Process. The objective was to develop technically accurate procedures which contained adequate, detailed instructions and addressed human factors. Cross-discipline reviews were performed by teams including representatives from maintenance foreman and

crew members, system engineers, system planners, operations, ALARA, and quality assurance as considered appropriate. The process specified that the procedures for each system component would be evaluated by desk review, actual performance, and procedure walk through. Feedback methods were provided for reviewer comments. The program was completed in May 1993, with a total of 543 procedures being upgraded or developed.

3.2.2 Procedure Review

The inspectors reviewed the below listed maintenance procedures to confirm that the procedures were prepared to adequately control maintenance of plant equipment within regulatory requirements. Observations were made in the areas of technical content and human factors. Observations were compared with the FSAR, Technical Specifications, vendor technical manuals, set point calculations, and other licensee documents.

Procedures Reviewed

Procedure	Rev/(Date)		Title/Subject
AP-004	48	09/30/95	Procedure Control
AP-006	7	04/26/95	Procedure Use and Adherence
AP-022	24	12/28/95	Document Change Procedure
CM-033	8	04/24/95	Charging Pump Power Frame Maintenance
CM-101	14	05/09/95	Quick Change Trim Air Operated Control Valves Maintenance
CM-127	18	05/30/95	Valve Packing Using the Chesterton Packing System
CM-617	3	01/11/94	DSD Starting Air Compressor
LP-017	6	11/06/93	Pressurizer Level Protection and Control (Refueling Interval)
MI-018-1	-	02/12/90	Maintenance Upgrade Process
MI-506	6	02/02/96	Maintenance Procedure Permanent Revision/ New Procedure Development Guidelines
MMM-001	30	11/23/95	Maintenance Administration Program
MMM-002	15	07/18/95	Maintenance Procedure Preparation
MMM-003	47	12/05/95	Maintenance Work Requests
MST-005	14	11/01/94	Pressurizer Water Level Protection Channel Testing (Monthly)
PLP-013	6	07/22/95	Maintenance Program
PM-029	1	09/24/91	Planned Valve Repacking Program
PM-176	0	02/23/95	Packing 'nForcer Testing Procedure
SPP-001	6	12/31/93	Freeze Plugging of Piping Requiring Brittle Fracture Control (Normally Carbon Steel)

Relative to the procedure review discussed above the inspectors noted the following:

- Of all the procedures reviewed, only the APs were marked with the usage level (Information, Reference, or Continuous Use). All the remaining procedures were marked *"Determine Procedure Use PRIOR To Each Use."* All

the plant technical procedures (CM, MST, LP, PM, and SPP) reviewed by the inspectors, were of such complexity that they clearly meet the criteria for Level I - Continuous Use for all applications. Marking those procedures *"Continuous Use"* would eliminate the need to complete a "PROCEDURE USAGE COVER SHEET" prior to each usage, when every usage of those procedures are by definition "Continuous Use."

- Lack of consistency in the definitions of data entry points. An example: CM-033 defined the boxed V as "Technician and Independent Verifier" and MST-005 defined the boxed V as "Operator and Technician" . The MST-005 definition does not convey the notion of "Independent Verification" as the CM-033 definition does.

- Steps in the body of some procedures are not as definitive as the sign-off points on the data sheets. Example: CM-101 Paragraph 7.4.43.4.

- Formulas and the sources of the inputs to the formulas were not included on some data sheets. Example: CM-101 Step 7.1.8.5.

- Sign-off instructions in the body of some procedures were different than those on the data sheets. Example: CM-033 step 7.11.3.

- CM-617 had insufficient sign-off and data input steps. The licensee indicated that they would make necessary corrections.

Relative to procedure SPP-001 the inspectors noted the following:

- It is not appropriate to substitute before and after freeze seal diameter measurements for before and after freeze seal Liquid Penetrant examinations, because elastic deformation caused by the expanding ice in the piping could damage the piping without a detectable change to the post freeze seal pipe diameter.

- The procedure did not specify where on the pipe circumference, to place the thermocouples.

- The procedure did not provide any guidance in the restoration of safety related pipe supports.

- The procedure did not prohibit concurrent multiple freeze seals with a single source of liquid refrigerant.

- The licensee did not have a documented standing contingency plan for freeze seal failure. A review of the WR/JO for the last freeze seal job revealed the following contingency plan: *"Install expand-a-seal plugs in the 1/2" pipe and the 3/4" elbow as directed by operations."* The licensee indicated that the freeze seal contingency was verbally expanded at the pre job briefing.

- The procedure does not make any provisions for evaluation by a qualified engineer of any potential structural damage.

3.2.3 Maintenance Procedure Change Backlog

The licensee continues to make progress in reducing the maintenance procedure change backlog. They started the year with a backlog of 400 and established a goal of 1 percent (a reduction of four) per month. With the exception of February and the three months following the summer refueling outage, the licensee consistently met their goal. The population of maintenance procedure revision requests grew significantly after the outage, but the licensee recovered by November and finished the year ahead of their goal. The licensee has established a goal of 2 percent per month decrease in the procedure change backlog for 1996, almost double the 1995 goal.

3.2.4 Maintenance Procedure Deficiencies

The inspectors reviewed all the Condition Reports (CR)/Adverse Condition Reports (ACR) for the year 1995 that were associated with maintenance procedure problems (Content, Presentation, or No Procedure) to evaluate: the cause; the licensees evaluation and corrective action; the licensee's determination of the problem extent and effect on other procedures; and actions to prevent recurrence.

Condition Reports/Adverse Condition Reports Reviewed

Event ID	Event Date	Description
95-00068	01/09/94	Procedure PIC-002 was determined to be confusing and became the subject of a Non-Cited Violation 94-27-02.
95-00337	01/26/95	MST-903 Station Battery Charger - Monthly was ambiguous and confusing. Procedure was revised.
95-00440	01/31/95	LP-705 was written to be accomplished at cold shutdown, but was scheduled to be worked on line. Problems with the procedure were identified when work started.
95-01156	05/13/95	Procedure FP-004 requirement for ensuring communications availability was not met when performing SPP-014.
95-01826	08/04/95	Thermocouple wires on circulating water pump motor were wired backwards resulting in the creation of a new procedure to verify proper termination.
95-01882	08/15/95	Followup review of an out-of-specification calibration data sheet was not completed in a timely manor. MMM-006 was revised to give more guidance and better criteria for the required actions when out-of-specification conditions are identified.

