U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-325, 50-324
License Nos:	DPR-71, DPR-62
Report No:	50-325/96-15, 50-324/96-15
Licensee:	Carolina Power & Light (CP&L)
Facility:	Brunswick Steam Electric Plant, Units 1 & 2
Location:	8470 River Road SE Southport, NC 28461
Dates:	September 15 - October 26, 1996
Inspectors:	C. Patterson, Senior Resident Inspector M. Janus, Resident Inspector E. Brown, Inspector In Training J. Coley, Reactor Inspector (Section M2 and M8) M. Miller, Reactor Inspector (Section M1.3) R. Chou, Reactor Inspector (Section E2.4, E2.5) J. Lenahan, Reactor Inspector (Section E1.1, E2.3, E5.1)
Approved by:	M. Shymlock, Chief, Projects Branch 4 Division of Reactor Projects

9612030012 961122
PDR ADOCK 05000324
G PDR

EXECUTIVE SUMMARY

Brunswick Steam Electric Plant, Units 1 & 2
NRC Inspection Report 50-325/96-15, 50-324/96-15

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 maintenance, in-vessel inspections, and engineering inspections by regional inspectors.

Operations

An unresolved item was identified concerning vessel disassembly while secondary containment was inoperable. (Section O1.1). This was a conscious action by the licensee although contrary to technical specification requirements. This item was unresolved pending further review of the technical specifications and licensee's risk assessment.

An unresolved item was identified concerning a loss of shutdown cooling. (Section O2.2). Repairs were being made to an instrument rack that contained the pressure switch to isolate shutdown cooling. Further review of the shutdown risk assessment was being completed.

Maintenance

A noncited violation was identified concerning securing of wheeled equipment and carts in the plant. (Section M1.1). The licensee corrected the specific problems and revised their procedure.

The alternate remote shutdown equipment and panels have been maintained in a satisfactory manner except for the material condition of two main Remote Shutdown Panels which were considered poor. (Section M1.3).

The reactor vessel core shroud ultrasonic examination efforts observed by the inspector were conducted in an exemplified manner. (Section M2.1). Scan plans, procedures, personnel, and equipment were integrated to obtain the best possible inspection results. In-vessel visual inspections were also performed in an effective manner.

Engineering

The licensee's progress to correct EQ program deficiencies was satisfactory. (Section E1.1). No equipment operability issues were identified.

An apparent violation was identified concerning exceeding the maximum thermal power allowed by the license and a technical specification thermal limit. (Section E2.1). This occurred due to inadequate testing of the plant process computer after installation in 1994.

A repeat violation was identified concerning failure to take corrective action to correct the cause of chlorine detector failures. (Section E2.2). Five out of eight detectors failed on September 19, 1996. This

will be the third licensee event report concerning these failures in less than two years. Previous corrective action has been ineffective to correct the problems.

The licensee's inspections of Unit 1 feedwater system to evaluate the system for possible water hammer damage was inadequate. (Section E2.3). This was identified as a weakness.

One violation was identified concerning deficiencies in the modifications for USI A-46. (Section E2.5). One unresolved item was identified due to lack of inspection requirements by QC to inspect safety-related miscellaneous structural steel.

Plant Support

A violation was identified concerning failure to follow procedure to assess proper radiation monitor response. (Section R3.1).

Nuclear Assessment Section audits have become more aggressive in identification of issues. (Section R.7). However, key department managers have been reluctant at times to accept valid findings.

A noncited violation was identified concerning the conduct of a criticality monitor drill as required by 10 CFR 70.24. (Section P1). The licensee conducted a drill and revised their procedure to perform drills in the future.

Report Details

Summary of Plant Status

Unit 1 was at power at the start of this inspection report. The unit restarted after Hurricane Fran on September 7, 1996, and operated until the unit was shutdown on October 5, 1996, to begin a 35 day refueling outage. At the end of the inspection report period refueling was in progress.

Unit 2 likewise was restarted after Hurricane Fran and operated continuously without significant problems during this inspection report period. At the end of the inspection report period the unit had been on-line 44 days.

I. Operations

O1 Conduct of Operation

O1.1 Vessel Disassembly Without Secondary Containment

a. Inspection Scope (71707)

The inspector reviewed the vessel disassembly operations. This evolution was performed without secondary containment.

b. Findings and Observations

The inspector attended the 6:30 a.m. shift outage meeting on October 7, 1996. During review of the Key Safety Function Status it was discussed that both trains of Standby Gas Treatment (SBGT) were not available and secondary containment was functionally available. During the evening shift, at 10:24 p.m. on October 6, 1996, permission was given to lift the reactor pressure vessel head per OSPP-RPV501, Reactor Vessel (And Associated Components) Disassembly for Refueling. The inspector questioned licensee management concerning disassembly of the vessel internals without the benefit of secondary containment being operable, which becomes primary containment during refueling. This was of particular concern since during the operating cycle two control rods were inserted to suppress the flux around leaking fuel.

The inspector reviewed the operator and outage logs and noted the following sequence of operations:

10/06/96	22:24	Permission given to lift the vessel head
10/06/96	22:56	LCO #AI-96-490 SBGT Inoperable (Inoperable MCC)
10/07/96	00:10	LCO #AI-96-1003 SBGT (Valve Work)
10/07/96	06:45	Resident Inspector Questioned
10/07/96	07:39	Dryer set in Pool
10/07/96	12:10	LCO AI-96-1003 Canceled
10/07/96	13:40	Began to Unlatch Separator
10/07/96	19:54	Separation being lifted
10/07/96	20:14	Separation in storage pool
10/08/96	14:00	Canceled LCO #AI-96-490

The inspector reviewed Technical Specification (TS) 3.6.5.1, Secondary Containment Integrity. When in condition 5 (refueling) the TS states that secondary containment integrity shall be maintained. TS Table 1.2, Operational; Conditions, defines condition 5, refueling, as fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

The action statements are to restore secondary containment within eight hours or, if in refueling, to suspend irradiated fuel movement, core alterations, or activities that might reduce the shutdown margin. The inspector noted that secondary containment was not maintained or restored during vessel disassembly once the issue was raised to management. Heavy loads were lifted directly above a reactor vessel fully fueled without benefit of any containment. The licensee contended that they were in compliance with the TS action statement since no fuel movement was in progress. This item will be unresolved pending further review. This will be tracked as URI 325/96-15-01, Vessel Disassembly Without Secondary Containment.

c. Conclusion

The inspector immediately recognized that vessel disassembly was being performed without secondary containment. The licensee consciously entered a TS action statement after changing operational mode condition. Once questioned about this action, the licensee continued this activity although lifts of heavy loads were made above the vessel that was fully fueled. These actions were nonconservative and contrary to the defense in depth approach of shutdown risk management.

O2 Operational Status of Facilities and Equipment

O2.1 Containment Air Dilution Walkdown

a. Inspection Scope (71707)

The inspectors performed an Engineered Safety Feature (ESF) system walkdown of the Containment Air Dilution (CAD) subsystem.

b. Observations and Findings

The inspector performed an ESF walkdown of the CAD system, a subsystem of the containment atmospheric control system. This subsystem was designed to maintain oxygen concentration below 5% to prevent combustion following a loss of coolant accident by providing nitrogen makeup to the primary containment. The subsystem also provided nitrogen as back-up pneumatic power upon loss of the normal air supply. The inspector reviewed proper valve alignment, equipment operability, area housekeeping, and proper component labeling with the system engineer. Minor discrepancies were brought to the system engineer's attention.

