U. S. NUCLEAR REGULATORY COMMISSION

REGION II

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Licensee:	Carolina Power & Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	April 13 - May 24, 1997
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Enclosure

EXECUTIVE SUMMARY

Brunswick Steam Electric Plant, Units 1 & 2
NRC Inspection Report 50-325/97-07, 50-324/97-07

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of a corrective action inspection by a regional inspector and a fire protection program inspection by a regional inspector.

Operations

- The licensee did not exceed a limiting condition for operation (LCO) concerning primary containment integrity after a leak developed on a core spray system drain line. (Section O1.1) A more conservative decision to enter the LCO early could have been made based on existing procedural guidance.

- As unresolved item was identified for failure to enter an LCO per plant procedure during testing of the chlorine detectors. (Section O3.1)

- The Plant Nuclear Safety Committee (PNSC) meetings were conducted in accordance with Technical Specification (TS) requirements. (Section O7.1) An increased focus on the generic implications of NRC violations and recommendations to prevent recurrence was observed.

- A weakness in training and procedural guidance for a routine plant evolution of removing a reactor feed pump (RFP) from service was noted. (Section O8.1) During removal of an RFP from service a plant transient occurred causing a recirculation pump runback.

Maintenance

- The inspector observed good radiation work practices and communication during the performance of an instrument channel calibration. (Section M1.1)

- The inspector identified a difference in warning sign labeling between units concerning bumping or jarring an instrument that could cause a single failure scram. (Section M2.1) The licensee initiated corrective action to resolve the difference.

- The inspector concluded that the TS scram time testing data for the current unit's cycle of operation did not indicate any degradation of scram times. (Section M3.1) The licensee continued to monitor industry problems for applicability.

- The control of chemicals continues to be a challenge. (Section M7.1)

- The work activities associated with a reactor core isolation cooling system were properly planned and scheduled. (Section M7.2)

<u>Engineering</u>

- An NCV was identified for failure to properly identify plant design basis information and assure that it was correctly translated into calculations associated with the power uprate submittal. (Section E1.1)

- The licensee's evaluation of deficiencies identified during Environmental Qualification (EQ) walkdowns meets NRC requirements. (Section E1.2)

- The inspectors concluded that the corrective action program was generally good, however timely implementation could be improved. . Some condition reports (CRs) were not completed within the time constraints established in the procedure. Extensions were granted without a clear statement of why the deferral was needed. Items such as operating procedure changes were backlogged over a year. This was reflective of a potential weakness in the prompt correction of identified problems. CRs generated indicated that the threshold was sufficiently low to ensure identification of all meaningful problems. Furthermore, senior management training on corrective action programs (CAPs) and CRs was behind the scheduled NRC commitment. (Section E7.1)

- The inspectors concluded that the Self-Assessment Program description and implementation by the licensee was generally good. The licensee used CP&L employees from other sites, contractor personnel, and information from industry sources outside CP&L for an independent perspective and comparison of self-assessment programs and processes used for corrective actions and review of potential issues. The inspectors noted that self-assessments often provided detailed identification of numerous problems and appropriate recommendations for corrective actions. However, personnel interviewed expressed that program effectiveness could be enhanced by consolidation and standardization of the numerous assessments. (Section E7.2).

- The Operating Experience (OE) Program was determined to be effective. The completed OE evaluations reviewed were acceptable. The OE staff assessments and performance evaluation section (PES) audits of OE were thorough. The weekly status meeting, status reports and OE tracking provided good program oversight. The incorporation of OE data into routine and non-routine activities was viewed as a strength. (Section E7.3)

<u>Plant Support</u>

- The corrective actions taken by the licen[illegible]ce the num[illegible] personnel contamination events (PCEs) has [illegible]ective [illegible] R1.1) No PCEs occurred during a two month [illegible]

- The plant security guards successfully identif[illegible]ing a weapon into the plant during the performance of [illegible]Section S1.1) All personnel access portal equipment was properly tested. (Section S1.2)

- The licensee has been actively pursuing the resolution of the Thermo-Lag issue. (Section F1.1)

- Good compliance with plant fire prevention procedures has resulted in a low incident of fire within the plant protected area. (Section F1.2)

- There was not a backlog of open fire protection work requests. Corrective maintenance on degraded fire protection systems was being accomplished in a timely manner. The maintenance and material condition of the fire protection equipment was satisfactory. (Section F2.1)

- The fire protection impairment status report provided the licensee with a good means of identifying out of service fire protection equipment to assure that appropriate compensatory measures were implemented. (Section F2.1)

- An inconsistency in the implementation of compensatory measures for the High Pressure CO_2 Systems was identified and corrected. (Section F2.2)

- An unresolved item was identified concerning compensatory measures when Alternate Safe Shutdown equipment was out of service. (Section F2.3)

- The fire protection program implementing procedures were good and met licensee and NRC requirements. Implementation of the fire protection and prevention procedures and the general housekeeping for control of combustibles within the plant were satisfactory, except, the inspector observed reduced efforts in containing lubrication oil leaks in the diesel generator rooms. (Section F3)

- The fire brigade organization and training met the requirements of the site procedures and the performance by the fire brigade to a drill during this inspection was good. (Section F5)

- The coordination and oversight of the facility's fire protection program met the licensee's procedures and commitments in the Updated Final Safety Analysis Report (UFSAR). The personnel assigned various fire protection related functions within the Loss Prevention Unit were working together as a team and with coordination by the superintendent to implement the fire protection program at the site. (Section F6)

- The Nuclear Assessment Section assessments and Loss Prevention Unit self-assessments of the facility's fire protection program was comprehensive and was effective in identifying fire protection program performance deficiencies to management. Planned corrective actions in response to the audit issues were acceptable. (Section F7)

Report Details

Summary of Plant Status

On April 16, 1997, Unit 1 raised power to the new uprated 100% power level. This reflected a change to 2558 megawatts thermal and 844 megawatts electric. However, on April 20, 1997, indications that a fuel assembly leak had occurred were evident because of increases in iodine dose equivalent samples. On April 22, 1997, iodine levels had increased to a value of 1.27 E-2 μCi/ml up from previous levels of E-4 and E-5. The licensee reduced power level to 60 percent power and the values decreased. Power suppression testing began on April 22, 1997, and continued for four days, to determine the location of the leaking fuel. Control rods were inserted to suppress the flux around the identified fuel leaker and power was returned to 100%. At the end of the inspection period, the unit had been on-line 198 days.

Unit 2 operated continuously during this period. On May 7, 1997, the licensee reduced power to 50 percent due to a steam leak on the east second stage reheat drain tank manway. The licensee repaired the leak and returned to 100 percent power on May 11, 1997. At the end of the inspection period, the unit had been on-line 253 days.

The mechanical vacuum pumps remained tagged out on both units due to concerns about control room dose in the event of a Rod Drop Accident. The licensee, in a letter to the NRC dated February 13, 1997, committed to upgrade the mechanical vacuum pump trip function to implement a vacuum pump trip from the main steam line radiation monitor prior to the next startup.

Due to an identified discrepancy between TS required suppression chamber water level and water volume, the licensee has issued Standing Instruction, SI 97-031, to maintain a more conservative water level band until a TS amendment is approved. The SI directs the operations to maintain level between -27.5 inches and -29.5 inches compared to TS valves of -27 inches to -31 inches. The inspectors have observed compliance with this SI during routine tours of the control room.

Due to concerns about the control room dose the licensee imposed an administrative limit on Iodine until a TS amendment is approved. The licensee made a procedure change to Administrative Procedure 0AI-81, Water Chemistry Guidelines, setting the limit at 0.1 microcurie per gram dose equivalent Iodine 131 compare to a TS valve of 0.2 microcurie per gram. Also, the licensee has been providing weekly data to NRR and the resident inspector for review. None of the data reviewed has exceeded the administrative limit.

Eleven of twelve Justification for Continued Operation (JCO) in the Environmental Qualification (EQ) of equipment area remain open for both units. The following provides the status of the EQ JCOs and associated Engineering Service Requests (ESRs):

1) ESR 96-00425, Evaluation of EQ sealants was considered closed by the licensee.
2) ESR 96-00503, Associated Circuit EQ was scheduled for completion May 31, 1997.
3) ESR 96-00426, Evaluation Quality class and EQ classification of PASS valves was scheduled for completion June 6, 1997.
4) ESR 96-00501, Motor Control Center (MCC) EQ was scheduled for completion June 6, 1997.
5) ESR 96-00625, EQ Type JCO for EQ Fuses Without a Qualification Data Package (QDP) was scheduled for completion June 6, 1997.
6) ESR 96-00627, QDP for Marthon 300 Terminal Blocks was scheduled for completion December 31, 1997.
7) ESR 97-00087, EQ-Type JCO for Improperly Configured Conduit Seal was scheduled to be completed June 30, 1997.
8) ESR 97-00229, JCO for GE CR 151 B Terminal Blocks was scheduled to be completed July 15, 1997.
9) ESR 97-00238, JCO for Standby Gas Treatment (SGT) Motor Operated Valve (MOV) Position Indicator Rheostat, scheduled for completion in December 31, 1997.
10) ESR 97-00250, Conduit Union in EQ Boundary, was scheduled for completion December 31, 1997
11) ESR 97-00256, Main Steam Insulation Valve (MSIV) Hiller Actuator JCO, was scheduled for completion after walkdowns of the MSIV actuator on both units.
12) ESR 97-00289, Pass Valve Limit Switch Panel Wiring, schedule for completion was being developed.

In addition, a JCO remained in effect providing guidance and allowed out-of-service time for the three control building air-conditioning units. During a Safety System Functional Inspection conducted in May-June 1996, it was identified that the units were incorrectly downgraded from safety related or Q-list to non-safety related, ESR 96-00366, Evaluation of Using Existing Control Room Air Conditioners, provided a JCO evaluation until the issue was resolved. The issue remains open and the licensee committed in their February 15, 1997, letter to resolve all open issues by the completion of the Unit 1 refueling outage 12, scheduled to begin in the second quarter of 1998.

In summary, both units operated continuously during this report period. However, there were eleven outstanding JCOs in the EQ area and one JCO for the non-Q control building air-conditioning units. The mechanical vacuum pump remained tagged out due to concerns related to Rod Drop Accident analysis.