95-02440	10/12/95	Incorrect information in Appendix A to MMM-003 caused Procedures OST-302 and OST-306 not to be performed on valves V6-12C-MO and V6-12D.
95-02505	10/30/95	Liquid Penetrant pre-freeze seal inspection indication was not evaluated due to procedure ambiguity.
95-02650	11/03/95	Procedure PIC-605 specified incorrect lead time for TM-412.
95-02762	11/22/95	Safety injection accumulator tank level problems became the subject of Violation 95-30-01.
95-02913	12/12/95	Oil level control mechanism for spent fuel pump "A" was found missing which necessitated a procedure change.

The licensee's actions relative to the 1995 procedural CRs were appropriate.

The inspectors evaluated the trend of procedure deficiencies for the past four years. The large number in early 1992, was the result of growing pains of the procedure upgrade program (new procedure discrepancies). The trend of procedure deficiencies was downward to start and then has leveled off. In general, the recent deficiencies are minor in nature. The current deficiencies appear to occur spuriously. This points to procedure inconsistencies being missed in the verification and validation effort.

3.2.5 Conclusion

The procedure upgrade program appeared to be comprehensive. The procedures reviewed with the exception of the freeze seal procedure, were technically correct, however there were a number of human factor errors and inconsistencies. The procedure change backlog is well managed consistently meeting established goals.

3.2.6 Administrative Control of Maintenance Contractors

As previously stated, assessment of licensee performance indicated continuing problems in the area of contractor control. The licensee has responded to this challenge through the development of a contractor control program. This program is in the developmental stage and could not be directly evaluated. However, the inspector discussed the planned program controls with licensee management and reviewed some of the elements of the new program. Through a revised and more detailed Contract Procedures Manual and detailed Work Scope Documents the elements of the program are to be clearly defined. These elements will include Contract Administrator and Task Coordinator training; contractor safety, qualification, performance monitoring, training, procedure requirements, pre-work meetings and planning; and CP&L expectations for contractor performance. The thrust of the program is better qualified, trained, and managed contractors and closer CP&L monitoring.

The inspectors concluded that the licensee is actively pursuing an improved contractor control program.

3.3 Surveillance Observations

The inspectors evaluated certain surveillance activities to determine if these activities were conducted in accordance with license requirements. For the surveillance test procedures listed below, the inspectors determined that precautions and LCOs were adhered to, required administrative approvals and tagouts were obtained prior to test initiation, testing was accomplished by qualified personnel in accordance with an approved test procedure, test instrumentation was properly calibrated, the tests were completed at the required frequency, and the tests conformed to TS requirements. Upon test completion, the inspectors verified that the recorded test data was complete, accurate, and met TS requirements, test discrepancies were properly documented and rectified, and the systems were properly returned to service. Specifically, the inspectors witnessed and/or reviewed portions of the following test activities:

3.3.1 Retest of Service Water Valve FCV-1608A

The inspectors witnessed the retest of the North Service Water Strainer Backwash Valve FCV-1608B in accordance with OST-302-2, Service Water System Component Test. In a test on December 28, 1995, a stroke time of 9 seconds was recorded. While this time is within the acceptance limit of 16 seconds for this valve, it did represent an increase of greater that 50% from the previous test of 5.82 seconds. In accordance with ASME Section XI (1986), the frequency of valve testing was increased from quarterly to monthly. The valve was re-based on June 30, 1994 at 7.2 seconds. Over a span of seven tests the valve performed in a range of 5.8 to 8 seconds. While this is about a 30% range of scatter, it appeared attributable to the test method. Valve FCV-1608B is a 3 inch ball valve located in the strainer pit at the intake canal. Controls to operate the valve are located some distance from the valve. Communication between the operator stationed to observe the valve stroke and the operator who activates the valve is by radio. The operator measuring the valve stroke starts timing the stroke when he is notified by radio and stops timing when he observes the valve shaft has stopped turning. The inspector concluded that this method of testing introduces some uncertainty in the measurement. The licensee agreed to review the design basis documentation to identify the significance of the valve function. Acceptable scatter for this valve will be further reviewed.

From observations during the test the inspectors concluded that the licensee performed well. A pre-test briefing was held in the control room. The current procedure was verified, the applicable portion of the procedure was identified and discussed by the shift supervisor, test responsibility was assigned, and test equipment was verified calibrated and operable. The test result was acceptable and was promptly reviewed by the Shift Technical Advisor.

3.3.2 Dedicated Shutdown Diesel Generator Testing

On January 25, the inspectors witnessed licensee testing of the DSDG performed in accordance with OST-910, Dedicated Shutdown Diesel Generator (Monthly). The 2450 kw DSDG exists to carry loads necessary to bring the plant to safe shutdown condition in the event of a fire (10CFR50, Appendix R) or Station Blackout (10CFR50.63) when all other power and control is lost. The test was performed to verify the operational readiness of the DSDG. Testing involved manually starting the DSDG, paralleling it to its normal bus, and running the engine at rated load for 60 minutes. The inspectors attended the operations pre-job brief held by the Control Room Supervisor in the control room prior to testing. The brief adequately detailed the scope, precautions and limitations, and responsibilities of those involved with the test. Adherence to proper self-checking techniques was emphasized by the Control Room Supervisor. The Auxiliary Operators performing the test appeared knowledgeable of the equipment operation and properly followed the test procedure. After the DSDG was run for 30 minutes loaded, engine operating data was taken in accordance with OST-910. Preliminary review of this data indicated that the engine operating parameters were within their normal range as specified by the procedure criteria. The inspectors did not witness the remaining portion of the test which included engine operation for an additional 30 minutes at loaded conditions and then 20 minutes at idle speed.

At a later date, the inspectors reviewed the completed procedure to verify proper signoffs and acceptable engine operating data. The inspectors noted that a test discrepancy had been identified. When engine speed was decreased to the idle condition, the low limit relay light illuminated at 480 rpm as opposed to the 440-460 rpm prescribed by the procedure. The low limit light illuminates when the low limit relay actuates which prevents further engine governor speed reduction. The operators initiated a Work Request for the low limit relay light illuminating earlier than expected, the test was designated as unsatisfactory, but, the DSDG was considered to be operable.