Inspector examination of the valve lineup contained in operating procedure 1OP-24, Containment Atmosphere Control System, associated plant drawings, and drawings contained in the Updated Final Safety Analysis Report (UFSAR) revealed numerous inconsistencies between the documents. Several valves were shown mispositioned, some were missing a valve designation, and a few were incorrectly labelled. The system engineer was notified and informed the inspector that a review of the UFSAR was being conducted as part of the licensee's ongoing UFSAR review.

c. <u>Conclusion</u>

The inspectors walked down the CAD subsystem with the system engineer. During a review of the system operation procedure, the UFSAR, and plant drawings several discrepancies were noted.

O2.2 <u>Unit 1 Loss of Shutdown Cooling</u>

a. <u>Inspection Scope (71707)</u>

The inspectors reviewed the circumstances surrounding the loss of shutdown cooling for Unit 1.

b. <u>Observations and Findings</u>

On October 11, 1996, with the unit in mode 5 for refueling, a group 8 isolation caused the Unit 1 E11-F008 shutdown cooling (SDC) suction valve to close. Closure of valve E11-F008 resulted in the 1A residual heat removal pump trip and a loss of SDC. Coolant heatup was minimized to less that 1° F, with a calculated heatup rate at 4.83°F/hr and a calculated time to boil of 26 hours. The operators entered AOP-15, Loss of Shutdown Cooling, which gave instructions on resetting the group 8 isolation. Shutdown cooling was restored after 7 minutes using the 1D residual heat removal pump.

The inspectors discussed this event with licensee management. The inspectors also reviewed the work area, applicable work requests, engineering service requests, and other associated documentation. The licensee established an event review team, and they concluded that maintenance activities on instrument rack H21-P022 caused the isolation. Instrument rack H21-P022 contains reactor pressure switch 1-B32-PS-N018B. This switch provides an isolation signal to valve E11-F008 while in mode 5 to protect the low pressure shutdown cooling system. Maintenance activities were in progress to repair corroded anchors on the instrument racks. While repairing the anchor a maintenance worker jarred the reactor pressure switch causing an isolation signal.

The adequacy of the engineering service requests and the reason the maintenance activity was scheduled while SDC was required is still under investigation. Pending completion of the licensee's investigation this item is identified as unresolved item URI 50-325/96-15-02, Loss of Shutdown Cooling.

c. Conclusion

A loss of shutdown cooling event occurred while conducting maintenance on an instrument rack containing the shutdown cooling isolation pressure switch. This activity was scheduled while shutdown cooling was in operation. The licensee's root cause and licensee event report (LER) were being prepared at the end of the inspection period.

O2.3 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.

The inspector reviewed UFSAR Section 6.2.5.2, as part of the CAD walkdown activities. Inconsistencies were noted between the UFSAR drawing and plant drawing. This issue is discussed in Section O2.1. This item will be identified as part of URI 325(324)/96-05-02.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Use of Wheeled Equipment

a. Inspection Scope (62707)

The inspector reviewed preparation for and conduct of maintenance activities associated with the current Unit 1 refueling outage. One particular area of concern was temporary storage and prestaging of outage materials and equipment. The licensee had previously identified unsecured wheeled equipment on the refueling floor and documented it in Condition Reports (CR) 96-2958 and 96-2959.

b. Observations

While conducting tours of the Unit 1 Reactor and Turbine building the inspector identified a number of unattended wheeled carts and pieces of equipment which were not properly secured. These pieces of equipment were being prestaged/stored to support the upcoming outage work and were located in different areas of the Unit 1 Reactor and Turbine buildings.

The issue of unsecured carts and wheeled equipment was a previously identified observation in NRC Inspection Report (IR) 50-325(324)/96-01. At that time, the inspector identified a number of instrument carts used in the back panel area of the control room which had wheel locks on them that were not being consistently used. The unsecured carts were a concern because they could be inadvertently bumped into or seismically shaken into sensitive panels or equipment challenging plant safety systems.

The licensee addressed this concern with Action Item Project 96-00251, which was to revise Administrative Instruction 0AI-128, Control of In Process Materials. Revision 3 was issued March 25, 1996, and added the requirements for securing equipment on wheels when used within the plant and left unattended to preclude movement that could damage installed components in the plant.

Specifically, 0AI-128, Revision 3, Section 5.4.5.3 requires that the use of equipment on wheels within the Radiation Control Area (RCA), Service Water Building, Diesel Generator Building and Turbine building is allowed provided the following conditions are met:

a. The wheels of the equipment shall be locked when it is stationary or unattended, or the equipment shall be secured in some manner to prevent movement.

b. When used within the RCA, Service Water Building, Diesel Generator Building and Turbine Building, the equipment shall be labeled with a permanent sign, stating the requirement for the wheels to be locked or the equipment secured when unattended.

c. The equipment shall be labeled with a sign similar to Attachment 2 when left unattended in these buildings, unless it is normally located in the building.

d. Carts that are simply used to transport materials in and out of these buildings, without being left unattended, are exempt from the requirements of (a), (b), and (c) above.

e. The equipment shall be moved to the appropriate storage area when the work is complete.

Contrary to the requirements noted above, on October 2 and 11, 1996, the inspector identified six different areas within the Unit 1 Reactor Building where wheeled equipment was not properly secured or labeled in accordance with AI-128. These areas included: a scaffolding cart on the Unit 1 50 foot elevation by the A train of SBGT; a flat bed wagon on the Unit 1 20 foot elevation by the railbay doors; an environmental & radiation control (E&RC) equipment cart by the Unit 1 Drywell equipment hatch; flat bed tool cart outside the Unit 1 Reactor Water Clean-up room on the 50 foot elevation; a High Efficiency Particulate Air vacuum and filter located in the Unit 1 High Pressure Coolant Injection room and a handtruck on the Unit 1 20 foot elevation near the north Core Spray pump stair well. The failure to properly secure and label these pieces of equipment in accordance with the requirements of AI-128, is identified as NCV 50-325(324)/96-15-03, Failure to Secure Wheeled Equipment. This failure constitutes a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. All of these concerns were identified to licensee personnel who corrected the problems.

c. Conclusions

The issue of temporary storage of work equipment, in particular wheeled equipment, continues to be a problem. Despite the licensee's identification early on of two instances of unsecured equipment on the refuel floor, multiple instances of unsecured equipment were still identified following increase briefings and attention to the issue by work groups and licensee management.

The inspector notes that in response to the problems associated with this issue, the licensee revised the procedure to remove the requirement for securing carts and equipment within the Turbine Building. This revision was effective October 11, 1996. Included in the new revision are spacing requirements for equipment tipping, and temporary load requirements for attaching equipment to structure or other equipment.

M1.2 Dropped Fuel Support Piece

a. Inspection Scope (62707)

While conducting routine inspection activities the inspectors were notified that control rod drive (CRD) vacuuming had been stopped due to a fuel support piece being dropped.

b. Observations and Findings

On October 16, 1996, while performing routine inspecting activities the inspectors were informed that CRD vacuuming had been suspended. The fuel support piece or casting for cell 10-47 had disengaged from the grapple and slid down the control blade coming to rest on the velocity limiter. The inspector observed as the control blade and casting were returned to a safe condition. An event review team was formed. The investigation revealed that maintenance activities had been performed earlier on the grapple but no functional test was performed to verify proper operation. The team determined that failure to verify proper latching of the casting gravity lock resulted in the dropping of the casting. In addition, poor communication between vendor personnel performing the activity and licensee management, and an inadequate procedure where contributing factors to the event. The licensee re-established expectations with the vendor concerning proper supervision, procedural adherence, and communication, and revised the procedure to include proper verification of casting latching. The inspector reviewed the immediate corrective actions. This event occurred while the reactor was defueled and was of minor consequence.

c. Conclusion

A failure to verify proper latching on a fuel support casting during CRD vacuuming operations caused the casting to disengage from the grapple and slide down the control blade. Additional causal factors for this event included poor communication between the vendor and the licensee, and an inadequate procedure. The licensee proceeded cautiously once this problem occurred. No fuel was in the vessel during this evaluation. They stopped the activity and thoroughly reviewed the problem before resuming work.