I. Operations

01 Conduct of Operations

01.1 Core Spray Drain Line Leak

a. Inspection Scope (71707)

The inspector reviewed the circumstances concerning a leak that developed on a Unit 2 core spray (CS) drain line on April 28, 1997. This problem resulted in entry of the unit into an LCO that required the unit be in hot shutdown within 12 hours.

b. Findings and Observations

At 6:00 a.m. on April 28, 1997, Unit 2 entered a 12 hour to hot shutdown LCO based on primary containment being inoperable. A leak occurred on a weld in a 3/4 inch drain on the containment side of the CS to torus full flow test valve. TS 3.6.1.1 required that primary containment integrity be restored within two hours or be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours. The licensee made plans to begin a controlled shutdown at noon. The weld repair was completed around noon and shutdown was not required.

The inspector reviewed the operator logs and noted that entries were made at 11:55 p.m. on April 27, 1997, concerning a leak from the weld area. The inspector questioned licensee management as to why the LCO entry was not made earlier. The LCO entry was not made until after engineers were called out to the site and examined the weld. The two-hour LCO was entered at 4:20 a.m. on April 28, 1997. At 6:00 a.m. it was concluded that repairs could not be made within two hours and the 12 hours to hot shutdown LCO was initiated.

The inspector discussed with licensee management that all the information necessary for operations to enter the LCO based on leakage from a weld was available at 11:55 p.m. on April 27, 1997. The licensee contended that, due to the location of the drain line above the CS ventilation cooler ductwork, the leak was hard to pinpoint. Also, there was an outstanding work request/job order (WR/JO) for a packing leak on the drain valve which might have been the source of the leakage.

The inspector verified that WR/JO 96-AGILI concerning a packing leak on drain valve, 2-E21-V33, was outstanding. Also, the inspector looked at the physical location of the drain line in the overhead above the ductwork making it somewhat difficult to see.

However, during the licensee's review of the inspector's questions, it was noted that the plant Conduct of Operations procedure, 00I-01.08, required in Section 8.C that, upon discovery of leakage from a Class 1, 2, or 3 component pressure boundary (i.e. pipe wall, valve body, pump casing, etc.), the component SHALL be declared inoperable. The guidance provided to plant operators was clear as to what action to take upon discovery of leakage from a weld.

The inspector further discussed with licensee management that a conservative decision would be to declare the component inoperable and then the decision could be reversed, if necessary, after the engineers arrived. There was still ample time to review the decision and avoid an unnecessary plant transient.

c. Conclusions

The inspector concluded that the licensee did not exceed an LCO time limit because the leak was repaired. A more conservative decision to enter the LCO early could have been made based on existing procedure guidance.

O2 Operational Status of Facilities and Equipment

O2.1 Special UFSAR Review

A recent discovery of a licensee operating the 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 inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and/or parameters.

During a routine inspection of the Unit 1 reactor building on May 1, 1997, the inspector observed that the linkage for some ventilation dampers in the reactor building exhaust were not connected. The inspector reviewed a previous modification that had been made to this configuration to see that the present operating configuration was correctly reflected in the UFSAR.

The inspector reviewed plant drawing F-40073, Unit 1-Reactor Building Ventilation System Air Flow Diagram. At the damper location was referenced Note 2 which stated that the exhaust fan inlet dampers were bolted full open per ESR 9500213, Relocating the Vortex Damper Control. The scope of work performed under this ESR was to relocate the vortex damper control for the reactor building negative pressure control from the exhaust fans to the supply fans.

The inspector reviewed UFSAR Section 9.4.2, Reactor Building Ventilation System. Stated in this section was that the exhaust fan inlet dampers were bolted full open. This statement has a sidebar marking it as a recent UFSAR change.

The inspector concluded this 1995 ESR had been properly updated into the UFSAR. The operating configuration was as stated in UFSAR.

O3 Operations Procedures and Documentation

O3.1 Chlorine Detector Monthly Calibration Check

a. Inspection Scope (71707)

The inspector reviewed a detector failure during the performance of the chlorine detection system test plan. This plan was instituted as a result of multiple chlorine detector failures spanning over the last seven years.

b. Observations and Findings

As corrective action for multiple failures of the chlorine detectors located at the Control Building Intake Plenum and the Service Water Building, the licensee committed to several actions. Among those actions was a modification of the chlorine detector test plan to include monthly sensor calibration checks in addition to the TS required monthly Maintenance Surveillance Test OMST-CLDET11M, Chlorine Detection System Channel Functional Test. The monthly sensor calibration checks were added to the test plan in the form of Special Procedure OSP-97-006, Chlorine Detection System Test.

During a routine review of licensee CRs, the inspector noted CR 97-1475, Failed Cl Detector From Stock, which described a chlorine detector failure during post-maintenance TS operability testing in accordance with the test plan special procedure. The inspector reviewed the CR, work package, and associated procedure. The inspector reviewed OSP-97-006 and noted no substantial difference between the special procedure and OMST-CLDET21A, Chlorine Detection System Channel Calibration. Contained in Section 4.3, of OSP-97-006, was a note stating that the Control Building Heating, Ventilation and Air Conditioning would isolate as a result of Maintenance Surveillance Test OMST-CLDET11M, Chlorine Detection System Channel Functional Test. These isolations rendered the control room radiation and smoke protection functions inoperable due to these signals being blocked when an actual chlorine signal was initiated.

The inspector reviewed the applicable TSs, the completed procedures for the April performance of OSP-97-006 and OMST-CLDET11M and the Unit 1 and 2 operator logs. On April 24, 1997, OMST-CLDET11M was performed on both units. The inspector determined that the tank car was in the exclusion area and the surveillance test initiated actual chlorine detection signals which rendered the smoke and radiation protection modes of the Control Room Emergency Ventilation System (CREVS) inoperable. However, there was no entry in the operator logs into an LCO. With those modes inoperable, TS 3.7.2 Control Room Emergency Ventilation System required entrance into the ACTION statement. The ACTION statement allows 7 days to restore the control room emergency filtration unit or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours. The inspector determined that no entry was made into TS 3.7.2, however, the control room emergency filtration system was restored within the time allowed in the ACTION statement.

The inspector discussed these observations with the licensee. Subsequently the licensee issued CR 97-1787, Missed LCO (OMST-CLDET11M) addressing the deficiency. TS 6.8.1.a requires that procedures shall be implemented for log entries and equipment control. Operating Instruction 0OI-01.08, Control of Equipment and System Status requires that an entry describing the condition be entered into the LCO Tracking System. This unresolved item is identified as URI 50-325(324)/97-07-01, Failure to Enter TS ACTION Statement. This item will be further reviewed for other examples.

The inspector noted that additional failures of chlorine detectors occurred during the May 19, performance of OSP-97-006 as recorded in CRs 97-1766, Chlorine Detector Failure, and 97-1776, Chlorine detector, new, failed. All applicable TSs were entered appropriately and the associated ACTIONS were not exceeded.

c. Conclusions

During operability testing of the chlorine detectors following detector calibration, the licensee failed to enter the applicable TS ACTION statement. However, the licensee did not exceed the associated TS ACTION statement. A URI was identified for failure to enter the LCO per the plant operating procedure.

O7 Quality Assurance in Operations

O7.1 Plant Nuclear Safety Committee

a. Inspection Scope (40500)

The inspector attended several PNSC meetings during the inspection period to verify compliance with TS 6.5.3 requirements.

b. Observations and Findings

The inspector attended PNSC meetings on several occasion to verify compliance with the TS 6.5.3 requirements. The inspector noted that although the PNSC meetings did not always begin on time, the committee was appropriately staffed with the requisite number of members or alternates as required in TS 6.5.3.5.

The inspector verified that the PNSC reviewed proposed changes to the Fire Protection program as required by TS 6.5.3.8. The inspector noted that reports of violations in NRC inspection reports were reviewed in accordance with TS 6.5.3.8.e. In addition, the inspector noted an increased focus compared to other meetings on the potential generic implications of the violations and the adequacy of the recommendations to prevent recurrence.

c. Conclusions

The inspector verified PNSC compliance with TS 6.5.3 requirements. An increased focus on the generic implications of NRC violations and recommendations to prevent recurrence was observed.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) URI 50-325(324)/97-02-02: Recirculation Pump Transients

In NRC Inspection Report (IR) 97-05, the inspector reviewed the status of this item. Additional questions remained open regarding the actions taken concerning the recirculation pump runbacks. Three runbacks occurred on March 1, 1997. The first runback occurred following a planned evolution to remove an RFP from service during a downpower maneuver. Due to a flow mismatch between the RFPs, a transient occurred resulting in the runback. The licensee attributed the event to multiple root causes. The inspector reviewed the licensee's root cause completed March 19, 1997. The most significant item for this routine plant evolution was a weakness in training and procedure guidance for this maneuver. In response to the inspector's question concerning initial plant conditions that would prevent a reoccurrence of this situation, the licensee revised the plant operating procedure 1OP-32 and 2OP-32, Condensate and Feedwater System Operating Procedure. The procedure was changed to provide notes on RFP recirculation valve operation with an effective date of May 1, 1997.

The second runback occurred when power was lowered too close to the setpoint for a runback. This event was not that significant because plant conditions were near the point where a normal runback signal would occur. This action may have been prevented by allowing more margin or better indication.

The third event occurred primarily due to the RFP 2B minimum flow valve air line failure that caused the valve to fail open resulting in a transient. Although not conclusive why the air line failed, this might have occurred or started during the first runback event.

The inspector reviewed the licensee's closure package for these items, procedure revisions, and personnel statements. The inspector concluded that the first event was due to a weakness in training and procedure guidance for a routine plant evolution of removing an RFP from service. In addition, completion of the recirculation motor-generator on-line brush replacement modification will eliminate this operational challenge of having to conduct single loop operations for brush replacement. This modification had been completed on Unit 1. Unit 2 will be completed in the fall refueling outage of 1997. This item was closed.

O8.2 (CLOSED) LER 50-325/95-011-00: Manual Reactor Protection System Trip Due to Decreasing Condenser Vacuum.

A manual reactor trip was initiated due to decreasing condenser vacuum. The loss of vacuum was due to aquatic plant life clogging the circulating water intake screens.

Corrective actions included establishment of a task force to investigate and implement methods to prevent or mitigate biological impingement on the intake and traveling screens. An action plan report was issued on August 1995, this item is closed.

O8.3 (CLOSED) VIO (B) 50-325(324)/96-013-01: Equipment Clearance Error.

The license's response to this violation, dated November 11, 1996, was accepted by the NRC on November 27, 1996. The corrective actions described in the response letter were verified as completed by the inspector. This item is closed.

O8.4 (CLOSED) VIO (B) 50-325(324)/96-013-02: Failure to Complete the LSRO Training Program Prior to Taking the LSRO Audit Examination.