The inspectors discussed with system engineering personnel concerning the impact of the low limit relay actuating slightly above 440-460 rpm. The function of this circuitry is to limit engine speed to prevent lower than desirable hydraulic fluid flow in the electro-hydraulic governor. At speeds lower than 385 rpm, erratic governor operation can occur. Since the setting of the low limit relay determines the idle speed that the engine will reach when started, there was no potential adverse impact on engine operation should the idle speed be slightly higher than the desired 440-460 rpm range. Based on this information, the inspectors determined that the operators had adequately concluded that the DSDG was still operable with the discrepant condition. The licensee plans to perform additional testing of the low limit relay setpoint and recalibrate it if necessary when the engine is next operated for routine testing.

3.4 Close Out Issues

(CLOSED) Violation 50-261/95-13-01: Failure to Provide Adequate Work Instruction for Degraded Stud inspection

This violation concerned the licensee's failure to provide adequate work instructions for the inspection of the boric acid wastage of the "C" Reactor Coolant Pump studs. By letter dated July 17, 1995, the licensee admitted to the violation as cited, attributed the violation to personnel error, committed to a procedure revision and personnel training to be completed by September 9, 1995. The licensee's response has been reviewed and found acceptable by Region II. The inspectors surveyed the licensee actions and determined that the licensee had determined the full extent of the violation. The inspectors interviewed licensee personnel, reviewed the seismic stress analysis for the two failed studs Calculation RNP-C/STRU-1102, Engineering Service Request 9500433, Reactor Coolant Leakage on "C" RCP Studs, procedure PLP-40, Revision 7. Program for Prevention of Boric Acid Corrosion of RCS Carbon Steel Bolting (Generic Letter 88-05), dated September 29, 1995, and training records. Due to an administrative error, the licensee did not achieve full compliance until September 29, 1995, three weeks after the committed date. The delay in the full compliance date was communicated to the NRC by CP&L Letter dated October 8, 1995. The October 8th letter stated that "A review was conducted of commitments contained in 1995 violation responses, and no other administrative errors were found." The inspectors conducted an independent review. The inspectors determined that the licensee had taken appropriate actions to correct the violation and prevent a recurrence of similar circumstances. This violation was closed.

4.0 ENGINEERING (37551 and 92903)

4.1 Accumulator Level Calibration Error Results in Technical Specification Violation

On January 2, the licensee identified an error in the scaling calculation used in the Safety Injection accumulator level channel calibration. The level channel scaling calculation failed to take into consideration the weight of pressurized nitrogen resulting in a nonconservative scaling error of approximately 7 percent level (equivalent to five cubic feet of water). This resulted in the indicated accumulator level being lower (by 7 percent) than actual level.

Technical Specification 3.3.1.1.b requires each Safety Injection accumulator to contain at least 825 ft^3 and no more than 841 ft^3 of borated water during power operation. Technical Specification 3.3.1.2.a allows one accumulator to be isolated or otherwise inoperable relative to the requirement of 3.3.1.1.b for a period not to exceed four hours. Based on accumulator tank level data from June through November, 1995, all six accumulator level transmitters (two per accumulator) operated above the Technical Specification upper volume limit (equivalent to 80.4 percent level) at various time periods due to the calculation error.

The calculation error was identified while engineering personnel were discussing previously identified accumulator transmitter calibration procedure weaknesses and tubing configuration line problems with another utility. NRC Inspection Report 50-261/95-30, issued January 26, 1996, discussed details of these previous tubing configuration and calibration procedure weaknesses. As

a result of these problems, indicated accumulator level was higher than actual level. This was the subject of Violation 50-261/95-30-01 for an inadequate level transmitter calibration procedure.

On January 2, Night Order 96-001 was issued requiring the operators to maintain accumulator level below 73 percent level in order to account for the level error. Both the scaling calculation and calibration procedure were revised to account for the weight of nitrogen. On January 16, all accumulator level transmitters were recalibrated using the new calibration procedures. LER 95-009-00 and its supplement, 95-009-01 were submitted on December 28, 1995, and February 1, 1996, respectively, describing the calibration errors and the TS violation for exceeding accumulator level.

The inspectors reviewed Design Modifications 643 (implemented in 1984 to replace the Masonneilan transmitters with Rosemount transmitters) and Design Modification 911 (implemented in 1987 to upgrade to Environmentally Qualified Rosemount transmitters and reroute instrumentation tubing). Both of these modifications failed to consider the effects of pressurized nitrogen.

The inspectors also noted that there were several instances since implementing these modifications where opportunities existed to discover the calculation error. These included:

1) As part of the 1994 Engineering Setpoints Project (initiated to address problems relating to instrument setpoints), accumulator level setpoint calculation RNP-I/INST-1052 was developed. Neither the transmitter scaling calculation nor the loop uncertainty calculation considered the effects of the density of nitrogen.

2) Operating Experience Item OE-5101, "Error in SI Accumulator Level Indication Caused by Not Compensating for Effects of Nitrogen Density During Calibration," was received by the licensee on September 9, 1991. This operating experience item described an identical level scaling error that occurred at Kewaunee Nuclear Plant. The licensee incorrectly screened this operating experience item as not applicable to Robinson Nuclear Plant.

The licensee evaluated the safety effects of the scaling error and determined that the additional amount of water in the accumulators would have had minimal safety significance. During the most limiting Large Break LOCA, the additional water volume equates to an additional 0.4 seconds in the delivery time of water from the accumulators to the reactor core. This results in an 11°F increase in peak fuel cladding temperature. Based on previous sensitivity studies, the licensee determined that a margin of 194°F exists in the Large Break LOCA analyses. Therefore, the additional amount of water would have had no significant effect on containment temperature and pressure during a Large Break LOCA.

The failure of the accumulator transmitter scaling calculation to compensate for the weight of pressurized nitrogen was identified as a violation of 10CFR50, Appendix B, Criterion III, Design Control, which requires, that applicable regulatory requirements are correctly translated into

specifications, drawings, procedures and instructions. This licensee identified and corrected violation is being treated as a non-cited violation, consistent with Section VII of the NRC Enforcement Policy. This Non-Cited Violation is identified as NCV 50-261/96-02-02: Inadequate Accumulator Level Transmitter Scaling Calculation.

4.2 Close Out Issues

(Closed) LER 50-261/94-014-00 and -01: Technical Specification Violation Due to Improper High Steam Flow Setpoint

On June 14, 1994, with the unit operating at full power, an engineering review discovered that an incorrect assumption in a 1988 calculation had been used to establish Engineered Safety Features high steam flow setpoints. The main steam pressure instrumentation setpoints were rescaled in 1988 to a value that did not meet the TS requirements for high steam flow.