M1.3 Safe Shutdown Panels

a. Inspection Scope, Alternate Safe Shutdown Equipment, (62707)

The scope of the inspection was to inspect the safe shutdown components and panels to determine if they were adequately maintained and tested to assure operability. The inspector reviewed FSAR Chapter 7, Section 7.4, "Systems Required For Safe Shutdown", Sections 7.4.1 and 5.4.6, "Reactor Core Isolation Cooling (RCIC)", Sections 7.4.2 and 9.3.4, "Standby Liquid Control System (SLCS)", Sections 7.4.3 and 5.4.7, "RHR, Reactor Shutdown Cooling System Mode", and Section 7.4.4, "Remote Shutdown From Outside The Control Room" for a) RCIC, b) Safety/Relief Valves, c) RHR, and d) Motor Control Centers (MCC), Power Systems, and Emergency Diesel Generators for safe shutdown requirements. The (TS) requirements in Section 3/4.3.5, "Monitoring Instrumentation" and Table 3.3.5.2-1,"Remote Shutdown Monitoring Instrumentation, Surveillance Requirements" were also reviewed. Plant Operating Manual, Volume XXIII, Alternative Safe Shutdown Procedure, OASSD-02, CONTROL BUILDING, Revision 23, was examined to obtain a complete list of components and panels in the alternate safe shutdown systems. The "System Descriptions" for the above listed systems were reviewed for the identification of safe shutdown equipment and operation. Electrical drawings were reviewed to identify alternate remote shutdown control switches in MCCs and panels. The inspector conducted walkdown inspections to determine the material condition of the safe shutdown components and panels. Maintenance activities and procedures for testing, surveillance, and preventive maintenance (PM) tasks were examined to determine if safe shutdown equipment was being maintained in a satisfactory manner.

b. Observations and Findings

The licensee provided an equipment list for the "Alternate Safe Shutdown" components and panels from the Appendix R Report contained in the computerized Equipment Data Base System (EDBS). This list identified the components, systems, and the procedures for PM inspection, testing, and surveillance and the required frequency for performance. The licensee also provided the last completed PM tasks (work order) for all the safe shutdown equipment. The inspector verified that all of the safe shutdown equipment was in the PM Program.

The completed PM tasks identified when the last PM inspection, calibration, test, or surveillance was accomplished.

The inspector reviewed and verified that all 47 scheduled PM tasks (work orders) for the alternate safe shutdown components and panels were completed within the required time. Those tasks included surveillance tests performed by Operations to verify operability of the equipment. The inspector reviewed 26 procedures and verified that all the safe shutdown components and instruments were either calibrated and/or tested for operability as required. The procedures reviewed were for calibration of 18 monitoring instruments (nine per unit), testing of the safe shutdown circuit breakers in the 4160 VAC Emergency Buses E1, E2, E3, and E4, testing the alternate safe shutdown switches for the four emergency diesel generators, and testing the various Class 1E MCC in both Units 1 and 2. Each electrical panel cubicle had a local switch with an alternate safe shutdown position. This switch position allowed the circuit to be isolated from the control room and to be connected to an alternate source of electrical power for local operation during shutdown.

In both units, the inspector conducted walkdown inspections of the 1) four Class 1E 4160 VAC safety switchgear buses, 2) four emergency diesel generators, 3) all the 480 VAC Class 1E MCCs with alternate remote safe shutdown equipment, and 4) two Remote Shutdown Panels. During the walkdowns, all the 480 VAC MCCs cubicles, the 4160 VAC switchgear bus cubicles, and the emergency diesel generators panels were very clean and well maintained. The inspector verified that the control switches for the "alternate remote shutdown position were installed and in good condition. No deficiencies were identified. However, in the 4160 VAC cubicles, several "spared" cable ends were terminated with black electrical tape and several were loose. There was no operability concern since all the cables were neatly tied back and out of the way. The licensee agreed to correct the black taped spare cables.

During the walkdown inspections, the two main "Remote Shutdown Panels", one in each Unit, were found to be in an unsatisfactory condition. The outside was satisfactory. However, the interiors were in a deteriorated condition. The inspector identified a substantial amount of rust and corrosion in the internals of each panel. The internal wiring and cables were not tied together in a neat manner. The tarnish on several fuses was excessive. The internal paint was in a degraded condition and was worn off in several areas. This condition was not identified as an immediate operability concern since the surveillance tests and calibrations were performed to verify operability of the panel and instruments. However, the poor condition of the two "Remote Shutdown Panels" was identified as an Inspector Followup Item, IFI 50-325(324)/96-15-04, Material Condition Of Remote Shutdown Panels.

The licensee immediately initiated corrective action by issuing Condition Report, CR No. 96-02819 and Work Orders WRJO 96-AHCJ1 (Unit 1) and WRJO 96-AHCK1 (Unit 2) to address the condition of the "Remote Shutdown Panels".

During the review of the procedures and drawings, the inspector identified several discrepancies. Alternate Safe Shutdown Procedure BSEP-0/ASSD-02, Control Building, Revision 23 on page 124, identified Compt/Ckt, DM1 Row H3, as Equipment No. 1-SW-V10. It should have been 1-SW-V105. Electrical drawings 1-FP-05887, Sheet 3, Revision L, and 2-FR-05867, Sheet 3, Revision P, did not correctly identify safe shutdown control switches "RS1" in circuits for valves 1B21-F013E and G, and 2-B21-F013G. The inspector verified that the licensee initiated the appropriate corrective action for the procedure and two drawings.

c. Conclusion

The inspector concluded that the licensee has maintained the alternate safe shutdown equipment in a satisfactory manner, except for the interior condition of the two remote shutdown panels. The material condition of these two panels were considered poor.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Unit 1 Reactor Core Shroud Examination and Data Evaluation Activities

a. Inspection Scope (73753)

The inspector reviewed General Electric Nuclear Energy's (GENE) Shroud Examination Plan (Document No. GENE-B11-00715-1 Revision 0) and Ultrasonic Examination Procedure (Document No. UT-BRU-503V4 Revision 0) for the automated ultrasonic examination of shroud assembly welds to determine the examination scope, inspection technique and scan coverage, and equipment setup requirements. The inspector also observed GE's remote ultrasonic examination activities on the reactor vessel core shroud and evaluated ultrasonic data of discrepant areas on welds H-6A, H-6B and H-7 using GE's Smart 2000 ultrasonic system. Certification and qualification records for the two GE Level III data analyst were reviewed.

b. Observations and Findings

The inspector found GE's ultrasonic scan plan and procedure to be acceptable, approved by the licensee, and in accordance with the Boiling Water Reactor Vessel Internal Project (BWRVIP) Examination Guidelines (BWRVIP-03). The review also revealed that GE's examination techniques were in general agreement with the volumetric ultrasonic techniques described in the American Society of Mechanical Engineers (ASME) Code, Section XI (80W81) edition and addenda. However, the ASME Code does not have specific guidance for ultrasonic examination of the reactor vessel core shroud since the examination method delineated by this Code for reactor vessel internals is visual examination.