Corrective actions described in the license's response, dated November 11, 1996 were verified as completed by the inspector. The response was accepted by the NRC on November 27, 1996. This item is closed.

O8.5 (CLOSED) IFI 50-325(324)/96-013-04: Licensed Operator Requalification (LOR) Makeup for Licensed Operators.

The licensee has taken action to correct this deficiency. Out of 30 reactor operators (ROs) at the site, 24 have completed the AO Delta training and have completed their qualifications under the program. The remaining 6 ROs are currently working on completing their qualification card. To prevent recurrence the licensee has established administrative controls to track the status of qualifications for standing watch. This item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Automatic Depressurization System Pressure Permissive Channel Calibration

a. Inspection Scope (61726)

The inspector observed the performance of the Unit 2 Residual Heat Removal (RHR) discharge pressure Automatic Depressurization System (ADS) permissive instrument channel calibration.

b. Observations and Findings

On May 1, 1997, the inspector observed the performance of Maintenance Surveillance Test 2MST-RHR25Q, RHR Pump Discharge Pressure ADS Permissive Instrument Channel Calibration. The procedure was verified to ensure that the testing was done within the TS prescribed frequency. Test equipment used was verified to be within its current calibration cycle. Selected test data was independently verified for accuracy. The inspector observed good communication and procedural usage between work groups. The inspector noted good worker practices concerning work on a contaminated system and maintaining dose received As Low As Reasonably Achievable (ALARA). The test was completed satisfactorily with no deficiencies noted.

c. Conclusions

The inspector observed good radiation work practices and communication during the performance of the ADS pressure permissive channel calibration. The test was completed satisfactorily with no identified deficiencies.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Labeling of Instruments

a. Inspection Scope (62707)

The inspector reviewed instrumentation labeling in the turbine buildings on May 12, 1997.

b. Observations and Findings

The inspector observed a difference in instrument labeling between the Unit 1 and Unit 2 turbine buildings. The inspector observed that the Unit 1 pressure transmitters for low condenser vacuum, 1-B21-PT-N056-A-D, had a precaution label or sign but the corresponding instruments on Unit 2 did not. The sign was a warning sign that a single failure scram was possible and the instrument should not be bumped or jarred. The inspector questioned the licensee as to why the difference in labeling.

The licensee initiated CR 97-01753 to add a label to the Unit 2 instruments to determine if any discrepancies exist.

c. Conclusions

The inspector identified a labeling difference in the plant. The licensee initiated corrective action to address the issue.

M3 Maintenance Procedures and Documentation

M3.1 Scram Time Testing

a. Inspection Scope (61726)

The inspector reviewed the results of the current operating cycle scram time testing results for both units. This was reviewed due to the shutdown of another facility due to slow scram times.

b. Observations and Findings

The licensee had experienced slow control rod scram times in the past as reported in Licensee Event Report (LER) 1-96-002. Problems were found exceeding the first 5% insertion time allowed by TS of 0.358 seconds. In response to these problems the licensee changed the Scram Solenoid Pilot Valve exhaust diaphragms from Viton to Buna-N on both units. The inspector reviewed the scram time testing results for both units for the current operating cycles. This testing was performed using procedure OPT-14.2.1, Single Rod Scram Insertion Times Test.

The following data was reviewed:

UNIT	DATE	AVERAGE TIME
1	11/09/96	0.308
	3/15/97	0.303
2	3/16/96	0.298
	7/26/96	0.307
	9/05/96	0.302
	12/08/96	0.295
	4/12/97	0.302

The test data did not indicate any degradation of scram times. The problem that occurred at another facility was not occurring at Brunswick. The licensee was fully aware of this problem and continued to monitor the issue.

c. Conclusions

The inspector concluded that the TS required scram time testing did not indicate any degradation of scram times. The licensee continued to monitor industry problems for applicability.

M7 Quality Assurance in Maintenance Activities

M7.1 Chemical Labeling Practices

a. Inspection Scope (62707)

The inspector observed chemical labeling practices during routine surveillance of maintenance activities.

b. Observations and Findings

On April 30, 1997, the inspector observed the replacement of defective heater elements in the Unit 2 "B" instrument air dryer tower. During observation of the work area the inspector noted four large bags. Upon questioning the workers present, it was determined that the bags contained the desiccant removed from the instrument air dryer. The inspector determined that the chemicals should have been labeled in accordance with Administrative Instructions 0AI-121, Chemical/Consumable User Program. Additional questioning by the inspector revealed that the desiccant was to be reused upon replacement of the defective heater elements, however the workers present were not familiar with the hazards or restriction on its use.

On May 12, 1997, the inspector reviewed the work activities associated with the Diesel Generator (DG) cylinder replacement. Inspector review of the work areas revealed four barrels of Q-list DG lubricating oil. Attached to the barrels were pieces of tape with the statement "clean used oil", the date, and a set of initials. The laydown area posting indicated that this area was a designated Q-list storage area. Therefore the workers were required by Administrative Instruction 0AI-121 to affix a chemical control label to the lube oil containers.

These two incidents indicate that chemical control continues to be a challenge for the site. The need to handle and temporarily store the chemicals was known in advance. In addition, the 0AI-121 requirements were covered during the pre-job briefing. The inspector determined that the establishment of chemical control requirements should have been identified initially by the planner. However, additional opportunities to identify the need for labels were present because the topic was covered in the pre-job briefing. The work crew should have recognized that labels were required for the temporary storage of the lube oil.

c. Conclusions

The control of chemicals continues to be a challenge as evidenced by lack of labeling four barrels of Q-list lubricating oil and instrument air dryer desiccant.

M7.2 Scheduling and Planning

a. Inspection Scope (62707)

The inspector reviewed the licensee's process for scheduling and planning system outages as outlined in Plant Program Procedure OPLP-24, Work Management.

b. Observations and Findings

The inspector reviewed those activities associated with the planning and scheduling for the Unit 2 Reactor Core Isolation Cooling (RCIC) System. The inspector reviewed OPLP-24. This procedure described the process for identification, prioritization, planning, preparation, scheduling, approval, implementation, and documentation of maintenance activities. Through routine field observations the inspector noted that routine scheduling activities were adequately conducted. An integrated schedule was released outlining the weekly schedule. At the end of each day a Plan of the Day (POD) was issued statusing the progress of the work activities. POD items were reviewed daily by operations staff. A morning meeting was held daily to address the status of the more significant items. Daily work ticket reviews were conducted to discuss any scheduling or implementation concerns.

To aid in controlling worker dose, the licensee has integrated planners with radiation protection experience working in the planning and scheduling organization. These planners perform a review of the unit conditions, other scheduled work, the specific job schedule, frequency of job performance, worker familiarity with the job, and the methods used to perform the work. This review in combination with the job ALARA plan was integrated into a job dose projection. A challenge meeting was conducted with the representatives from each of the affected site organizations to discuss their readiness to implement the system outage as planned. After implementation, a project critique was performed to discuss lessons learned and possible work practice improvements.

The inspector reviewed those activities associated with the planning and scheduling for the Unit 2 RCIC System. The inspector discussed the process with the licensee, reviewed the proposed schedule, observed associated maintenance activities and attended the RCIC challenge meeting. Adequate defense-in-depth was maintained. During the RCIC outage, a soft patch on the RCIC turbine supply drain pot began leaking and complicated system outage activities. Despite this change in plant conditions, actual radiation exposure for the system outage was below the projection.

c. Conclusions

The inspector reviewed those activities associated with the planning and scheduling for the Unit 2 RCIC System. The work was performed satisfactorily with no identified problems or concerns.

M8 Miscellaneous Maintenance Items (92902)

M8.1 (CLOSED) LER 50-325/95-005-00: Unplanned ESF Actuation Due to Reactor Building (RB) Ventilation Exhaust Radiation Monitor Power Supply Failure.

The power supply for channel B RB ventilation exhaust radiation monitor failed resulting in a partial isolation of the containment atmospheric control system, a secondary containment isolation and an initiation of the standby gas treatment system. The event was caused by the misorientation of a ground terminal in the power supply. Corrective actions as stated in the LER were completed and were verified by the inspector, this item is closed.

M8.2 (CLOSED) LER 50-325/95-014-00, and 50-325/95-014-01: Unplanned ESF Action Due to Low Voltage Spike on B Reactor Protection System.

A momentary voltage spike in the RPS B bus voltage, resulted in a half scram. An investigation revealed the low voltage spike occurred due to an intermittent failure in the RPS M-G set output voltage regulator. The voltage regulator in the 2A and 1B MG set were replaced as a result of this event and the other two MG sets were replaced in November 1995.

Supplement 1 of this LER was issued following completion of the failure analysis of the components. The results were indeterminate, but all four MG sets voltage regulators were replaced, this item is closed.

M8.3 (CLOSED) LER 50-325/95-018-00: Automatic Reactor Shutdown Due to Condensate/Feedwater Transient.

Following removal of a conductivity cell on the condensate pump suction piping air in-leakage occurred. This resulted in vapor binding of the operating condensate pump and subsequent loss of pressure and flow causing an automatic reactor shutdown.

An investigation determined that uneven packing around the shaft of the conductivity cell resulted in excessive air in-leakage. Corrective actions as listed in the LER were completed and were verified by the inspector, this item is closed.

M8.4 (CLOSED) LER 50-325/96-006-00: Technical Specification Surveillance Acceptance Criteria Did Not Adequately Account for Head Losses.

The acceptance criteria for the RCIC reactor core isolation cooling surveillance test did not properly account for the difference in the test flow path versus the normal reactor vessel injection flow path. Data from recent tests for both units 1 and 2 revealed that pressure and flows were sufficiently above the acceptance criteria to account for the head losses, which demonstrated no operability problems.

The RCIC acceptance criteria has been revised. The acceptance criteria for the high pressure coolant injection surveillance test was also adjusted. This item is closed.

M8.5 (CLOSED) LER 50-325/96-010-00: Emergency Diesel Generator #1 DC Control Power Breaker Cable Termination.

The licensee discovered that the load side conductor installed on the EDG #1 normal DC control power were improperly secured. Proper installation requires the conductor terminating lugs to be secured between the threaded breaker stab and lug screw threads. In this case the load side lugs were resting between the breaker case and breaker stab.

An evaluation of the improperly installed cable could result in electrical discontinuity during a seismic event. The cable was correctly terminated and other normal and alternate control power supply breakers for the other EDGs were inspected. All terminations were found correctly secured, this item is closed.

M8.6 (CLOSED) LER 50-325/96-007-00: LER 50-325/96-007-01 and IFI 50-325(324)/96-008-02, Double Disc Gate Valve Leakage.