The corrective actions listed by this LER included rescaling the steam pressure instrumentation, Summators PM-446B and PM-447B, using the 1994 calculation which provided the basis for the setpoint as the design rated steam flow, which is based on assumptions of the Plant Parameters Document provided by the fuel vendor. Instrument setpoint calculations are now performed according to formal engineering guidelines. These calculations require an independent design verification to confirm that the correct inputs are provided.

The inspectors reviewed calculation No. RNP-I/INST-1045, Turbine First Stage Pressure Channel Accuracy and Sealing Calculation, and the data associated with completed WR/JOs 94-AILP1 and 94-AILQ1. This review verified that the steam instrumentation for Summators PM-446B and PM-447B had been set to the required values. This LER is closed.

The inadequate engineering evaluation which resulted in the incorrect high steam flow setpoint was identified as a violation of 10CFR50, Appendix B, Criterion III, Design Control, which requires, that applicable regulatory requirements are correctly translated into specifications, drawings, procedures and instructions. This licensee identified and corrected violation is being treated as a non-cited violation, consistent with Section VII of the NRC Enforcement Policy. This Non-Cited Violation is identified as NCV 50-261/96-02-03: Inadequate ESF High Steam Flow Setpoint Scaling Calculation.

5.0 PLANT SUPPORT (71707, 71750, 83750, and 92904)

5.1 Physical Security Program

The inspectors toured the protected area and observed the protected area fence, including the barbed wire, to ensure that the fence was intact and not in need of repair. Isolation zones were maintained and clear of objects which could shield or conceal personnel. Personnel and packages entering the protected area were searched by detection devices or by hand for firearms, explosive devices, and other contraband. Vehicles were searched, escorted,

and secured as required. Except as noted below, no deficiencies were identified in this area.

5.1.1 Hand Geometry System Implementation Review

On December 19, 1995, the licensee completed implementation of ESR Modification 94-00393, for installing hand geometry equipment to control unescorted personnel access into the protected area. Use of this alternative access control equipment was permitted via NRC approval of a license exemption from the requirements of 10CFR73.55(d), dated December 20, 1994. This exemption allowed an alternative unescorted access control system that would eliminate the need to issue and retrieve badges at the entrance/exit locations.

During this report period, the inspectors reviewed testing performed when the new access control equipment was installed. This included a review of modification package 94-00393, security testing procedures, and the vendor manual for the new equipment (Recognition Systems, Inc.). While the modification package contained a section describing "recommended" post-installation tests, it was unclear what testing had been performed. A detailed test procedure was not developed and there was no documentation describing details of the actual testing completed. Based on discussions with engineering personnel responsible for developing and implementing the modification, all of the recommended test items were completed. The inspectors were particularly interested in testing performed to verify proper operation of the interface between the cardkey reader and hand geometry reader. This should have included testing of valid (encoded) cardkeys with their respective valid hand counterparts, as well as valid cardkeys with invalid hands, to ensure proper system response. The inspectors had to rely on security computer historical records of card reader activity to verify that this testing was performed.

The inspectors also noted that security procedure SP-012, Verification of Security System Component Operation, was not included in the "documents affected" section of the modification package. This procedure is used to perform periodic testing of various components of the security system, including the access control system. Although not listed in the modification package, the inspectors noted that this procedure had been revised to incorporate changes in the access control system due to the new hand geometry system. However, revised testing was only limited to verification of the valid cardkey with valid hand function. The inspectors questioned whether other testing, such as, valid cardkey with invalid hand, was necessary to ensure that the system was operating properly. The licensee planned to contact the vendor, as well as their security counterparts at CP&L's Brunswick and Harris Nuclear facilities to determine the scope of necessary periodic testing to verify operation of the equipment. The inspectors will review the results of the licensee's efforts to resolve this issue during a subsequent inspection. This issue was identified as URI 50-261/96-02-04: Review Adequacy of Testing of New Hand Geometry Access Control System.

5.1.2 Potential Safeguards Material Found in Declassified Safeguards Filing Cabinet

On February 21, the inspectors toured the Central Alarm Station area. During this tour, the inspectors noted that a filing cabinet, which was known to have been locked and classified as "Safeguards" during previous tours of the area, was now unlocked and apparently de-classified and prevented from storage of safeguards material. However, the inspectors found several of the drawers to be filled with documents that were unknown as to their security classification. While none of the material sampled had been stamped with typical safeguards markings or numbers, some of the material had pages with typewritten security/safeguards references in the page headings. There were also numerous electrical drawings of the plant security equipment. While these drawings were old and appeared to be out-dated, the inspectors were concerned with the control of this material.

The inspectors informed the security manager, who immediately had the material moved to a safeguards storage facility. The inspectors later learned that a member of the security staff had recently removed all safeguards material (based safeguards information location records) that was known to be stored in the filing cabinet. The filing cabinet was then de-classified from a safeguards storage location and the lock was removed. However, the rest of the material had not been thoroughly examined prior to de-classifying the filing cabinet. The licensee indicated that the material would be examined to determine whether any would be considered safeguards. The inspectors will review the results of the licensee's examination during a subsequent inspection. Pending completion of this review, this issue was identified as URI 50-261/96-02-05: Review Licensee Examination of Potential Safeguards Material Found Unsecured.

5.1.3 Protected Area Lighting Equipment Degradation

On February 21, at approximately 6:15 p.m., the inspectors observed that the protected area mast lights were not on as expected for the decreasing daylight conditions which existed at the time. The inspectors contacted plant security personnel, who indicated that the automatic controls (photo-cell control device) for actuating the lights was not functioning properly. As a result of the photo-cell malfunction, the security staff had previously deactivated the automatic light control function and were manually energizing/deenergizing the lights as necessary each day.