The inspector's evaluation and length sizing of cracks in the heat affected zone of welds H-6A, H-6B and H-7 was in agreement with the results obtained by GE's analysts. Certification and qualification records were satisfactory, and examination personnel were knowledgeable, highly skilled, and professional in performing their assigned duties. In addition, equipment used to scan the core shroud welds was state-of-the-art technology, obtaining maximum coverage in areas where structural interferences were common place.

c. Conclusion

The reactor vessel core shroud ultrasonic examination efforts observed by the inspector were conducted in an exemplified manner. Scan plans, procedures, personnel, and equipment were integrated to obtain the best possible inspection results. Approximately 77 percent of welds H-6A, H-6B and H-7 were examined and based on preliminary data, the maximum flawed length of weld examined for these three welds were 5.8, 10.5, and 1.6 percent respectively.

M2.2 In-Vessel Visual Inspection Of Unit 1 Reactor Vessel Internals

a. Inspection Scope (73753)

The inspector reviewed Period Test Procedure, No. OPT-90.1 Revision 17, for the remote visual examination of vessel internals to determine whether the scope of these examinations met the examination requirements of ASME Section XI (80W81) and licensee commitments. In addition, video recordings of remote visual examinations for the components examined by GE at this point in this refueling outage (B1-11R1) were re-examined by the inspector.

b. Observations and Findings

The inspector found the examination procedure to be acceptable in that, it was approved by the licensee, in accordance with ASME Section XI (80W81) and properly implemented augmented inspections required by NRC and additional concerns addressed by the industry. The inspector's review of visual examinations performed on Weld H-8 at 0° and 180°; Shroud Access Hole Covers at 0° and 180°; Core Spray Downcomer Piping Welds at 350°; Core Spray Downcomer Piping Welds at 10°; Core Spray T Box Cover Plate and Weld 11 at 90°; and Feedwater Sparger and Thermal Sleeve to Tee at 45°, concluded that GE effectively covered the components identified and documentation properly reported the condition of these components.

c. Conclusion

In-vessel visual inspections performed by GE of reactor vessel internals were conducted and documented in an acceptable manner by knowledgeable and qualified examiners. No significant finding had been identified by GE at this point in the examinations.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Inspector Followup Item No. 50-325(324)/95-19-02:

"Recurring Issues in Flow Accelerated Corrosion Program".

This item was opened to track the remaining replacement of small bore carbon steel drain piping on the moisture separator reheater with stainless steel piping material which is less susceptible to erosion/corrosion. A review of drawings for this piping and discussions with cognizant personnel revealed that the affected piping on Unit 1 will be replaced during the present outage. The remaining carbon steel piping on Unit 2 has been examined and found to be satisfactory, but will be replaced during the next Unit 2 refueling outage.

III. Engineering

E1 Conduct of Engineering

E1.1 Environmental Qualification

a. Inspection Scope (37551)

The inspectors reviewed the licensee's Environmental Qualification (EQ) program, specifically their corrective actions to respond to findings identified during Self-Assessment numbers 95-0041 and 96-0271 and issues identified in NRC IR 50-325(324)/96-14.

b. Observations and Findings

The inspectors reviewed the status of the licensee's corrective actions to resolve problems identified in the EQ program. The following issues were discussed with the licensee's EQ Task Force Manager:

- Corrections to the EDBS and corrections to the EQ equipment list.
- Updating of Qualification Data Packages (QDPs).
- Revision of the Reactor Building Environmental Report.
- Qualification of the post accident sampling system and associated components.
- Resolution of the associated circuits issue and repairs to motor control centers.
- Revisions to procedures which control the EQ program.

The discussions disclosed that the licensee's actions were on schedule to correct the program deficiencies. The inspectors re-examined the four justifications for continued operations (JCOs) issued to address equipment operability related to the EQ deficiencies. The four JCOs involved operability of the post accident sampling system, associated circuits, motor control centers, and effect of procedure deficiencies regarding thread sealants. The JCOs contained sufficient information to permit continued operation of both units pending resolution of the EQ program deficiencies.

The inspectors reviewed the licensee's program to inspect the MCCs installed in the reactor building to identify openings in the MCCs which could affect the qualification of equipment installed in the MCCs. The inspections had been completed in Unit 1 and repairs were in progress. The repairs, which included sealing of openings in the MCCs and replacement of worn gasket around doors, were scheduled to be completed prior to Unit 1 restart from the current refueling outage. The inspections were in progress in Unit 2 and the licensee was issuing work requests (WRs) to cover the repairs. The Unit 1 repairs were scheduled to be implemented after the Unit 2 repairs were completed.

c. Conclusions

The inspectors concluded that EQ JCOs contained sufficient information to permit continued operation of both units pending resolution of the EQ program deficiencies. The inspectors concluded that the licensee's progress to correct the EQ program deficiencies was satisfactory. No equipment operability issues were identified.

E2 Engineering Support of Facilities and Equipment

E2.1 Inadequate Testing of Plant Process Computer Modification

a. Inspection Scope (37551)

The inspector reviewed the events and casual factors associated with the licensee's identification that the plant process computer for Unit 2 was not applying the correct compensation for feedwater temperature during thermal power calculations. The resultant error caused the unit to operate in excess of its licensed thermal power limit of 2436 Megawatts thermal (Mwt).

b. Observations and Findings

On August 28, 1996, a reactor engineer reviewing core thermal power calculations associated with the Power Uprate Project identified that the Unit 2 Plant Process Computer (PPC) algorithm for feedwater flow did not include temperature compensation. This lack of temperature compensation caused the core thermal power calculated and displayed by the PPC to be less than the actual core thermal power. This error resulted in unit operation at power limits in excess of license limits.

Immediately following the identification of this problem, the licensee reduced reactor power to avoid continued operating in excess of the license limit of 2436 Mwt. The problem was corrected in the PPC approximately two and a half hours after identification and the unit returned to full power operation. The licensee reviewed the problem and discovered that the problem existed since unit start-up in July 1994, when the PPC was installed. Based on this information, the licensee determined that the unit operated at 2446 Mwt (101.4%) from July 5, 1994, through September 6, 1995. Additionally, from March 26, 1996, through August 28, 1996, the unit operated at 2441 Mwt (100.2%). During these time periods three notable exceptions were identified: February 26, 1995, the unit operated at 2460 Mwt (101%); April 17 through 26, 1996, the unit operated at 2492 Mwt (102.3%); and July 19 through 26, 1996, the unit operated at 2494 Mwt (102.4%). Operation in excess of the licensed thermal power limit of 2436 Mwt is identified as apparent violation EEI 324/96-15-05, Operation In Excess of License Thermal Power Limit.

In addition to operating in excess of the thermal power limit, the licensee identified that from December 10 through 20, 1995, the unit operated in excess of the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) TS thermal limit. The licensee issued LER 2-96-003, Operation in Excess of Maximum Power Level Specified in Operating License, to document these findings in accordance with the requirements of 10 CFR 50.73.

The inspector reviewed the licensee findings discussed in LER 2-96-003, in particular the licensee root cause determination. The licensee determined that the problem was the result of a database error associated with Plant Modification 90-005, Unit 2 Plant Process Computer Replacement. This modification installed the new PPC and software on Unit 2 during the 1994 refueling outage. Plant Modification 90-004, Unit 1 Plant Process Computer Replacement had previously been installed on Unit 1 during 1993. A review of the modification packages indicated that the design team, in an effort to reduces differences between the two units, copied the existing PPC database configuration from the Unit 1 PPC to the Unit 2 PPC. This strategy required the Unit 1 database to be copied, unloaded, all the Unit 1 data points renamed as Unit 2 data points, and then reinstalled on the Unit 2 database. An error in the internal representation of the data points from Unit 1 to Unit 2 resulted in the Unit 2 feedwater points being incorrectly labeled and indexed in the database. This error made it impossible for the PPC to find the compensation values in the database for feedwater temperature, and thus a value of 1 was applied for compensation. The correct compensation at normal operating temperatures of 412 degrees F is 1.003. Using the value of 1.000 at normal operating temperatures results in a 0.3% error.