The packing in HPCI valve 1-E41-F001 was damaged due to galling during a valve stroke. Investigation determined the leakage was caused by an off-centered valve stem. New packing was installed and the stem was adjusted to eliminate contact with the gland follower.

The licensee issued LER 96-007-00 to report the event, and LER 96-007-01 (supplement 1) was issued to describe the investigative findings.

IFI 96-008-02 was issued to track the investigative findings. Corrective actions have been completed. This item is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Power Uprate Deficiencies

a. Inspection Scope (37700, 92903, 92720)

Following licensee identification of deficiencies in two calculations supporting the power uprate program, an NRC review was performed of licensee actions to determine root causes, look for additional problems, and correct identified deficiencies.

b. Observations and Findings

Background:

On November 1, 1996, Amendment Nos. 183 and 214 for Units 1 and 2 respectively, were issued authorizing an increase in each unit's maximum power level from 2436 megawatts thermal (MWt) to 2558 MWt. The licensee planned to implement the amendment in early November 1996 on Unit 1 during that unit's startup from the then current refueling outage B111R1. The amendment was to be implemented on Unit 2 no later than the startup from its next refueling outage, which is currently scheduled for the fall of 1997.

Prior to reaching uprated power levels the licensee identified several deficiencies. First, the licensee determined that a non-conservative input assumption had been made in the Station Blackout (SBO) event analysis that pre-existed power uprate and that the error had been carried forward into the more recent analysis of the impact of power uprate on the SBO event. Both analyses assumed an initial suppression pool water temperature of 90°F whereas TS allow a higher suppression pool temperature of 95°F. Second, the licensee determined that an acceptance criterion for maximum suppression pool temperature of 220°F had been used in the SBO and Loss of Coolant Accident (LOCA) power uprate analyses; whereas TS 5.2.2.b indicates that the suppression chamber design temperature is 200°F. Third, the licensee's contractor performing the bulk of the analyses supporting power uprate, was not sufficiently familiar with the licensing basis of the facility and selected an inappropriate peak suppression pool temperature as a starting point in determining the impact of power uprate on the BSEP SBO event.

At the time of discovery of the deficiencies it appeared that under uprated power conditions suppression pool temperature could exceed the 200°F suppression chamber design limit indicated by TS during both an SBO event and a LOCA. Additional licensee review has shown, however, that the 200°F suppression chamber temperature criterion will not be exceeded for SBO or LOCA at the uprated maximum power level.

On November 5, 1996, immediately after notifying the NRC of the deficiencies and before reaching the uprated power range on Unit 1 startup, the licensee established a 95 percent hold on reactor power (equivalent to the former value for licensed maximum power) and advised the NRC of this decision. During a telephone conference with the NRC on November 14, 1996, the licensee committed to (1) carry out an action plan calling for a review for root causes and additional problems and (2) continue the 95 percent power restriction until the NRC was satisfied that the root causes and implications of the issue were understood and appropriate corrective actions taken. The licensee reiterated these commitments in a letter to the NRC dated December 3, 1996.

Further licensee review under the action plan identified a non-conservative assumption in a licensing basis calculation of control room operator dose following a control rod drop accident (CRDA) initiated at power levels of less than 5 percent. The assumption was that control room dose would cease at 10 minutes due to manual operator action to isolate the CREVS, whereas such dose would continue due to recirculation of radioactive isotopes that entered the control room envelope before CREVS isolation.

The licensee submitted the results of its power uprate review on December 23, 1996.

Less significant deficiencies identified through the action plan included an error in the calculation of suppression chamber volume, reference to the wrong ASME Code addenda applicable to control rod drive mechanism design, an error in the reported flowrate for the standby liquid control system, an error in the estimated reactor water cleanup heat-exchanger room peak pressure, and a failure by the licensee to recognize in its power uprate submittal that the main steam line break accident (MSLBA) was more limiting with respect to control room dose than the LOCA.

In letters dated January 22, 1997, February 15, 1997, and February 28, 1997, the licensee provided commitments for resolution of outstanding deficiencies. These included commitments to improve the integrity of the control room pressure envelope and thereby reduce control room dose during the MSLBA and CRDA.

Root Causes:

The licensee found two primary causes for the identified discrepancies. They were (1) a failure by the licensee to clearly define inputs for calculations and analytical and licensing acceptance criteria, which exacerbated a lack of vendor personnel familiarity with the SBO design basis, and (2) an error by the vendor in selecting an incorrect peak suppression pool temperature as a starting point in determining the impact of power uprate on the SBO event. the licensee's failure to clearly define calculation inputs and acceptance criteria early in the design process was the principal contributor to the delay in identification of the problems.

The licensee missed an opportunity to resolve some of the power uprate submittal deficiencies by failing to address findings identified by a consultant (ALTRAN) in September 1995.

NRC Review:

The NRC reviewed licensee activities and conclusions throughout the conduct of the action plan.

The licensee's failure to properly identify plant design basis information and assure that it was correctly translated into calculations associated with the power uprate design change was identified as a violation of the requirements of 10 CFR 50, Appendix B, Criterion III, "Design Control". This licensee identified and corrected violation was treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy. This was identified as NCV 50-325(324)/97-07-02, Failure to Properly Translate Design Basis Information Into the Power Uprate Design Change.

Given the licensee's corrective actions, the NRC determined that the conclusions reached in the NRC's November 1, 1996, Safety Evaluation regarding the power uprate remain valid, and, on March 18, 1997, released the licensee from the 95 percent hold on reactor power.

c. Conclusions

An NCV was identified for failure to properly identify plant design basis information and assure that it was correctly translated into calculations associated with the power uprate design change.

E1.2 Environmental Qualification

a. Inspection Scope (37550, 37551)

The inspectors reviewed the results of field inspections to determine Environmental Qualification (EQ) of Barton pressure switches.

b. Observations and Findings

The inspectors reviewed the results of the EQ walkdowns performed to determine the as installed configuration of the Barton pressure switches, numbers CAC-PDS-4222 and -4223. These walkdowns were performed as part of the corrective actions to address EQ program deficiencies. Review of the results of the EQ walkdown documentation showed that while the Barton pressure switches are EQ qualified, the type of wire splices used to connect the leads from the switches to plant wiring and the lack of seals in the flexible conduit did not meet the requirements for installation in an environment with a high humidity and moisture.

The purpose of the switches is to control position of the reactor building to suppression pool vacuum breaker isolation valves. Section 15 of the UFSAR, Figure 15.0A.6-22, Protection Sequences for Pipe Breaks Inside Primary Containment, and the discussion for Accident Event 34, specifies that these valves are required to be operable during a loss of coolant accident (LOCA). For pipe breaks which occur outside of primary containment, such as a high energy line break (HELB), the valves are not required to be operable. This is documented in UFSAR Figure 15.0A.6-24, Protection Sequences for Pipe Breaks Outside Primary containment, and the discussion for Accident Event 36. The environment in the reactor building would contain high humidity and moisture after a HELB. After a LOCA, the environment in the reactor building would remain dry since the pipe break would be confined inside primary containment. Therefore the switches are EQ qualified for a LOCA, as required. A possible scenario was investigated regarding the consequences of failure of the valves during a HELB. Review of the Containment Atmosphere Control (CAC) system vacuum breaker containment control wiring diagram, CP&L drawing number LL-90046, showed that electrical failure of the pressure switches due to the presence of moisture in the reactor building could cause the circuit breaker to open and the valves either to remain closed, or close if they were in the open position. This would result in loss of position indication. The electrical failure would not affect other safety related equipment. Another scenario investigated was the effect of shorting across the switch which could possibly cause the valve to open. If this occurred, which is highly unlikely, loss of position indication would not occur, and operator action would be to remove power to the valves resulting in closure of the valves. The licensee did not consider failure of the switches during a HELB. The switches are not required to be operable during and after a HELB.

c. Conclusions

The inspectors concluded that the Barton Pressure switches will be operable during a LOCA as stated in UFSAR Section 15.

E7 Quality Assurance in Engineering Activities

The licensee has provided a site-wide, common program for identifying issues that require corrective action. The goal of the program was to improve overall plant performance by correcting conditions adverse to quality. During this inspection, the inspectors reviewed the administrative and procedural aspects of the program, the effectiveness of the corrective actions, the self-assessments and audits of the program, and other processes that provided for the incorporation of operating experience feedback.

E7.1 Corrective Action Program

a. Inspection Scope (40500)

The licensee's procedure, OPLP-04, Corrective Action Management, Revision 20 was reviewed by the inspectors. The procedure defines the policy for identifying, evaluating and correcting conditions not meeting expectations. The mechanism for accomplishing this policy is provided through the use of CRs.

The inspectors reviewed samples of the licensee documentation, observed CR review meetings, and conducted interviews with management and non-management personnel to evaluate the effectiveness of the licensee's Corrective Actions Program (CAP) and associated problem identification and corrective actions measures. The inspectors examined the licensee CR identification process, CR evaluation process, training and qualification of Root Cause Investigators and Quality Safety Reviewers (QSRs), CAP-related communications, implementation of corrective actions, and effectiveness of corrective actions. Licensee documents reviewed included a sampling of control room and turbine building operating logs, LERs, training and qualification records, root cause evaluation reports, licensee internal memoranda, CRs, and statistical/trending CR data. In particular, the inspectors reviewed all CRs associated with Unit 1 power uprate. The inspectors observed a CR review "working" meeting in which emergent CRs were classified, prioritized, and assigned responsibility for resolution. In addition, the inspectors observed a CR review "management" meeting in which newly dispositioned CRs were discussed.

b. Observations and Findings

The inspectors reviewed CRs which were classified as significant or important CRs in accordance with OPLP-04. The inspectors verified that operability assessments were performed, management reviews were completed, evaluations and recommended corrective actions were appropriate, and when required, appropriate immediate corrective actions were performed based on the nature of the problem.

CRs were the primary means through which the licensee identified, documented, and tracked the status of CAP items. The use of CRs was defined in OPLP-04, Revision 20. CRs were categorized into three levels of priority such that level 1 CRs were "significant conditions," level 2 CRs were "important conditions," and level 3 CRs were "minor conditions." During calendar year 1995 there were a total of 2153 CRs written (1 level 1, 174 level 2, and 1978 level 3). During calendar year 1996 there were a total of 3253 CRs written (1 level 1, 285 level 2, and 2967 level 3). The scarcity of level 1 CRs relative to the large quantity of level 3 CRs suggested that the delays in resolution of many of the more meaningful level 3 CRs, as described in NRC IR 50-325/96-14 and 50-324/96-14, could be aided by additional prioritization. Interviews with non-management licensee personnel indicated a trend toward increased acceptance and unfettered use of the CR process.