The inspectors discussed the status of efforts to repair the degraded security equipment with the security manager. The security manager was unaware of the condition and immediately initiated actions to investigate the matter. Subsequent investigations revealed that maintenance had been performed on the photo-cell control circuitry in September 1995 in an effort to get the mast lights to turn on later in the evening and shut off earlier in the morning. Attempts to adjust the photo-cell setting were unsuccessful. Until a replacement photo-cell could be acquired, temporary repairs included taping of a portion of the photo-cell sensor (to limit light exposure) to adjust the photo-cell response to the desired setpoint. Sometime after this work, the security shifts began operating the lights manually due to concerns that the

lights were not energizing early enough in the evening. However, at this time, a Work Request was not issued to repair the light controls, nor were controls established to ensure that compensatory actions were implemented (manual light operation) for the degraded equipment. At the end of the report period, the licensee was still investigating details of the incident and why a work request and compensatory actions were not implemented when it was identified that a problem with the lights existed. The inspectors will review the results of the licensee's investigation upon completion. This issue was identified as URI 50-261/96-02-06: Review Results of Licensee Investigation of Degraded Protected Area Lighting Equipment.

5.2 Radiological Protection Program

5.2.1 Organization, Training, and Management Controls

The inspectors reviewed the licensee's organization, staffing levels, staff qualifications, and lines of authority as they relate to maintaining an effective E&RC organization. During 1995, the E&RC organization was reduced from 62 positions to 56 positions, which was the on-board strength of the E&RC organization at the time of the inspection. Within the E&RC organization, there are two main subordinate organizational units, Radiation Control and Environment & Chemistry, each of which has a Superintendent reporting to the Manager, E&RC. Within the Radiation Control organization there are currently 25 Radiation Control Technicians reporting to the Superintendent, Radiation Control. The licensee plans to implement in 1996 an expanded ARW training program that will supplement RC Technician resources needed for RC coverage on jobs with more routine RWPs. The licensee also plans to share E&RC staff resources with other CP&L sites during their peak outage workloads. During this inspection, five E&RC personnel from the Robinson site were assigned temporary duty at the Brunswick site to support the Brunswick outage which commenced February 2, 1996. Robinson had just completed 200 days of full power operations prior to this inspection and has no planned outage activity until a refueling outage scheduled for September, 1996.

The inspectors evaluated the adequacy of operational RC Technician/RC Analyst coverage at Robinson in view of the above changes in resource utilization and staffing levels and determined that adequate shift coverage was available to support operational requirements. No other significant changes had occurred within the E&RC organization and no other concerns were noted.

The inspectors reviewed training records as well as education and experience qualifications for selected RC Technicians assigned to the E&RC organization. Levels of education and experience, as well as the specific supporting health physics and related site training completed for each level of RC Technician certification held, were evaluated against specific training requirements. No discrepancies were identified. For the records reviewed, the inspectors determined that the technicians met or exceeded ANSI Standard N18.1 qualifications and had completed general employee training and specific job qualification training in accordance with procedural requirements. No concerns were noted.

5.2.2 Program Audits and Self Assessment

Program audits and self assessments were evaluated to determine the adequacy of the licensee's ability to identify problems and to take effective corrective action.

5.2.2.1 NAS Audits and Observations

10 CFR 20.1101 (c) requires that the licensee shall periodically (at least annually) review the RP program content and implementation.

TS 6.5.3.2.c requires audits of the facility to be performed by the NAS encompassing conformance of facility operation to the provisions contained within the TS and applicable license conditions at least once per 12 months.

The NAS staff conducted one major E&RC program audit in the radiological protection program area since the last inspection conducted May 30-June 2, 1995, as documented in Inspection Report 50-261/95-15. NAS Audit R-ERC-95-01, issued January 5, 1996, was an assessment of the site's radiation protection program and was conducted during the period of November 27 - December 8, 1995. The inspectors discussed the scope and findings of the audit with representatives of the licensee's E&RC staff. The assessment was thorough and sufficient in scope to effectively address the principal E&RC areas reviewed. The inspectors determined that the audit results were reported to appropriate management levels for review and corrective action, as appropriate.

The audit identified several issues and weaknesses requiring corrective action and response on the part of the E&RC organization. The audit cited examples of radioactive material not being controlled in accordance with applicable procedures. The audit report documented specific examples of increased risk for personnel contaminations and spread of contamination into clean areas. Areas of weakness identified included locked high radiation area key control, handling and storage of chemicals, routine monitoring of an area posted as a very high radiation area, and E&RC improvement initiatives not meeting timeliness expectations. Strengths included the capturing of specific Harris outage "lessons learned" for implementation at Robinson and identifying the E&RC self assessment program as the most effective self assessment program at the site.

A number of items for management consideration were also identified in the audit report. The inspectors inquired about the E&RC response to the audit and status of corrective actions. However, since the audit was recently issued, specific corrective actions associated with each of the findings were not complete and, hence, not evaluated. However, select corrective actions in response to NAS assessment findings from the January 1995 program audit and the 1995 outage assessment were evaluated for corrective action timeliness and adequacy. For each issue reviewed, licensee corrective actions to date were judged to be appropriate and timely.

5.2.2.2 E&RC Self Assessment

Licensee procedure PLP-057, Licensee Self Assessment Program, established a self-assessment program for plant staff with the purpose of involving plant staff in achieving higher standards. The inspectors noted that the E&RC staff conducted many self-assessments in 1995 in specific program areas which significantly exceeded procedural expectations. No significant concerns were noted.

5.2.2.3 Condition Reports

The inspectors reviewed the E&RC related Condition Reports data base, which documented an extensive variety of E&RC issues and deficiencies. These E&RC issues were assessed and classified by the staff according to the safety significance of each. All items requiring corrective action were entered into the site's corrective action tracking program. The E&RC staff continued to review all E&RC related CRs to trend problems and look for common root causes. Based on the review of E&RC CRs, no problems or concerns were noted. The inspectors determined that all corrective actions assigned and completed were appropriate and no significant adverse trends were identified. A review of timeliness statistics for corrective action completion disclosed that the licensee was meeting, on average, its 14 day goal for evaluation and 30 day goal for completion. A relatively small number of needed corrective actions exceeded completion timeliness goals.

Generally, E&RC personnel demonstrated a strong commitment to the self assessment and corrective action process. The staff's ability to identify adverse conditions and safety issues at a relatively low safety threshold, determine root causes, identify performance trends, and implement corrective actions in a generally timely manner is identified as an E&RC program strength.

5.2.3 External Exposure Control

5.2.3.1 Whole Body Exposure

10 CFR 20.1201(a) requires each licensee to control the occupational dose to individual adults, except for planned special exposures under 20.1206, to the following dose limits:

(1) An annual limit, which is the more limiting of: (i) The total effective dose equivalent being equal to 5 rems; or, (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems;

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) An eye dose equivalent of 15 rems; and, (ii) A shallow-dose equivalent of 50 rems to the skin or to any extremity.