Due to the size of this error, this was not identified during the reactor engineer hand calculations of thermal power performed during reactor start-up. These calculations performed in accordance with OPT-50.0, Reactor Engineering Refueling Outage Testing. The acceptance criteria between the hand calculations and the PPC calculations is two percent. The 0.3% error at normal operating temperatures was well within this error band and was not identified at the time or reactor start-up.

The licensee failed to identify this database error during the acceptance testing because these tests did not verify that process point numbering was the same in both units. Additionally, the testing did not verify that the correct relationships between process points and compensation formulas were preserved. It was determined that the design team incorrectly took credit for some of the Unit 1 PPC acceptance testing for the Unit 2 PPC modification. One of the tests which was waived would have verified the feedwater temperature compensation factor. This failure to adequately perform the required post modification acceptance testing resulted in the database error which caused the unit to be operated in excess of its licensed thermal power limit.

c. <u>Conclusions</u>

The licensee's failure to adequately test plant modification 90-005, Unit 2 Plant Process Computer Replacement, resulted in a database point identification error which caused the unit to operate at thermal power levels in excess of the Operating License limit.

E2.2 <u>Repeat Failures of Chlorine Sensors</u>

a. <u>Inspection Scope (37551)</u>

The inspectors reviewed the degradation of the capability to isolate the control room in the event of a chlorine emergency as a result of failures of five of eight chlorine sensors during a surveillance test conducted on September 19, 1996.

b <u>Observations and Findings</u>

On September 19, 1996, during the performance of Maintenance Surveillance Test, 0MST-CLDET21A, Chlorine Detection System Channel Calibration, all four of the detectors located at the control building intake plenum and one of four detectors located at the service water building failed the operability test. These detectors are part of the control building emergency air filtration system and are required to be operable to protect the personnel in the control room in the event of a chlorine emergency. Licensee practice upon determining a detector failure allows the failed detector to be replaced and calibrated satisfactorily before determining the operability of the remaining sensors. This practice does not provide for proper control room protection. The inspectors determined that all detectors in both divisions could potentially be inoperable, without isolation of the control room in accordance with the technical specifications. This surveillance was being performed quarterly in accordance with preventive maintenance route (PMR) AOVV which was established as a result of chlorine detector failures in May 1995. The 1995 failure of five of eight detectors was addressed in LER 1-95-02, Multiple Chlorine Sensors Used For Control Building Isolation Logic Were Found To Be Outside Technical Specification Tolerances During Routine Calibration.

LER 1-95-02, comments that "the design process failed to determine the sensors limitations in an environment of high velocity unfiltered air such as is present in the Control Buildings's 70' air intake plenum. Previous experience existed with chlorine sensors in HVAC ductwork applications, but not at the velocities present in the Control Building plenum." With the increased wind velocities seen as a result of Hurricane Bertha in July, the inspectors questioned why detector operability was not determined after the severe weather had passed. On July 14, 1996, during the restart Plant Nuclear Safety Committee meeting following Hurricane Bertha, the licensee decided against testing to verify detector operability due to vendor recommendation that the dust shields should have prevented moisture from affecting the sensors. In addition, following Hurricane Fran and startup of both units on September 13, the decision was made to wait to perform the testing until PMR AOVV was due on September 19.

Chlorine Detector Failure History

3/95	LER 1-95-02	5 detectors are found failed during annual detector replacement. Staggered calibration testing instituted.
4/23/95		All 4 control building detectors fail staggered testing.
5/4/95		Licensee installs weather shields to minimize detector dirt impingement.
5/8/95		2 control building detectors fail staggered testing.
9/95		Licensee changes staggered testing duration to quarterly.
10/95		All detectors pass staggered calibration testing.
1/96		Licensee fails to perform testing due to inability to obtain testing gas. Testing postponed until April 1996.

4/10-11/96	LER 1-96-05	6 detectors fail quarterly testing. All 4 detectors fail at the control building plus 2 at the service water building.
6/6/96		Violation issued in IR 50-325(324)/96-01, for failure to implement measures to identify and correct conditions adverse to quality in accordance with 10 CFR 50 Appendix B, Criteria XVI
9/19/96	CR 96-02838	5 detectors fail quarterly testing. All 4 detectors fail at the control building plus 1 at the service water building.

The September 19 failures represent the second chlorine detector failures in 6 months. The inspectors reviewed the condition report, the associated procedures, LERs, PMR AOVV and observed the affected instrumentation. In discussions with the system engineer, it was determined that the detectors were last tested during the performance of PMR AOVV on June 26, 1996. During this testing all detectors were determined operable with minor adjustments made to address detector sensitivity. During the September failures, the system engineer indicated that previous adjustments to detector sensitivity may have contributed to the failures. Despite a long history of multiple sensor failures, the licensee has decided that the chlorine detectors were considered non-safety related functions and therefore were not considered to be in the scope of the maintenance rule. On April 10-11, 1996, six out of eight detectors were found inoperable; consequently, violation 96-05-01 was issued to address the failure of the licensee to implement measures to promptly identify and correct conditions adverse to quality. The inspector concluded that the failure to prevent recurrence of the chlorine detector failures, is a repeat of violation 96-05-01 as discussed in IR 50-325(324)/96-05. This failure to establish measures to promptly identify and correct conditions adverse to quality is a violation of 10 CFR 50 Appendix B, Criteria XVI, Corrective Action and is identified as VIO 50-325(324)/96-15-06, Repeat Failure To Take Adequate Corrective Actions For Chlorine Detector Failures.

c. Conclusion

The failure of five out of eight chlorine detectors was the second failure in the last six months and the third known failure in 18 months. Despite several instances of multiple sensor failures the control building chlorine detectors have not been included in the scope of the maintenance rule. The failure to promptly identify the cause and correct the nonconformance surrounding the chlorine sensor failures was identified as a violation.

E2.3 Followup on Feedwater System Waterhammer

a. Inspection Scope (37551)

The inspectors reviewed the licensee's actions to followup on the two waterhammer events which occurred on Unit 2 during startup after Hurricanes Bertha and Fran.

b. Observations and Findings

During restart after Hurricane Bertha in July 1996 and Hurricane Fran in September 1996, a waterhammer occurred on the Unit 2 feedwater system which damaged 4A feedwater heater drain valve 2-HD-LV-75. The inspectors reviewed the licensee's actions to evaluate and correct the Unit 2 waterhammer event and to assess the effect of the waterhammer on Unit 1. The licensee's actions included repairs to the damaged valve, changes to operating procedures, and initiation of a contract with an independent consultant to analyze the feedwater system to identify causes of the transients.

The inspectors discussed the licensee's actions during the current Unit 1 refueling outage to inspect the Unit 1 piping for possible damage from a waterhammer. The piping is inaccessible during plant operation. These discussions disclosed that a detailed walkdown of the Unit 1 feedwater piping had not been performed during the current outage. Licensee engineers told the inspectors that they planned to perform a walkdown prior to restart using procedure PLP - 29, Self-Assessment for Readiness to Startup Following an Outage.