The inspectors reviewed a licensee memorandum, dated April 14, 1997, which reported the monthly status of Procedure Action Requests (PARs). This memorandum reported that 31 new PARs had been logged, 17 PARs had been closed out, and 117 PARs were outstanding. PARs were prioritized into three levels. There were no priority 1 PARs outstanding. The inspectors noted that two of the outstanding priority 2 PARs had been initiated in 1995. The first, entry 95-309, was a change to 0OP-37.5, "Service Building Heating and Ventilating System Operating Procedure." The second, entry 95-912, was a change to 0OP-59.1, "Hydrogen/Oxygen Storage Facility Operating Procedure." Revision 7 to 0OP-59.1 had been made April 9, 1996 and did not include the change associated with the PAR.

While assessing the program, the inspectors noted that some CRs were not completed within the time constraints established in OPLP-04. Deferrals are permitted by OPLP-04.1, Site Action Item Management, Rev. 5, with restrictions. The restrictions include stating the specific reason for extending the completion due date, status of efforts to satisfy the commitment, a revised schedule and the effect on safety if due date is not met. Approval of the responsible Superintendent/Unit Head is required to obtain the extension. The deferral rate was acceptable. The inspectors concluded that the controls in place have kept the number of deferrals in check, but this area of control needs improvement. The inspectors found a few deferrals without clear statement of why the deferral was needed. However after discussing the deferral with licensee personnel the reason was clarified. None of the deferrals were found to have an adverse effect on plant safety, but a stronger commitment toward timely completion of corrective actions would improver implementation of the CAP. The Self-Assessment manager periodically reviewed deferred CR's to assure they were in agreement with OPLP-04.1 and the number deferred was acceptable.

During interviews, licensee personnel described coordination difficulties relating to activities regarding the prioritization of computer software changes.

The inspectors reviewed the status of senior plant management training regarding the CAP and CRs conducted in response to the Notice of Deviation, identified in IR 50-325(324)/96-08. CR 96-02161 was initiated July 23, 1996 to generate this action item and track it to completion. Thirteen of the 36 training sessions listed in CR 96-02161 were associated with a licensee commitment to the NRC. The inspectors noted that a "Memo to Site Management Regarding CR 96-02161," dated April 17, 1997, reported six of the 36 training sessions were not completed. Five of these six, scheduled for Environmental & Radiation Control (E&RC) and Brunswick Engineering Support Section (BESS) management, were due for completion by August 31, 1996 to meet the NRC commitment date, September 30, 1996.

Nuclear safety reviews for activities at the Brunswick Nuclear Plant were performed by Qualified Safety Reviewers (QSRs) who are listed on the approved QSRs List in accordance with OAI-71. The inspectors reviewed the training requirements of OAI-109, "Performance of Nuclear Safety Reviews," and a sample of five training records associated with QSRs in the Maintenance, BESS, Material & Contract Services (M&CS), and Operations groups. The records reviewed indicated that those personnel received the required initial and continuous QSR training.

OPLP-04.3, "Root Cause Investigations," provided guidance on the process and techniques for Root Cause Investigation of significant or important conditions, groups of similar issues of minor significance considered collectively, or events which require action to prevent recurrence. This vital part of the CAP required that Root Cause Investigations of events and conditions involving human performance problems be conducted by trained investigators designated by the Supervisor-Corrective Action Program/Operating Experience. The individual investigator's qualification requirements were not specified in written procedures.

In practice, the investigators were trained by completion of a three-day licensee Root Cause Investigation course and a three-day course by Failure Prevention, Inc. (FPI) International in Root Cause Analysis and Corrective Actions. In addition, the inspectors noted that investigations involving equipment problems could be performed by a "knowledgeable engineer" who should consult with a Root Cause Investigator. There was no written requirement for Root Cause Investigation training for a "knowledgeable engineer."

The inspectors noted that some personnel interviewed expressed a need for: (1) enhanced clarity in CR guidance regarding recording the "most useful information" succinctly and consistently, (2) prompt communication to all personnel associated with an issue and personnel involved in issue investigation regarding the status of recommended corrective actions, and (3) enhancement of the continuous training provided to root cause investigators.

On April 16, 1997, the inspectors reviewed the 25 Power Uprate Projects items associated with licensee response to NRC Request For Additional Information (RAI)s and Commitments as recorded on CR 96-02758 and CR 97-01023. The inspectors noted that each item was appropriately addressed. In addition, the inspectors reviewed the April 16, 1997 listing of 26 other Unit 1 Power Uprate Project Condition Reports. The inspectors noted that 13 of the 26 CRs were closed, two regarding Final Safety Analysis Report mark-ups were completed and awaiting review by Regulatory Affairs, and 11 were on schedule for completion.

OPLP-04 states conditions whereby CRs which have been submitted to the CAP can be voided or downgraded. The reason for the change is required to be documented on the CR and the database. A sample of CRs that were voided were examined by the inspectors and the reason for voidance was stated as required by the procedure.

c. Conclusions

The inspectors concluded that the CAP was generally good, however timely implementation could be improved. Some CRs were not completed within the time constraints established in the procedure. Extensions were granted without a clear statement of why the deferral was needed. Items, such as operating procedure changes were backlogged over a year. This was reflective of a potential weakness in the prompt correction of identified problems. There were 3253 CRs generated in 1996 and the inspector reviewed a sample of CRs which indicated that the threshold for CR generation was sufficiently low to ensure identification of all meaningful problems. Furthermore, senior management training on CAP and CRs was behind the scheduled NRC commitment.

E7.2 Evaluation of Self-Assessment Program

a. Inspection Scope (40500)

The inspectors reviewed a variety of licensee self-assessments, self-assessment identified initiatives, and associated materials including: Brunswick Nuclear Plant 1997 Self Assessment Plan; Brunswick Training Section-Operations Initial Training Subunit Self Assessment Report, 96-014, dated November 21, 1996; Brunswick Engineering Support Section Primary Organizational Initiatives - Corrective Action Effectiveness, dated April 7, 1997; Power Uprate Response to Request for Additional Information, dated December 23, 1996; and Brunswick Nuclear Plant Corrective Action Management Monthly Status Report, dated February, 1997. In addition, the inspectors conducted interviews with management and non-management personnel involved in the performance and review of self-assessments.

b. Observations and Findings

The inspectors reviewed the Brunswick Nuclear Plant 1997 Self Assessment Plan which was promulgated by memorandum dated January 12, 1997. The plan documented and provided data related to 139 assessments, for example: Group, Due Date (original and new), Status, Lead and Assessor. The inspectors noted that the listing of assessments and information was extensive. Licensee personnel interviewed expressed that management review and tracking may be enhanced by consolidation of similar assessments and that there was a need for better standardization among the assessments.

OPLP-25, "Self-Assessment," Revision 5, did not clearly indicate how the disposition of Items for Management Consideration was to be documented. However, during interviews licensee management expectations were that managers should indicate the disposition of Items for Management Consideration. An example of this problem was evidenced in the lack of management's disposition of items on the Brunswick Training Section-Operations Initial Training Subunit Self Assessment Report, 96-014.

The inspectors reviewed the Brunswick Training Section-Operations Initial Training Subunit Self Assessment Report, 96-014. This self-assessment was conducted in response to CR 96-01932 and focused on the fundamentals training portion of the Non-Licensed Operator (NLO) Initial Training Program as it relates to ACAD 91-016, "Guidelines for Training and Qualification Activities of NLOs," dated September 1990. The inspectors noted that Report 96-014 addressed this limited subject to an appropriate level of detail. Report 96-014 identified two weaknesses: (1) NLO Initial and Requalification Training Programs did not include learning objectives for several important topics which were identified in ACAD 91-016, and (2) lesson plans which are used to fulfill Chemistry, Electrical, and Atomic fundamental topics did not include appropriate learning objectives. CR 96-03769 and CR 96-03770, respectively, were initiated to address these items.

The inspectors reviewed the BESS Primary Organizational Initiatives - Corrective Action Effectiveness. The actions outlined in this initiative were the result of BESS Self-Assessments and external audits indicating that the consistency of BESS corrective action implementation had not met its management's expectations. The inspectors found that this initiative was a comprehensive listing of the issues and provided an appropriate schedule of tasks. These issues included: (1) Organization - CAP ownership, trend analysis and corrective action effectiveness monitoring, and Root Cause Investigator training and qualification; (2) Management Involvement - establishment of a weekly CAP Review Board, a weekly status report of level 2 investigations, and a weekly CAP feedback to BESS supervision; (3) Backlog Management - disposition of the backlog of completed level 2 investigations, evaluation of the backlog of CR Action Items, establishment of reactor unit level CR and CR Action Item performance indicators and goals, identification and establishment of actions to enhance level 3 CR closure, integration of CAP and Maintenance Rule functions, and development of a fully electronic CAP tracking process; and (4) CAP Output/Trending Effectiveness - provide quarterly trend reports to the BESS management team and BESS staff, evaluation of BESS CR trend process methodology and use, and dissemination of significant CR investigations to BESS personnel.

c. Conclusions

The inspectors concluded that the Self-Assessment Program description and implementation by the licensee was generally good. The licensee used CP&L employees from other sites, contractor personnel, and information from industry sources outside CP&L for an independent perspective and comparison of self-assessment programs and processes used for corrective actions and review of potential issues. The inspectors noted that self-assessments often provided detailed identification of numerous problems and appropriate recommendations for corrective actions. However, personnel interviewed expressed that program effectiveness could be enhanced by consolidation and standardization of the numerous assessments.

E7.3 Operating Experience Program

a. Inspection Scope (40500)

The inspectors reviewed the licensee's Operating Experience (OE) Program.

b. Observations and Findings

Operating Experience is a part of the overall licensee's quality assurance process. The Operating Experience program ensures industry data are sent to applicable work units and that specific experience and data are supplied to other CP&L sites and the nuclear industry as appropriate. The inspectors reviewed Brunswick's procedure 0AI-02, Feedback of Operating Experience, Rev. 27. This procedure provided the guidance on reviewing and processing operating experience feedback. It covers source document receipt, screening evaluation, recommended actions, action tracking, action closeout, and program status reporting.

OE feedback item applicability screening, OE item evaluations, OE unit self-assessments, OE program Nuclear Assurance Section (NAS) audits and OE tracking and work backlogs were reviewed. The inspectors attended the weekly Operating Experience Assessment meeting, interviewed plant personnel and observed end use activities of the OE program. OE source document screening items B 11770, B 11235, B 11743 and B 11495 were reviewed. The screening reviews were performed in accordance with OAI-02 and the applicability reviews and recommended actions were acceptable. The inspectors reviewed completed OE item evaluation for B 11495. The evaluation was completed in accordance with OAI-02 and was timely and through.