The inspectors reviewed cumulative whole body exposures for plant and contractor employees and determined that all whole body exposures assigned since the previous NRC inspection of this area (May 30-June 2, 1995) were within 10 CFR Part 20 limits. The typical administrative dose limit was 2,000 millirem utility-acquired administrative dose limit plus the amount of year-to-date incoming dose. In 1995, the licensee granted no dose extensions, and the maximum individual whole body dose for the year was 1425 millirem and the maximum skin dose for the year was 15.84 rem.

The inspectors reviewed procedural controls of external and internal exposures and verified that the controls met applicable regulatory requirements and were designed to maintain exposures ALARA. The inspectors reviewed several RWPs utilized to control work within the RCA and noted that the radiological controls observed were appropriate for the described tasks and radiological conditions.

5.2.3.2 Personnel Dosimetry

10 CFR 20.1502(a) requires each licensee to monitor occupational exposure to radiation and supply and require the use of individual monitoring devices by:

(1) Adults likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits in 10 CFR 20.1201(a);

(2) Minors and declared pregnant women likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of any of the applicable limits of 10 CFR 20.1207 or 10 CFR 20.1208; and,

(3) Individuals entering a high or very high radiation area.

The inspectors evaluated the site's personnel dosimetry program to ensure the licensee was meeting the monitoring requirements of 10 CFR Part 20. During tours of the plant, the inspector observed proper positioning of TLDs and use of EDs.

Since the last inspection, the licensee has fully implemented the use of EDs and is now using EDs for occupational worker dose of record as required by regulation. The licensee plans to continue to reduce the use of TLDs and rely primarily on EDs except in certain situations, such as for work on non routine or higher dose RWPs, or for special situations such as containment entries at power. The licensee received approval from the NVLAP on September 19, 1995, for use and processing of EDs based on the licensee's demonstration of compliance with ANSI-N13.11. The inspectors discussed the use of EDs with licensee representatives as to the general reliability of electronic dosimetry and the regulatory concern that occupational dose be accurately measured. The licensee reported no known reliability problems with electronic dosimetry based on experience to date and indicated good

correlation had been achieved to date between cumulative dose tracked by EDs with cumulative dose tracked by TLDs.

5.2.3.3 High and Very High Radiation Areas

10 CFR 20.1601, 10 CFR 20.1602 and 10 CFR 20.1902 specify the control and posting requirements for high radiation areas and very high radiation areas. In addition, TS 6.13 provides additional requirements for the control of high radiation areas.

The inspectors reviewed and discussed with licensee representatives the program for controlling access to HRAs, LHRAs, and VHRAs. These areas were inspected during tours for proper postings and access controls. No areas were identified where required posting were needed but not posted, and the posting observed met procedural and regulatory requirements with no concerns noted. Areas controlled as LHRAs were inspected and found locked in accordance with licensee procedure. Key controls for entry into locked and very high radiation areas were evaluated against the requirements of Administrative Procedure AP-031. The inspectors examined keys to LHRAs in the control room as well as at the RCA access point, found all keys to be of consistent D-11 cores, and all keys controlled in accordance with procedure AP-031, Administrative Controls for Entry Into Locked and Very High Radiation Areas.

5.2.4 Internal Exposure Control

10 CFR 20.1204(a)(3) requires, in part, that the licensee, as appropriate, use measurements of radioactivity in the body, measurements of radioactivity excreted from the body, or any combination of such measurements as may be necessary for timely detection and assessment of individual intakes of radioactivity by exposed individuals.

10 CFR 20.1703(a)(3) permits the licensee to maintain and to implement a respiratory protection program that includes, at a minimum: air sampling sufficient to identify the hazard; surveys and bioassay to evaluate the actual intakes; testing of respirators immediately prior to each use; written procedures regarding supervision and training of personnel and monitoring, including air sampling and bioassays; record keeping; and determination by a physician prior to the use of respirators, that the individual user is physically able to use respiratory protective equipment.

The inspectors reviewed and discussed the licensee's bioassay program in general. Whole body counts were conducted and followed up as necessary, such as annually, at termination, and following certain types of PCEs as required per procedure. In 1994, the licensee had no assignable internal dose. However, during 1995 the licensee experienced at least eighteen instances where a measurable internal dose was incurred with a CEDE ranging from as low 1 millirem to a high of 54 millirem. Internal doses incurred during 1995 were largely attributable to outage work. The inspectors selected for review several of the jobs for which internal doses were incurred and determined that an appropriate pre job total effective dose equivalent (TEDE) ALARA analysis had been performed as appropriate based on the known rad hazards prior to job

initiation. In each case, the rad controls specified on the RWP were determined appropriate for the work performed based on the known airborne hazard and other rad conditions known to exist. Engineering controls were used as appropriate to reduce airborne radioactive material, and respirator reduction continued to be emphasized by the licensee as determined appropriate as a result of the TEDE ALARA analysis. The maximum exposures received by individuals during 1995 were determined to be minor and all exposures were well below regulatory limits of 5 Rem CEDE. No problems were noted by the inspectors during a review of select bioassay program records.

The inspectors reviewed records for selected employees who had recently worn respiratory protection equipment. The inspectors verified that for the records reviewed, each worker had successfully completed respiratory protection training, was medically qualified, and was fit tested for the specific respirator type used in accordance with licensee procedural requirements.

No violations or deviations were identified.

5.2.5 Surveys, Monitoring, and Control of Radioactive Material and Contamination

5.2.5.1 Surveys

10 CFR 20.1501(a) requires each licensee to make or cause to be made such surveys as (1) may be necessary for the licensee to comply with the regulations and (2) are reasonable under the circumstances to evaluate the extent of radioactive hazards that may be present.

The inspectors reviewed selected records of radiation and contamination surveys performed during 1995. During tours of the plant, the inspector observed HP technicians performing radiation and contamination surveys. No concerns were identified.

5.2.5.2 Posting and Labeling

10 CFR 20.1904(a) requires the licensee to ensure that each container of licensed material bears a durable, clearly visible label bearing the radiation symbol and the words "Caution, Radioactive Material," or "Danger, Radioactive Material." The label must also provide sufficient information (such as radionuclides present and an estimate of the quantity of radioactivity) to permit individuals handling or using the containers to take precautions to avoid or minimize exposures.