The inspectors walked down portions of the feedwater piping and examined the piping and supports for possible damage from transients. The inspectors identified a trapeze support which was mispositioned so that it was not providing support to two of three lines. The licensee initiated CR 96-03426 to document and disposition this problem. The inspectors also identified an adjacent support on one line which appeared to have been damaged. The probable cause of the mispositioned/damaged supports was a waterhammer event.

The inspectors discussed the adequacy of the licensee's inspection of the feedwater piping system with the engineering manager and other engineering personnel in consideration of past events which affected this system and waterhammer events which recently occurred at other sites on balance of plant (non-safety related) systems. The inspectors concluded that the system walkdown covered by procedure PLP - 29 may not have been adequate to identify damage to piping/supports resulting from waterhammer events. The inadequate followup to inspect Unit 1 feedwater piping for possible waterhammer damage was identified by the inspectors as a weakness in the licensee's engineering program to adequately assess and evaluate events.

c. Conclusions

The inspections of the Unit 1 feedwater system to evaluate the system for possible waterhammer damage was inadequate. This was identified as a weakness.

E2.4 Review of Unresolved Safety Issue (USI) A-46 (Units 1 & 2)

a. Inspection Scope (37551)

The inspectors reviewed the licensee's program for USI A-46, Seismic Qualifications of Equipment in Operating Plants.

b. Observations and Findings

The inspectors discussed the Brunswick USI A-46 program with licensee engineers and reviewed documents related to the program. Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors for USI A-46," dated February 19, 1987, was issued for the resolution of USI A-46. The majority of the licensees who were required to resolve USI A-46 formed the "Seismic Qualification Utility Group (SQUG)" in order to take a uniform process and standard toward the resolution of USI A-46.

On February 14, 1992, SQUG issued Revision 2 to Generic Implementation Procedure, also called GIP-2, and submitted it to the NRC for review and approval. GIP-2 established commitments and implementation guidance for its members to resolve USI A-46. On May 22, 1992, the NRC approved GIP-2 with provisions and issued Supplement 1 to GL 87-02.

In a letter to the NRC, Serial NLS-92-252, dated September 19, 1992, the licensee committed to the implementation guidance provided in GIP-2 for the resolution of USI A-46 for the Brunswick Nuclear Plant Units 1 and 2.

The licensee retained EQE Engineering consultants to complete the USI A-46 program. EQE reviewed the safe shutdown systems and equipment, performed the walkdowns, identified any outliers, and issued a final report, EQE Report No. 52213-R-002, for the project. The licensee is currently issuing work orders to correct any apparent deficiencies identified during the walkdowns. Work involving simple modifications will be addressed initially, after engineering evaluations are completed. The complicated outliers requiring more detailed analyses will be completed by either licensee engineers or a consultant in the near future. The current schedule for completing the modifications for USI A-46, is April 1998 for Unit 1 and June 1997 for Unit 2. The inspectors reviewed EQE Report Number 52213-R-002, Brunswick Nuclear Plant USI A-46 Seismic Evaluation Report. The report included the qualifications of the evaluation engineers, selection of safe shutdown systems and equipment, details for walkdowns of equipment, outliers, engineer evaluations, resolution on the outliers, and third party audits.

c. Conclusions

The inspectors concluded that the licensee's resolution for USI A-46 was performed based on commitments and the implementation guidance described in the GIP-2.

E2.5 Walkdown Inspection for USI A-46 Unit 1 Modification

a. Inspection Scope (37550, 37551)

The inspectors inspected modifications to motor control center (MCC) cabinets and control room electrical cabinets implemented to resolve deficiencies identified during the USI A-46 walkdowns.

b. Observations and Findings

The inspectors randomly selected 18 electrical cabinets for inspection to determine if the modifications were implemented in accordance with design requirements. The 18 cabinets were 1-1XA, 1-1XDA, 21A-1, 21A-2, 22B-1, 22B-2, P615 through P618, P620, P622, XU53, XU54, XU55, XU58, XU63, and XU64. Modification requirements are specified in Engineering Service Request (ESR) 9600407 and WRs documents. The inspectors identified following discrepancies:

Cabinet No.	ESR Page No.	WR No.	Discrepancies
21A-1	33	96AECBB	Insufficient thread engagements in installed bolts.
21A-2	33	96AECBB	Same as Cabinet 21A-1
22B-1	34	96AFNY4	Same as Cabinet 21A-1
22B-2	34	96AFNY4	Same as Cabinet 21A-1
P615	23	96AFNY1	A loose latch for the rack was not repaired due to a drawing error.
XU63	16	96AECB2	Incorrect size plates installed.
XU64	16	96AECB2	Same as XU63

The inspection elements included member sizes, bolt diameters and thread engagement, weld types and sizes, repairs to loose hardware, etc. On electrical cabinets 21A-1, 21A-2, 22B-1, and 22B-2, the inspectors found that the bolts installed for the modification had insufficient thread engagement. Procedure 0MMP-004 was specified in the work request as the procedure to be used for the installation. Section 9.14.6 of Procedure 0MMP-04 states that bolts shall have full thread engagement. The discrepancies identified for cabinets P615, XU63, and XU64 were due to deviations from the design drawings, or inadequate drawings. The discrepancy for cabinet P615 was due to drawing error which showed the incorrect location for the loose latch. The plate sizes to tie cabinets XU63 and XU64 were measured 3/8" X 2-5/8" X 5-5/8". The drawing showed plate sizes of either 3/8" X 2" X 4-3/4" or 3/8" X 2" X 5". The drawings were not revised to show the actual plate sizes installed for cabinets XU63 and XU64 after the plates were fabricated. Section 9.7 of CP&L Procedure EGR-NGGC-005, Engineering Service Requests, requires that changes to ESRs, such as drawings revisions, be documented and approved.

The discrepancies identified for the insufficient thread engagement on the modifications to electrical cabinets 21A-1, 21A-2, 22B-1 and 22B-2 and for the errors between the design drawings and the as-built installation for cabinets P615, XU63, and XU64 collectively constitute a Violation of 10 CFR 50 Appendix B, Criteria V which states, in part, "that activities affecting quality shall be accomplished in accordance with documented procedures or drawings." The failure to properly implement or have adequate procedures is identified as Violation 50-325/96-15-07, Engineering and Installation Problems for USI A-46 Electrical Cabinet Modifications.

Discussions with licensee engineers and review of CP&L Specification 248-107, Installation of Seismic Pipe and HVAC Supports and Miscellaneous Structural Steel, disclosed that modifications to miscellaneous structural steel are not required to be inspected by quality control personnel. There are no QC inspection requirements for miscellaneous structural steel installation in Specification No. 248-107. The inspectors also found that fit-up for penetration welds are not inspected by QC inspectors. The inspectors questioned whether the licensee's program for inspection of miscellaneous structural steel complies with the UFSAR and their commitments to NRC. Pending further review by NRC, this issue was identified to the licensee as Unresolved Item (URI) 50-325, 324/96-15-08, QC Inspection Requirements for Miscellaneous Structural Steel.

During the walkdown inspections, the inspectors found a bolt on the top of the cabinet 1-1XDA was missing. The missing bolt was not part of one of the USI A-46 modifications being inspected, but was a material condition issue. The licensee issued Trouble Ticket 035885 to install this bolt.

c. Conclusions

The inspectors concluded that the modifications for USI A-46 were adequately implemented except for the deficiencies noted in the violation. One violation was identified for the modification errors. One unresolved item was identified due to lack of inspection requirements by QC to inspect safety-related miscellaneous structural steel.