Self assessments and NAS audits were performed on the OE program. The inspectors reviewed OE self-assessment RA-96-03, SOER/SEL Implementation Effectiveness and PES audit, dated November 3, 1996, Report on assessment of some Aspects of Nuclear Plant Operating Experience Program, were reviewed. Both the PES audit and the self-assessment identified weaknesses and strengths. CRs were initiated for resolution of weakness items. The inspectors concluded that the self-assessments and PES audits were thorough and that the findings were substantive. The licensee was taking actions to address the findings.

The inspectors attended the Operating Experience Assessment (OEA) unit weekly status meeting and observed that the staff reviewed the OE items screened for the week and verified the acceptability of the item dispositioned. Condition Reports processed for the week were reviewed including their classification. The OEA staff selected one or more OE item for discussion at the meeting. This provided positive feedback for the Operating Experience Feedback (OEF) process and demonstrated a commitment to problem identification and resolution.

The OEA unit tracking and work backlog was reviewed on May 1, 1997. A report was prepared by the OE reviewer which addressed the items processed for the month and tracked the evaluations issued, and the status and age of open evaluations. The report indicated 13 items were awaiting review and assignment. The inspectors determined that the backlog was not excessive and no items were being deleted or deferred. Interviews with personnel indicated the OE program was being evaluated and utilized in a timely manner.

The licensee had incorporated OE feedback data into several routine activities. The inspectors observed that OE items were discussed at morning shift turnover meetings. The procedure for conducting pre/post job briefings for routine and infrequently performed tasks requires that a briefing include a discussion of applicable OE data for the evolution. An OE file was maintained for use in conjunction with the OE database for conducting briefings. The files contained experience data identified by the NRC, other CP&L sites, and industry events. The frequent use of OE data during routine activities was viewed as a strength.

Discussions with site personnel indicated that overall performance was improved with the frequent use of OE data. OE information was used in briefings for all major evolutions of shutdown, cooldown, startup, and heatup.

c. Conclusions

The Operating Experience Program was determined to be effective. The completed OE evaluations reviewed were acceptable. The OE staff assessments and PES audits of OE were thorough. The weekly status meeting, status reports and OE tracking provided good program oversight. The incorporation of OE data into routine and non-routine activities was viewed as a strength.

E8 Miscellaneous Engineering issues (92903)

(CLOSED) VIO (C), 50-325(324)4/96-001-04: Failure to Implement Corrective Action to Resolve Errors in Design Guide II.1.

The inspector verified the corrective actions described in the licensee's response, dated April 17, 1996 and accepted by the NRC on April 30, 1996. DG II.1 has been converted to NGGC procedure EGR-NGGC-0352 with all outstanding discrepancies corrected. Further, training has been completed for appropriate Engineering, Regulatory Affairs personnel and corrective action coordinators to ensure they are sensitized to what is required for action item closure. This item is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Control

R1.1 Status of RP&C Facilities and Equipment

a. Inspection Scope (71750)

The inspector reviewed the status of ongoing licensee efforts to reduce PCEs.

b. Observations and Findings

In NRC IR 50-325(324)/97-02 the inspector discussed the corrective action taken by the licensee for contamination controls. During daily morning meetings the licensee had seen the results of these efforts by a reduction in PCEs. On May 22, 1997, the licensee discussed that there had been no PCEs for the past two months. The licensee reviewed records for the past three years noting there had never been a single month with no PCEs.

c. Conclusions

The inspector concluded that corrective action taken by the licensee had been effective to reduce the number of PCEs.

R8 Miscellaneous RP&C Issues (92904)

R8.1 (CLOSED) VIO (D), 50-324/96-001-05: Failure to Maintain Control Over Locked High Radiation Area.

Corrective actions described in the licensee's response, dated April 17, 1996, were verified as completed by the inspector. The actions were accepted by the NRC on April 30, 1996 and included additional training and several procedure revisions. This item is closed.

R8.2 (CLOSED) VIO (E) 50-325(324)/96-001-06: Failure to Have a Procedure for Resin Flush.

The license's response to this violation, dated April 17, 1996 described corrective actions to resolve this issue. The response was accepted by the NRC on April 30, 1996. The actions were verified by the inspector as completed. They included issuance of procedures for performing "hot spot" flushing and an assessment of effectiveness of the interim organization established for control of Rad Waste. The procedure, OE&RC-0010 was issued May 8, 1996 and the interim organization assessment was completed October 19, 1996. This item is closed.

R8.3 (CLOSED) VIO (C) 50-324/96-013-05: Failure to Correctly Update ARM Alarm Setpoint.

The inspector reviewed and verified the licensees corrective actions described in the response letter, dated November 11, 1996. The response was accepted by the NRC on November 27, 1996. Corrective actions were completed by December 18, 1996. This item is closed.

R8.4 (CLOSED) VIO (D), 50-325(324)/96-013-06: Failure to Follow Radiological Control Procedures.

The licensees response, dated November 11, 1996 was reviewed and accepted by the NRC on November 27, 1996. Corrective actions were verified by the inspector as completed. The actions included counseling personnel on human performance errors, and providing increased supervisory oversight for refueling floor activities. The lessons learned from this event were included in the radiation protection technician continuing training program, which was completed on April 2, 1997. This item is closed.

R8.5 (CLOSED) VIO (E) 50-325(324)/96-013-07: Failure to Perform Surveys.

The licensees response, dated November 11, 1996 was reviewed and accepted by the NRC on November 27, 1996. The inspector verified the corrective actions as completed. The actions included holding stand-down meetings with personnel to discuss the importance of Radiation Work Permits (RWPs) and staff expectations. Additional training was completed by April 2, 1997. This item is closed.

S1 Conduct of Security and Safeguards Activities

S1.1 Security Drills

a. Inspection Scope (71750)

On May 14, 1997, the inspector observed the performance of two security drills.

b. Observations and Findings

The inspector observed a security drill performed at 5:50 a.m., just prior to the security shift turnover. Under the supervision of the Site Security Manager, a plant worker placed a bag containing a hand gun onto the x-ray machine. Once the guard saw the gun image on the x-ray machine, the guard promptly announced the situation. Another guard told others in the area to stand aside and proceeded to place the individual attempting to bring the gun into the plant in handcuffs. The Site Security Manager promptly announced this was a drill and closely monitored the drill. The drill was terminated and a critique was conducted.

At 2:00 p.m., the inspector observed the performance of another security drill. The drill scenario adequately simulated a plant worker attempting to gain access while in possession of a weapon. The inspector noted that search techniques used effectively identified the weapon. The security responder communicated the situation appropriately upon discovery. The inspector observed that the security responders maintained control of the armed individual throughout the scenario and at no time was access to the plant physically challenged. Site Security supervision and training personnel were present and maintained satisfactory oversight of the scenario. The drill was terminated and a critique conducted. The inspector noted that the drill scenario was terminated before completion of the drill search activities. Despite early drill termination, the drill responder completed all search activities to ensure no additional weapons existed.

c. Conclusions

The inspector concluded that the plant security guards successfully identified attempts to bring a weapon into the plant during the performance of two drills.

S1.2 Operability Test of Search Equipment

a. Inspection Scope (71750)

On May 14, 1997, the inspector reviewed operability testing of the personnel access portal test equipment.

b. Observations and Findings

The inspector observed the testing of the explosive detector, metal detector, x-ray machine, and associated alarms. This testing was performed at the beginning of the shift. The explosive detector was tested using a test sample. The metal detector was tested using a test source. In addition, the hand-held metal detector was tested. The x-ray machine was tested using a wedge source. All equipment passed the operability test.

c. Conclusions

The inspector concluded that all personnel access portal equipment was properly tested. No equipment problems were identified.

S8 Miscellaneous Security and Safeguards Issues (92904)

(Closed) IFI 50-325(324)/97-05-10: Personnel Access Search Training

This item was opened due to recent errors that occurred during a controlled drill scenario as part of an NAS assessment. In response to these findings additional training was conducted for all site security personnel. This training included a security stand-down to discuss the errors. Training was conducted during March 1997. The inspector reviewed the training lesson plans entitled, Personnel Access Control LP 3003 and Vehicle Search/Control Procedures LP 3005. This training discussed the proper method of conducting a hands-on search and utilizing the hand-held metal detector. Both the x-ray operator and search line officer responsibilities and response during discovery of contraband were discussed. In addition, on May 14, the inspector observed the satisfactory performance of two security drills as discussed in paragraph S1.1. Based on the actions taken this item is closed.

F1 Control of Fire Protection Activities

F1.1 Resolution of Thermo-Lag Fire Barrier Issue (64704)

a. Inspection Scope

The inspector reviewed the action taken to resolve the degraded Thermo-Lag fire barrier issue to determine if the licensee's action was consistent with commitments made to the NRC.

b. Observations and Findings

In 1991, the NRC identified that Thermo-Lag fire barrier material did not perform to the manufacturer's specifications. NRC Bulletin 92-01 "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage" was issued which requested licensees with Thermo-Lag fire barriers to consider these fire barriers to be degraded and take appropriate compensatory measures for the areas where the Thermo-Lag fire barriers were installed.

Brunswick Units 1 and 2 originally had Thermo-Lag protection for approximately 2300 linear feet of conduit, 1250 square feet of junction box/door transom protection and 600 square feet of cable tray protection. The material is located in the Control Building, Reactor Buildings, Diesel Generator Building and Service Water Building.

The licensee has evaluated the results of data from various tests performed by the nuclear industry on Thermo-Lag fire barrier installations. The licensee has developed and implemented design changes to reroute approximately 3000 feet of safe shutdown cables per unit to eliminate the need for the existing Thermo-Lag safe shutdown circuit protection. The cable rerouting was completed during the respective 1996 unit refueling outages. Removal of Thermo-Lag enclosing the diesel generator exhaust fan power cables has been completed.

The licensee has completed engineering evaluations for use of Thermo-Lag on fire door transoms. Revisions to combustible loading calculations to reflect Thermo-Lag combustibility and ampacity derating evaluations have been completed. Fire and seismic evaluations are complete to support abandoning non-credited Thermo-Lag in-place.