During tours of the plant and selected outside radioactive material storage areas, the inspectors noted that the licensee's posting and control of radiation areas, high radiation areas, contamination areas, and radioactive material areas was good and in conformance with regulatory requirements. The inspectors noted radioactive material observed throughout the facility was properly labeled. Minor instances of poor radiation safety work practices were observed, however, in that an empty but labelled "yellow bag" used for radiation material control

purposes was found mixed with waste controlled as clean in a "green bag". The licensee corrected the poor radiation material control practice promptly. Also, during inspection of the tool issuance room inside the RCA, some fixed contaminated tools were found mixed with clean tools, creating a slight potential risk of cross contamination on tools issued as clean. The licensee indicated tool room controls were being developed to properly segregate clean and contaminated tools.

5.2.5.3 Area Contamination Control

During facility tours, the inspectors noted that contamination control and general housekeeping practices were good. Surface contamination was aggressively being controlled at its source, as evidenced by the low number of catch containments utilized throughout the plant (only one being used in the Safety Injection pump room) and the relatively low amount of controllable contaminated area (1367 square feet as of January 30, 1996) in the RCA (approximately 87,000 square feet). Most of the contaminated square footage in the relatively small RCA was located in the waste hold up tank room, in the pipe alley, in the #1 Sump Tank Room, and in a gas stripper area. Overall, during 1995, the licensee controlled contaminated square footage to a site record low average of 918 square feet.

5.2.5.4 Personnel Contamination Events

The inspectors reviewed the licensee's PCEs for 1995 and evaluated the adequacy of related controls. A total of 129 PCEs, up from 54 PCEs in 1994, were documented in 1995. As of the date of this inspection, only one clothing PCE had occurred in 1996. The 1995 increase in PCEs was largely attributable to increased activity in contaminated zones during the 1995 outage. The most safety significant PCE was a hot-particle Co-60 skin contamination on a worker's lower back which resulted in an assigned skin dose of 14.3 rem. A review of the assessment methodology indicated conservative Varskin dose assessment assumptions utilized in the final dose assessment with no discrepancies identified.

The inspectors selectively reviewed PCE reports for 1995 and noted no assessment or procedural errors. Where a skin dose assessment was required by licensee procedure HPP-005, Control of Personnel Decontamination Techniques, Rev. 27, based on the level of skin activity in corrected counts per minute, the inspectors were able to verify the assessment had been performed as per procedure. Skin contaminations were assessed appropriately in accordance with procedure for the PCEs reviewed with no other concerns noted. Individuals with facial contamination were whole body counted as required to check for internal dose.

5.2.5.5 Radiation Detection and Survey Instruments

During facility tours, the inspectors noted that survey instrumentation and continuous air monitors in use within the RCA were operable and displayed current calibration stickers. The inspectors noted an

adequate number of survey instruments were available for use, and background radiation levels at personnel survey locations were observed to be within the licensee's procedural limits. Two instruments (one RO-2A and one Ludlum frisker) were identified with expired calibration stickers located in the instrument issuance area and not specifically stored. However, further checking disclosed the instruments were not indicated as available for use on the licensee's computer based instrument control system which would have reasonably precluded issuance in the event an RC Technician attempted check out in accordance with procedures.

The inspectors verified that selected radiation monitors observed were operable during plant tours. Additionally, a review of radiation monitoring system availability for area and accident monitors indicated a 1995 average percent availability of 99.3% for radiation monitors and 98% for accident monitors. Availability below 100% was explained primarily due to recalibration down time and for operability checks. No concerns were noted.

No violations or deviations were identified.

5.2.6 Program for Maintaining Exposures As Low As Reasonably Achievable

10 CFR 20.1101(b) requires that the licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable.

This area was evaluated to determine whether the licensee was establishing and tracking ALARA goals and objectives and to evaluate the overall effectiveness of the ALARA program.

Collective dose of 214.85 rem for 1995 was a record for the site for a refueling outage year. Also, collective dose in 1994, at 63 rem total, was the lowest dose in the site's history. The annual goal for 1994 was set at 58 rem initially but unplanned outages caused the actual site dose to exceed the planned dose goal that was established for the year. Similarly, the site collective dose goal for 1995 was set at 172 person-rem, which represented an aggressive dose goal, and was exceeded primarily due to emergent work performed (such as reactor coolant pump and containment vessel sump pump work) during the 1995 refueling outage.

The inspectors reviewed and discussed the implementation of the ALARA program with licensee representatives and noted that many initiatives to reduce overall dose were implemented in 1995 and more were underway or planned for 1996. During 1995 ALARA initiatives included: installation of shielding on reactor coolant system piping while at power; residual heat removal lines in containment were directly shielded rather than shadow shielded; outage plans were developed using a surrogate video tour system; and ALARA work plans were fully developed prior to the outage for plans with dose estimates greater than 5 rem. During 1996 specific dose reduction actions targeted for implementation include: chemical decon of the residual heat removal system;

expanded use of video monitoring; increased long term shielding; and expanded advanced rad worker training.

Overall, the inspectors concluded that the licensee's ALARA program was effective in controlling collective dose.

No violations or deviations were identified.

5.3 Fire Protection Program

The inspectors periodically reviewed aspects of the licensee's fire protection program including fire brigade staffing controls, flammable materials storage, housekeeping, control of hazardous chemicals, and maintenance of fire protection equipment. With one exception described below, no discrepancies were identified.

On February 6, the inspectors attended the operations morning turnover meeting. The fire brigade staffing complement is normally discussed at this meeting. At this particular meeting, the on-duty Fire Protection Technical Aide, who is responsible for ensuring that the fire brigade team is properly staffed, indicated that the on-duty chemistry technician was not fire brigade certified, and thus, could not serve on the fire brigade. As part of normal shift practice, the on-duty chemistry technicians complement the fire brigade. The Aide indicated that prior to the meeting, he had made arrangements for a Unit 1 fire brigade qualified individual to serve on the brigade, however, that individual would not be available until several hours. At this time, an SRO in-training and an Auxiliary Operator volunteered to fill the position until the Unit 1 individual became available. The SRO in-training indicated that he had recently completed the annual fire brigade training the previous week making him fire brigade qualified. This was accepted and the meeting was ended.