E5 Engineering Staff Knowledge and Qualification

E5.1 Training and Qualification of System Engineers

a. Inspection Scope (37551)

The inspectors reviewed the licensee's program for training and qualification of plant (system) engineering personnel.

b. Observations and Findings

The inspectors reviewed the current revisions of the following procedures which specify the requirements for qualification of plant (system) engineers:

- Training Instruction OTI-116, Engineering Support Personnel Training Program

- Training Administrative Procedure TAP-6.04, Job Specific Training Guides for Plant Engineer

These procedures establish the guidelines for training and qualification of plant (system) engineers. The licensee's program requires plant engineers to be qualified on their assigned system and qualified as a modification engineer in at least one discipline (i.e. civil, mechanical, I&C, electrical, etc.). The training program establishes training requirements for modification engineers and system engineers. The licensee was in the process of revising procedure TAP-6.04 to include newly issued and revised procedures. The inspectors reviewed the proposed procedure revision and noted that procedure ENP 33.6, EDBS Control and Revision, was not listed in Attachment 9 as required reading for EQ engineers. The licensee indicated that review of TAP-6.04 was still in progress.

The inspectors reviewed the status of the plant engineers' qualification program. This review disclosed that only 17 engineers were qualified as system engineers, while approximately 40 engineers were fully qualified modification engineers. As of the inspection date only three engineers were fully qualified plant engineers. The inspectors reviewed the schedule for completion of the qualification process for plant engineers. The schedule showed that the majority of the engineers will be fully qualified plant engineers by mid 1997. Individual training schedules have been developed for all NED engineers which document required training and the scheduled completion dates for the training.

c. Conclusions

The inspector concluded that the licensee's program for training and qualification of plant engineers meets NRC requirements. However, implementation of the program has proceeded slowly since June 1995, due to reorganization of Engineering and changes to the training program.

R1 Radiological Protection and Chemistry Controls (RP&C)

R1.1 General Comments

a. Inspection Scope (71750)

The inspectors reviewed worker radiation practices and controls to verify that selected activities were implemented in conformance with licensee policies and procedures and in compliance with regulatory requirements.

b. Observations

During routine tours of the plant, the inspectors physically challenged various doors throughout both units and found that access to high radiation areas was properly controlled. Throughout both units area postings and labelling of containers was adequate. Licensee personnel were observed during access and egress from the RCA. During RCA access, the inspector observed individuals properly wearing mandated dosimetry. Personnel were observed to properly exit the RCA using portal monitors and hand carried items were consistently monitored in the small article monitors.

The inspector reviewed licensee posting of notices to workers and verified that NRC Form 3 was posted in accordance with 10 CFR 19.11. The licensee maintains several copies of NRC Form 3 in several locations throughout the site in locked cabinets. On two of the forms reviewed by the inspector, the information was partially covered. The licensee was informed and the forms were cleared of any visual obstructions.

In addition, the inspectors reviewed radiation worker knowledge and practice, and adherence to Environmental and Radiological Control procedures. These items are discussed further in sections R1.2 and R3.1.

R1.2 Radiation Worker Knowledge and Practices

The inspector observed personnel enter and exit radiologically controlled areas. During pre-outage work activities, good radiation worker performance was seen in the proper donning and removal of protective clothing. On the refueling floor, health physics technicians (HP techs) were present and maintained adequate awareness and supervision of workers in the contaminated area. The inspector questioned several contractor workers on information present on their radiation work permit (RWP) or alarm limits for their electronic dosimetry. All workers questioned were able to identify the RWP number that they signed in on, but very few were able to remember alarm setpoints or general area dose limits. On one occasion the inspector questioned a contract worker immediately after signing on to a RWP; the worker could not remember any RWP information other than the RWP number.

The inspector discussed these findings with the licensee radiation protection staff. The licensee promptly began questioning workers in the radiation control area on their RWP information. In addition, a site wide memorandum was issued reenforcing site expectations regarding RWP knowledge. Subsequently, the inspector has observed an increased awareness by site personnel of RWP information and requirements. For example, the inspector has noticed individuals including RWP and electronic dosimetry information in work packages, with many individuals maintaining personal copies along with the pocket site outage handbooks.

R1.3 Conclusions

During pre-outage activities, the inspectors observed good radiation worker performance was seen in the proper donning and removal of protective clothing. An improvement in radiation worker knowledge was seen after the inspectors found several workers unable to relate RWP or electronic dosimeter alarm requirements.

R3 RP&C Procedures and Documentation

R3.1 Procedural Adherence to Monthly ARM Response Test

a. Inspection Scope (71750)

The inspectors reviewed adherence to a monthly ARM response test.

b. Observations and Findings

During inspection activities concerning the new fuel vault criticality monitor (IR 325(324)/96-13), the inspectors determined that E&RC procedure 0-E&RC-0358, Area Radiation Monitors Radiation Response Monthly Test, was not properly revised to reflect the correct new fuel fault criticality monitor setpoints. The purpose of this test was to record the as found ARM readings and determine rather the reading fell into the expected area radiation range. If found outside of the range a judgment would be made whether actual area readings had changed or the monitor required maintenance.

While reviewing the data for the July 1996 performance, the inspectors observed that several points did not meet the expected radiation range. The inspectors discussed this discrepancy with the licensee. The licensee indicated that the acceptance criteria was incorrect and the procedure would be revised. The August 1996 performance contained multiple points that still did not meet the expected radiation range. During the inspectors's review it was determined that the E&RC procedure had not been revised since February of 1993. No temporary change form against the procedure indicating that the procedure was incorrect or in the process of revision could be located. After additional discussions with the licensee, a revision was issued revising the procedural requirements and clarifying the acceptance criteria.

Technical Specification 6.8.1 Administrative Controls requires that written procedures shall be implemented for area radiation monitors as referenced in Appendix A of Regulatory Guide 1.33, November 1972. The inspectors identified that during the July 25 and August 14, 1996, performances of the ARM monthly response test the technicians did not properly implement the procedure to ensure that the ARM was providing proper indication of area radiation levels. The failure to properly implement the ARM monthly response test is identified as VIO 50-325(324)/96-15-09, Improper Implementation of ARM Response Procedure.

c. Conclusion

A violation was identified by the inspectors concerning improper implementation of an E&RC procedure.

R7 Quality Assurance in RP&C Activities

a. Inspection Scope (40500)

The inspector monitored activities performed by the Nuclear Assessment Section (NAS). The inspector attended an NAS debrief, B-ES-96-03, concerning In-Service Inspection and Flow Accelerated Corrosion.

b. Observations and Findings

The inspector observed that the debrief was formally conducted and the findings were good. The inspector has noted over the past several months that NAS has been more aggressive in identification of issues. Many of these issues, although independent and different, have been similar to NRC issues identifying problems in certain areas.

However, in several debriefs conducted by NAS there was noted a reluctance by key department managers to acknowledge and accept the findings. Findings were often challenged as not legitimate. This discussion was more than a mere discussion of issues to understand the problem or issue.

While the NAS findings represent a key role in the licensee's self-assessment and identification problem, the findings have not always been willingly accepted. Acknowledgement of a problem is the first step toward correcting a problem.

c. Conclusion

The licensee's NAS organization has become more aggressive in identification problems at the site. However, department managers have been at times reluctant to accept valid findings by NAS.

R8 Miscellaneous RP&C Issues

R8.1 Potentially Contaminated Steel

a. Inspection Scope (71750)

Another site had made a report on August 29, 1996, identifying the receipt of contaminated steel from a vendor.

b. In reviewing the issue, it was determined that steel from that same heat lot was also shipped to various other utilities. The Brunswick facility was identified as having received that material. A review of records indicated that 96 square feet of the quarter inch steel plate was received for use by Brunswick.