The license has upgraded existing concrete walls as fire barriers and performed revisions to the safe shutdown analysis to reduce dependence on Thermo-Lag protection. On August 31 1995, January 10, 1996, and November 21, 1996, the licensee submitted to the NRC a request for exemption from certain technical requirements of Sections III. G. and III. L. of Appendix R to 10 CFR 50 to permit use of Automatic Depressurization and Low Pressure Injection as an alternate means of shutdown. This item is presently under review by the NRC.

As of the date of this inspection, the licensee had initiated the implementation of corrective actions for Thermo-Lag issues, except for the removal and replacement with a non-combustible material of approximately 20 linear feet of Thermo-Lag material enclosing cable trays, which act as an intervening combustibles and the installation of approximately 25 feet of three hour fire wrap for conduit protection. NRC approval of the exemption request and implementation of the changes associated with the exemption are pending. Following completion of these items, no Thermo-Lag will be used to provide safe shutdown circuit protection at Brunswick.

c. Conclusions

The licensee has been actively pursuing the resolution of the Thermo-Lag issue.

F1.2 Fire Reports

a. Inspection Scope (64704)

The inspector reviewed the plant fire incident reports for 1996 and 1997, to assess maintenance related or material condition problems with plant systems and equipment that initiated fire events. The inspector verified that plant fire protection requirements were met in accordance with procedure FPP-019, Incident Report and Investigation, Revision 11, when fire related events occurred.

b. Observations and Findings

The fire incident reports indicated that there were four incidents of fire in 1996, and two fire events in 1997, which required fire brigade response. No significant fires had occurred during this period. Only one of the six fires had occurred within the plant power block within the protected area.

c. Conclusions

Good compliance with plant fire prevention procedures has resulted in a low incident of fire within the plant protected area.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Operability of Fire Protection Facilities and Equipment (64704)

a. Inspection Scope (64704)

The inspectors reviewed the open maintenance work orders, maintenance history, and daily Impairment Reports on the facility's fire protection systems and features, and inspected these items to determine the performance trends and the material conditions of this equipment.

b. Observations and Findings

Maintenance Observations:

As of May 20, 1997, the total number of open maintenance work requests related to the fire protection systems and features was 56. These work requests were grouped as follows:

Fire Protection Water Systems/ Penetration Seals (System 6175)	36
CO_2 (System 6195)	4
Halon (System 6205)	0
Communications (System 6030)	3
Emergency Lighting (System 5215)	[illegible]
Fire Detection (System 6180)	1
	56

All except two of these work requests were issued in late 1996 or 1997. The work requests issued prior to late 1996 were minor repairs or painting which did not affect the operability of these systems. Work was in process to correct these issues.

There was not a backlog of open work requests. Corrective maintenance on degraded fire protection systems was being accomplished in a timely manner. The emergency lighting system sustained reliability problems during the past year due to a number of operational failures. The licensee had taken positive corrective action initiatives to resolve these concerns.

Fire Protection System Status:

A review of the Loss Prevention Unit daily Impairment Reports for May 17-21, 1997 indicated the following fire protection components or systems were out of service:

Fire Protection System	Impairment Number(s)
Thermo-Lag Fire Barriers	A2-93-F025
Fire Doors	A2-96-F582 &F607
Cable Coating	A2-96-F157
Emergency Lighting	A1-97-0405 through -0411

The inspector noted that a number of emergency lighting units were out of service. This high number was attributed to the current periodic testing and the associated repairs in process for the identified discrepancies. Appropriate compensatory measures had been implemented for the fire protection features which were out of service.

The impairment status report provided the licensee with a good means of identifying out of service fire protection equipment to assure that appropriate compensatory measures were implemented.

During the plant tours, the inspector noted that the maintenance and material condition of the fire protection equipment was satisfactory.

c. Conclusions

There was not a backlog of open work requests. Corrective maintenance on degraded fire protection systems was being accomplished in a timely manner. The maintenance and material condition of the fire protection equipment was satisfactory.

The fire protection impairment status report provided the licensee with a good means of identifying out of service fire protection equipment to assure that appropriate compensatory measures were implemented.

F2.2 Unit 1 HPCI CO_2 System Inoperable

a. Inspection Scope (71750 and 64704)

On April 9, 1997, the inspectors reviewed the compensatory fire protection measures associated with the Unit 1 High Pressure Coolant Injection (HPCI) System Carbon Dioxide (CO_2) System being out of service.

b. Observations and Findings

On April 19, 1997, the Unit 1 HPCI System and associated CO_2 system were taken out of service for scheduled maintenance. The inspectors toured the HPCI room and observed a sign on a fire hose station in the adjacent residual heat removal (RHR) pump room. The sign stated that the hose station at RB-25 was the compensatory measure for HPCI CO_2 system being inoperable. There was no additional equipment or compensatory measures established other than placing a sign on an existing hose station.

The inspectors reviewed the active fire impairment with the Loss Prevention Unit (LPU). Impairment AI-97-F093 was initiated at 11:42 p.m. on April 8, 1997, due to the Unit 1 HPCI CO_2 being under clearance due to work activities. The fire detectors associated with the Unit 1 HPCI CO_2 system had been disabled with the system impairment, however, general plant fire detection remained available in the room. The compensatory measures were listed as back-up fire suppression. The inspectors discussed with LPU that no fire watches were established for the unprotected area and that the measures taken to establish back-up fire suppression did not provide hose protection of both the RHR pump room and the HPCI room. The inspectors stated that typically when an existing hose station was being used as a backup fire protection for another hose station or unprotected area a gated-Y connection was installed with an additional hose reel laid of sufficient length to reach the backup area.

The inspectors reviewed plant procedure OPLP-01.2, Fire Protection System Operability, Action, and Surveillance Requirements. Section 6.3, High Pressure CO_2 Systems, which requires that with one or more of the required CO_2 systems inoperable, backup fire suppression equipment must be established for the unprotected area within one hour and signs placed at the backup fire suppression equipment to identify the proper fire hose to be used. No requirement to establish a fire watch was included as a requirement. The inspectors also reviewed the Brunswick original TSs for the High Pressure CO_2 Systems, and noted that the present OPLP-01.2 requirements were consistent with the TSs. The operability, surveillance and test requirements for the fire protection systems and features had been removed from the TSs and incorporated into plant procedure OPLP-01.2. However, Sections 6.2, Spray and/or Sprinkler Systems, and 6.7, Halon System of OPLP-01.2 both require that a fire watch with backup fire suppression for the unprotected area be established within one hour. The inspectors identified this inconsistency of fire watch compensatory measures with operations LPU.

The licensee corrected this example of inconsistent fire protection compensatory measures with an upgrade (Revision 9) to plant procedure OPLP-01.2, that added a requirement to establish a continuous fire watch and backup fire suppression equipment with additional equivalent fire hose capacity for the unprotected area within 1 hour when a CO_2 system is inoperable. The inspectors considered this acceptable.

c. Conclusions

An inconsistency in the implementation of compensatory measures for the High Pressure CO_2 Systems was identified and corrected.

F2.3 HPCI/RCIC Alternate Safe Shutdown Compensatory Measures

a. Inspection Scope (71750)

The inspector reviewed the actions taken upon the unsatisfactory performance of Periodic Tests, OPT-09.10.L, HPCI System Component Local and Alternate Safe Shutdown (ASSD) Control and Manual Operability Test and OPT-10.12.L, RCIC Steam Supply Inboard and Outboard Isolation Valves, Turbine Steam Supply Valve, and Turbine Trip and Throttle Valve Local ASSD Operability Test.

b. Observations and Findings

On May 8, 1997, during routine review of the Unit 2 control room logs, the inspector noticed the partial performance of Periodic Tests OPT-9.2, HPCI Operability Test and OPT-9.3A, HPCI System Component Test on Unit 2. The logs indicated that the 2-E41-F002, HPCI Steam Supply Inboard Isolation Valve failed to stroke to the closed position during testing of the valve local control station at Motor Control Center (MCC) 2XB. The purpose of OPT-09.10.L was to verify that various HPCI valves could be stroked from their remote and ASSD control stations. The additional tests, PTs 9.2 and 9.3A, were performed to ensure HPCI operability and primary containment isolation capability per TS 4.5.1 and 4.6.3.1.

On May 16, 1997, during routine review of the Unit 2 control room logs, the inspector noticed the unsatisfactory completion of OPT-10.12.L. The logs indicated that during the performance of the test the 2-E51-F007, RCIC Steam Supply Isolation Valve invertor failed to energize therefore the ASSD function of that valve could not be verified. A partial performance of OPT-10.1.1 was conducted to verify the RCIC operability and primary containment isolation capability function.

The inspector questioned licensee staff about compensatory measures for the loss of the ASSD capability for both valves. The licensee indicated that a 14 day impairment was entered into the operator log. The impairment allows 14 days for restoration of equipment's ASSD function. No additional actions were performed to provide compensatory actions for the loss of the ASSD function. The inspector reviewed associated ASSD procedures, plant fire protection procedures, the physical location of the components, and verified that fire detection and suppression systems were operable during the duration of degraded ASSD function. The inspector determined that no assessment was conducted to determine if additional compensatory measures were necessary to address degradation in ASSD response capability due to failures of ASSD components on both trains of ASSD equipment.

Pending further licensee investigation and NRC review this item is identified as Unresolved Item URI 50-325(324)/97-07-03, Lack of ASSD Compensatory Measures.

c. Conclusions

The inspector determined that no compensatory measures were taken when ASSD equipment was taken out of service. These issues were identified as unresolved pending further review.

F3 Fire Protection Procedures and Documentation

a. Inspection Scope (64704)

The inspector evaluated the adequacy and implementation of the licensee's Fire Protection Program described in the Updated Final Safety Analysis Report (UFSAR) and in Plant Operating Manual Fire Protection Procedure OPLP-01, "Fire Protection Program Document." In addition, a comparison was made of the program to selected NRC Safety Evaluation Reports which approved the station fire protection program. The inspector reviewed the following procedures for compliance with the NRC requirements and guidelines:

- OPLP-01, Revision 5, Fire Protection Program Document
- OPLP-01.1, Revision 10, Fire Protection Commitment Document
- OPLP-01.2, Revision 8, Fire Protection System Operability, Action, and Surveillance Requirements
- FPP-013, Revision 24, Transient Fire Load Evaluation
- FPP-014, Revision 15, Control of Combustible, Transient Fire Loads and Ignition Sources
- FPP-017, Revision 15, Hot Work Permit

- FPP-019, Revision 11, Incident Report and Investigation

Plant tours were also performed to assess procedure compliance.

b. Observations and Findings

The above procedures were the principle procedures issued to implement the facility's fire protection program. These procedures contained the requirements for program administration, controls over combustibles and ignition sources, fire brigade organization and training, and operability requirements for the fire protection systems and features. The procedures were well written and met the licensee's commitments to the NRC.