Following the meeting, the inspectors reviewed the list of qualified fire brigade team members dated January 8, 1996. While the Auxiliary Operator was listed, the SRO in-training was not. This was brought to the attention of the Fire Protection Technical Aide, who then recognized that the SRO in-training individual could not serve on the fire brigade.

The inspectors later discussed the fire protection training requirements with the responsible training coordinator. It was learned that SRO in-training had completed the required annual Fire Protection Academy Training, but, still lacked other remedial classroom training and fire drill training necessary for meeting the fire brigade qualifications.

The inspectors reviewed OMM-02, Fire Protection Manual, which establishes the responsibilities and methods for implementing the licensee's fire protection program. The fire brigade is required to be composed of at least five members at all times. The brigade may have less than the minimum required for a period of time not to exceed two hours in order to accommodate unexpected absence. The inspectors determined that the minimum fire brigade composition was not compromised as a result of this incident. The inspectors considered this incident to be an example where the Fire Protection Technical Aide did

not adequately verify the requirements of individuals temporarily assigned to the fire brigade.

Within the area examined, no violations or deviations were identified.

5.4 Close Out Items

(Closed) LER 93-022-00: Technical Specification Failure to Obtain Hydrogen/Oxygen Analysis

The inspectors reviewed the identified problem - failure to take a grab sample from the Waste Gas Decay Tank every 4 hours when degassing the Reactor Coolant System with the Gas Analyzer out of service (TS Table 3.5-7). Corrective actions identified in the LER and ACR 93-302 involved counseling the individuals involved in the event and revision to Environmental Monitoring Procedure EMP-024 (Revision 23 effective 12/11/93), Chemistry Procedure CP-003 (Revision 17 effective 12/25/93), and General Operating Procedure GP-007 (Revision 31 effective 2/24/94).

Licensee cause analysis for this event recognized that the cause involved several elements. In that the monitoring procedure EMP-024 specified the compensatory sampling, failure to follow procedures is identified (lack of attention to the details of the procedure). Also, the failure of Operations to communicate to Chemistry that the degassing evolution was in progress and sampling frequency should be increased (poor coordination between Operations and Chemistry) contributed to the event.

The inspectors verified that the procedure revisions were made and concluded that the changes should prevent recurrence.

6.0 REVIEW OF UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) COMMITMENTS

A recent discovery of a licensee operating their facility in a manner contrary to the 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 inspection discussed in this report, the inspectors reviewed selected portions of the UFSAR that related to the areas inspected. The inspectors verified that for the select portions of the UFSAR reviewed, the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

7.0 EXIT

The inspection scope and findings were summarized on March 6, 1996, by Paul Byron with those persons indicated by an asterisk in paragraph 1. Interim exits were conducted on February 2 and February 16. The inspectors described

the areas inspected and discussed in detail the inspection results. A listing of inspection findings is provided. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Type/Item Number	Status	Description and Reference
NCV 96-02-01	Closed	Failure to Follow LCTR Resulting in Deenergization of Boric Acid Transfer Pump Heat Tracing (paragraph 2.3).
LER 94-019-00	Closed	TS Violation Due to Exceeding Pressurizer Cooldown Rate (Paragraph 2.5).
VIO 95-13-01	Closed	Failure to Provide Adequate Work Instruction for Degraded Stud inspection (paragraph 3.4).
NCV 96-02-02	Closed	Inadequate Accumulator Level Transmitter Scaling Calculation (paragraph 4.1).
LER 94-014-00	Closed	Technical Specification Violation Due to Improper High Steam Flow Setpoint (paragraph 4.2).
LER 94-014-01	Closed	Technical Specification Violation Due to Improper High Steam Flow Setpoint (paragraph 4.2).
NCV 96-02-03	Closed	Inadequate ESF High Steam Flow Setpoint Scaling Calculation (paragraph 4.2).
URI 96-02-04	Open	Review Adequacy of Testing of New Hand Geometry Access Control System (paragraph 5.1.1).
URI 96-02-05	Open	Review Licensee Examination of Potential Safeguards Material Found Unsecured (paragraph 5.1.2).
URI 96-02-06	Open	Review Results of Licensee Investigation of Degraded Protected Area Lighting Equipment (paragraph 5.1.3).
LER 93-022-00	Closed	Technical Specification: Failure to Obtain Hydrogen/Oxygen Analysis (paragraph 5.4).

8.0 ACRONYMS

ACC - Air Condition Compressor
ARW - Advanced Rad Worker
ACR - Adverse Condition Report
AFW - Auxiliary Feedwater
ALARA - As Low As Reasonably Achievable
AMSAC - ATWS Mitigation Actuation Circuitry

AP	-	Administrative Procedure
ASME	-	American Society of Mechanical Engineers
BATS	-	Boric Acid Transfer System
CEDE	-	Committed Effective Dose Equivalent
CFR	-	Code of Federal Regulations
CM	-	Corrective Maintenance
CP&L	-	Carolina Power & Light Company
CR	-	Condition Report
DSDG	-	Dedicated Shutdown Diesel Generator
ED	-	Electronic Dosimeters
ESR	-	Engineering Service Request
E&RC	-	Environmental and Radiation Control
FSAR	-	Final Safety Analysis Report
ESF	-	Engineered Safety Feature
FP	-	Fire Protection
HRA	-	High Radiation Areas
I&C	-	Instrumentation & Control
ID	-	Identification
LCO	-	Limiting Condition for Operation
LCTR	-	Local Clearance and Test Request
LER	-	Licensee Event Report
LOCA	-	Loss of Coolant Accident
LHRA	-	Locked High Radiation Areas
LP	-	Loop Calibration
MI	-	Maintenance Instruction
MMM	-	Maintenance Management Manual
MST	-	Maintenance Surveillance Test
NAS	-	Nuclear Assessment Section
NVLAP	-	National Voluntary Laboratory Accreditation Program
NCV	-	Non-Cited Violation
OAO	-	Outside Auxiliary Operator
OMM	-	Operations Management Manual
OST	-	Operations Surveillance Test
OP	-	Operating Procedure
OST	-	Operations Surveillance Test
PCE	-	Personnel Contamination Events
PIC	-	Process Instrument Calibration
PLP	-	Plant Program
PM	-	Preventive Maintenance
RWP	-	Radiation Work Permits
RCA	-	Radiologically Controlled Area
RC	-	Radiation Control
RNP	-	Robinson Nuclear Plant
SDAFW	-	Steam Driven Auxiliary Feedwater