Based on this information, the licensee identified 11 different work tickets and jobs which used steel from this order. A review of the work tickets indicate that nearly all the material was used within the RCA in areas with backgrounds too high to determine if the material was contaminated. Material used outside the RCA which was accessible was surveyed and found to be < 100 Counts per minute above background. The licensee documented the results of this investigation and survey in Condition Report 96-2890. All scrap/trash material removed from the protected area was monitored, and any contaminated material detected would have been removed and disposed of properly as Radioactive Material.

c. The inspector reviewed the information provided by the licensee and noted the material was received in early 1994. No problems or discrepancies were noted.

P1 Conduct of EP Activities

a. Inspection Scope (71750)

The inspectors reviewed conformance with the 10 CFR 70.24 requirement concerning procedures governing the conduct of drills upon the sounding of a criticality area radiation monitor.

b. Observation and Findings

10 CFR 70.24 addresses monitoring of areas where licensed special nuclear material is handled, used, or stored. These requirements include procedures for the conduct of a drill for each area where new fuel is handled, used or stored to ensure that all personnel withdraw to an area of safety in the event the criticality monitor sounds. The licensee was

questioned regarding conformance with the 10 CFR 70.24(a)(3) requirement. The licensee could not locate documentation to verify that any drills had been conducted since issuance of the license. The concern was discovered by the inspector as a followup question during a review of new fuel vault criticality monitor setpoints. Additionally, the inspector discussed this requirement with NRR. Many licensees are exempt in their operating license from this requirement, but this facility was not exempted.

The inspector discussed the concern with the licensee and reviewed the immediate corrective actions. The actions for resolution included licensee review of radiation worker training and existing procedures to verify conformance with 10 CFR 70.24. The licensee generated two procedure action requests to update two procedures prior to the next shipment of new fuel being brought onsite. The updates would include the performance of a drill to familiarize personnel with the evacuation plan. Other resolution actions included the performance of a drill conducted on August 28, 1996, by three members of the refuel crew. The inspector questioned the effectiveness of the drill upon discovery that only 3, out of almost 20 personnel observed during new fuel receipt and inspection, had been present for the drill. The licensee indicated that at the time the drill was performed only 3 members of the new fuel inspection crew were working in the proximity of the new fuel vault.

10 CFR 70.24(a)(3) requires that licensees maintain procedures for those areas in which licensed special nuclear fuel is handled, used, or stored to ensure that all personnel withdraw to an area of safety upon the sounding of the alarm. These procedures are required to include the conduct of a drill. The inspector reviewed site procedures governing personnel actions in the event any ARM sounds in the plant. Among these procedures, the inspector could find no evidence that any procedure governed conduct of a drill for an inadvertent criticality in the new fuel vault. Emergency procedures reviewed dealt with actions associated with indications of increased radiation levels in different areas of secondary containment including the refuel floor. In addition, general employee training material reviewed covered exiting any area in which an ARM sounds. In the event of an inadvertent criticality, an audible alarm would sound in the control room. Annunciator procedures 1(2)APP UA-03, Annunciator Procedure for Panel UA-03 direct the operators to evaluate entrance into emergency operating procedure EOP-03-SCCP, Secondary Containment Control. Procedure 1(2)APP-UA-03 directed entrance into abnormal operating procedure, AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity and, if necessary, directs the operators to initiate a trouble tag if a circuit malfunction is suspected to be the cause of the alarm. These procedures provided reasonable assurance that procedures existed governing response to a radioactive release on the refueling floor.

The failure to have an emergency procedure which includes conduct of drills to familiarize personnel with the evacuation plan is identified as a violation of 10 CFR 70.24(a)(3). This violation is identified as NCV 50-325(324)/96-15-10, Failure To Perform Criticality Drill. This failure constitutes a violation of minor significance and is being treated as a non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

c. Conclusion

The inspector discovered that the licensee did not have an emergency procedure governing the conduct of a drill in the event of an inadvertent criticality in the new fuel vault as required by 10 CFR 70.24. The inspectors identified a Non-cited violation for the failure to include evacuation drills in the event of an inadvertent criticality in the new fuel vault.

S1 Conduct of Security and Safeguards Activities

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 November 1, 1996. Post inspection briefings were conducted on September 20, October 18, and October 25, 1996. The licensee acknowledged the findings presented.

The licensee did not identify any materials used during the inspection as proprietary information.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Aukland, Supervisor, Engineering
G. Barnes, Manager Training
E. Black, Nondestructive Examination Level III
A. Brittain, Manager Security
W. Campbell, Vice President, Brunswick Steam Electric Plant
W. Flippin, Maintenance
N. Gannon, Manager Maintenance
J. Gawron, Manager Nuclear Assessment
S. Hardy, Project Analyst, Engineering
D. Hicks, Manager Regulatory Affairs
R. Knot, Civil Engineer, Corporate Engineering
J. Landon, Engineer
W. Levis, Director Site Operations
B. Lindgren, Site Support Services
R. Lopriore, General Plant Manager
J. Lyash, Brunswick Engineering Support Section
C. Osman, Engineer
C. Pardee, Manager Operations
R. Schlichter, Manager Environmental and Radiation Control
S. Tabor, Senior Specialist, Regulatory Affairs
J. Thompson, Superintendent, Plant Operation Assessments
M. Turkal, Supervisor Licensing and Regulatory Programs
S. Vann, Superintendent, Maintenance
H. Wall, Training Supervisor
R. Williams, Manager, EQ Task Force, BESS
W. Wilton, Supervisor, Reactor Systems

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

E. Brown
R. Chou
R. Coley
M. Janus
J. Lenahan
M. Miller
C. Patterson

INSPECTION PROCEDURES USED

IP 37550: Engineering
IP 37551: Onsite Engineering
IP 40500: Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 73753: Inservice Inspection
IP 92902: Followup - Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-325/96-15-01	URI	Vessel Disassembly Without Secondary Containment (paragraph O1.1)
50-325/96-15-02	URI	Loss of Shutdown Cooling (paragraph O2.2)
50-325(324)/96-15-03	NCV	Failure to Secure Wheeled Carts (paragraph M1.1)
50-325(324)/96-15-04	IFI	Material Condition of Remote Shutdown Panels (paragraph M1.3)
50-324/95-15-05	EEI	Operation In Excess of License Thermal Power Limit (paragraph E2.1)
50-325(324)/96-15-06	VIO	Repeat Failure to Take Adequate Corrective Actions For Chlorine Detector Failures (paragraph E2.2)
50-325/96-15-07	VIO	Engineering and Installation Problems for USI A-46 Electrical Cabinet Modifications (paragraph E2.5)
50-325(324)/96-15-08	URI	QC Inspection Requirements for Miscellaneous Structural Steel (paragraph E2.5)
50-325(324)/96-15-09	VIO	Improper Implementation of ARM Response Procedure (paragraph R3.1)
50-325(324)/95-15-10	NCV	Failure to Perform Criticality Drill (paragraph P1)

Closed

50-325(324)/95-19-02	IFI	Recurring Issues in Flow Accelerated Corrosion Program (paragraph M8.1)

<u>Discussed</u>

50-325(324)/96-05-01	VIO	Chlorine Sensors Inoperable (paragraph E2.2)
1-96-05	LER	Multiple Chlorine Sensors Used for Control Building Isolation Logic Were Found to be Outside Technical Specification Tolerances During Routine Calibration (paragraph E2.2)
1-95-02	LER	Multiple Chlorine Sensors Used for Control Building Isolation Logic Were Found to be Outside Technical Specification Tolerances During Routine Calibration (paragraph E2.2)