A general plant walkdown inspection was performed by the inspector to verify: acceptable housekeeping; compliance with the plant's fire prevention procedures such as "Hot Work" permits and transient combustibles; operability of the fire detection and suppression systems; emergency lighting; and, installation and operability of fire barriers, fire stop and penetration seals (fire doors, dampers, electrical penetration seals, etc.).

Within the areas observed, the inspector determined that general housekeeping was satisfactory. Fire retardant plastic sheeting and film materials were being used. Lubricants and oils were properly stored in approved safety containers. Appropriate controls for cutting and welding operations were being enforced. Controls were being maintained for transient combustibles, however, the inspector observed reduced efforts in containing lubrication oil leaks within the diesel generator rooms. Within the diesel rooms numerous oil soaked absorbent pads and filled catch containers were observed. Both the NRC Resident and regional staff discussed the observed reduced efforts in containing diesel generator lubrication oil leaks with Operations and Engineering management.

c. Conclusions

The fire protection program implementing procedures were good and met licensee and NRC requirements. Implementation of the fire protection and prevention procedures and the general housekeeping for control of combustibles within the plant were satisfactory, except, the inspector observed reduced efforts in containing lubrication oil leaks in the diesel generator rooms.

F5 Fire Protection Staff Training and Qualification

a. Inspection Scope (64704)

The inspectors reviewed the fire brigade organization and training program for compliance with the NRC guidelines and requirements.

b. Observations and Findings

The organization and training requirements for the plant fire brigade were established by OFPP-051, Loss Prevention Emergency Response Qualification/Training and Drill Program. The fire brigade for each of five shifts was composed of a LPU fire brigade commander and at least four additional brigade members. Each operations shift also had a SRO Advisor assigned to respond to fires with the fire brigade.

As of the date of this inspection, there were a total of 22 fire brigade members from operations LPU and 16 from Environmental and Radiation Control (E&RC) on the plant fire brigade. The inspector verified that sufficient shift personnel were available to staff each shift's fire brigade with at least five qualified fire brigade members. Each fire brigade member was required to receive initial, quarterly and annual fire fighting related training and to satisfactorily complete an annual medical evaluation and certification by a physician for participation in fire brigade fire fighting activities. In addition, each member was required to participate in at least two drills per year.

On May 19, the inspector observed a fire brigade drill involving a simulated fire at the plant paint storage building in the plant yard area. The fire brigade commander and four fire brigade members responded with the plant fire engine and in full fire fighting turnout gear. Personnel from operations with the shift advisor also responded to the drill. An offensive fire attack was mounted utilizing a 1 1/2-inch attack fire hose line, followed by additional fire hose line attack practice in the yard area. The fire brigade commander properly deployed the fire brigade personnel, established a command post and effectively used radio communications. A drill critique was conducted with the fire brigade members following the drill to discuss the drill, participants performance and recommendations for improvements. The action by the brigade met the established drill objectives.

c. Conclusions

The fire brigade organization and training met the requirements of the site procedures and the performance by the fire brigade to a drill during this inspection was good.

F6 Fire Protection Organization and Administration

a. Inspection Scope (64704)

The licensee's management and administration of the facility's fire protection program were reviewed for compliance with the commitments to the NRC and to current NRC guidelines.

b. Observations and Findings

The designated onsite manager responsible for the administration and implementation of the fire protection program was the Operations

Manager. This responsibility had been delegated to the Superintendent Loss Prevention Unit. The Superintendent Loss Prevention Unit reports to the Operations Manager and was responsible for the station fire protection program, general maintenance of fire protection systems and equipment, and ensuring that the appropriate fire prevention procedures and fire brigade programs were implemented. Plant work requests were reviewed by the Loss Prevention Unit for identification of fire safety hazards. Coordination of the station's fire protection program requirements was provided by a fire protection system engineer in the Loss Prevention Unit.

c. Conclusions

The coordination and oversight of the facility's protection program met the licensee's procedures and commitments in the UFSAR. The personnel assigned various fire protection related functions within Loss Prevention Unit were working together as a team and with coordination by the Superintendent to implement the fire protection program at the site.

F7 Quality Assurance in Fire Protection Activities

a. Inspection Scope (64704)

The following audit report and the plant response to the issues were reviewed:

- Nuclear Assessment Section Report B-FP-96-01 Brunswick Annual Fire Protection Assessment, dated July 23, 1996
- Loss Prevention Unit Self Assessment Report OPS-97-04, Fire Detection System Performance and Reliability, dated April 8, 1997

b. Observations and Findings

The licensee's Nuclear Assessment Section performed a two week assessment of fire protection on June 24 through July 1, 1996. The report for this assessment was Report No. H-FP-97-01. Findings from these assessments were categorized as strengths, issues, or weaknesses.

The assessment report identified one strength, two issues and two weaknesses. The strength identified by the Nuclear Assessment Section assessment noted a high degree of ownership for fire protection activities by the Loss Prevention Unit. The issues included problems with the physical condition of fire doors in the Emergency Diesel Building (I1) and maintenance of qualification cards for Loss Prevention Unit technicians (I2). The Loss Prevention Unit Self Assessment on the fire detection system identified that false detection alarms were not excessive and corrective maintenance and system impairments were addressed in a timely manner. Planned corrective actions in response to the identified issues were addressed in the licensee response and were acceptable.

c. Conclusions

The Nuclear Assessment Section assessments and Loss Prevention Unit self-assessments of the facility's fire protection program was comprehensive and was effective in identifying fire protection program performance deficiencies to management. Planned corrective actions in response to the audit issues were acceptable.

F8 Miscellaneous FP Issues (92904)

(CLOSED) Deviation 50-325(324)/96-008-12: Failure to Implement a Fire Protection Procedure.

The inspector verified the corrective actions described in the licensee's response, dated August 19, 1996 which were accepted by the NRC on September 27, 1996 as adequate. The primary corrective action was a revision of OFPP-14, Control of Combustibles, Transient Fire Loads, and Ignition Sources. Rev. 15 was issued July 1, 1996. This item is closed.

V. Management Meetings

XI Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on May 29, 1997. Post inspection briefings were conducted on April 18, May 1, May 20, and May 23, 1997. The licensee acknowledged the findings presented.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Barnes, Manager Training
A. Brittain, Manager Security
N. Gannon, Manager Maintenance
J. Gawron, Manager Nuclear Assessment
S. Hardy, Engineer, Brunswick Engineering Support Section
E. Harkcom, Superintendent, Loss Prevention Unit
S. Hinnant, Vice President, Brunswick Steam Electric Plant
L. Illy, Principle Engineer, EQ, Nuclear Engineering Department (NED)
K. Jury, Manager Regulatory Affairs
W. Levis, Director Site Operations
B. Lindgren, Manager Site Support Services
R. Lopriore, General Plant Manager
J. Lyash, Manager, Engineering Support Section
R. Miller, Superintendent, Design Control Unit, NED
R. Mullis, Manager Operations
J. Reinsburrow, Manager Operations Support
R. Schlichter, Manager Environmental and Radiation Control
R. Sims, Fire Protection Engineer
M. Turkal, Supervisor Licensing and Regulatory Programs

Other licensee employees or contractors included office, operation, maintenance, chemistry, radiation, and corporate personnel.

E. Brown
J. Arildsen
F. Jape
J. Lenahan
C. Patterson
D. Trimble
G. Wiseman

INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 37551:	Onsite Engineering
IP 37700:	Design Changes and Modifications
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 64704:	Fire Protection/Prevention Program
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92720:	Corrective Action
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-325(324)/97-07-01	URI	Failure to Enter TS ACTION Statement (paragraph O3.1)
50-325(324)97-07-02	NCV	Failure to Properly Translate Design Basis Information into the Power Uprate Design Change (paragraph E1.1)
50-325(324)/97-07-03	URI	Lack of ASSD Compensatory Measures (paragraph F2.3)

Closed

50-325(324)/97-02-02	URI	Recirculation Pump Transients (paragraph O8.1)
50-325/95-011-00	LER	Manual Reactor Protection System Trip Due to Decreasing Condenser Vacuum (paragraph O8.2)
50-325(324)/96-013-01	VIO	Equipment Clearance Error (paragraph O8.3)
50-325(324)/96-013-02	VIO	Failure to Complete the LSRO Training Program Prior to Taking the LSRO Audit Examination (paragraph O8.4)
50-325(324)/96-013-03	IFI	LOR Makeup for Licensed Operators (paragraph O8.5)
50-325/95-005-00	LER	Unplanned ESF Actuation Due to RB Ventilation Exhaust Radiation Monitor Power Supply Failure (paragraph M8.1)

50-325/95-014-00	LER	Unplanned ESF Action Due to Low Voltage Spike on B RPS (paragraph M8.2)
50-325/95-014-01	LER	Unplanned ESF Action Due to Low Voltage Spike on B RPS (paragraph M8.2)
50-325/95-018-00	LER	Automatic Reactor Shutdown Due to Condensate/ Feedwater Transient (paragraph M8.3)
50-325/96-006-00	LER	Technical Specification Surveillance Acceptance Criteria Did Not Adequately Account for Head Losses (paragraph M8.4)
50-325/96-010-00	LER	Emergency Diesel Generator #1 DC Control Power Breaker Cable Termination (paragraph M8.5)
50-325/96-007-00	LER	Double Disk Gate Valve Leakage (paragraph M8.6)
50-325/96-007-01	LER	Double Disk Gate Valve Leakage (paragraph M8.6)
50-325(324)/96-008-02	IFI	Double Disc Gate Valve Leakage (paragraph M8.6)
50-325(324)/96-001-04	VIO	Failure to Implement Corrective Action to Resolve Errors in Design Guide II.1 (paragraph E8)
50-324/96-001-05	VIO	Failure to Maintain Control Over Locked High Radiation Area (paragraph R8.1)
50-325(324)/96-001-06	VIO (E)	Failure to Have a Procedure for Resin Flush (paragraph R8.2)
50-324/96-013-05	VIO (C)	Failure to Correctly Update ARM Alarm Setpoint (paragraph R8.3)
50-325(324)/96-013-06	VIO (D)	Failure to Follow Radiological Control Procedures (paragraph R8.4)
50-325(324)/96-013-07	VIO (E)	Failure to Perform Surveys (paragraph R8.5)
50-325(324)/97-05-10	IFI	Personnel Access Search Training (paragraph S8)
50-325(324)/96-008-12	DEV	Failure to Implement a Fire Protection Procedure (paragraph F8)