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

Report Nos.: 50-325/93-52 and 50-324/93-52

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

Docket Nos.: 50-325 and 50-324 License Nos.: DPR-71 and DPR-62

Facility Name: Brunswick Units 1 and 2

Inspection Conducted: November 1 - December 3, 1993

Lead Inspector: 12/28/93
R. L. Prevatte, Senior Resident Inspector Date Signed

Other Inspector: P. M. Byron, Resident Inspector
Other Personnel: M. T. Janus, Resident Inspector (in training)

Approved By: 12/28/93
H. O. Christensen, Chief Date Signed
Reactor Projects Section 1A
Division of Reactor Projects

SUMMARY

Scope:

This routine safety inspection by the resident inspector involved the areas of operations, maintenance, engineering support, plant support, and other areas. Inspections were conducted on backshift (before 7:00 a.m., after 4:00 p.m.) and on deep back shift (10:00 p.m. to 6:00 a.m.) or on holidays and weekends.

Results:

Three violations were identified in operations. The first was the result of inadequate control room logs, paragraph 2.e. The second, a non-cited violation, involved the failure to accurately document auxiliary operator rounds and readings, paragraph 2.h. Continued management attention is required on clearance problems, which are further examples of Violation 93-41-02, paragraph 2.d.

A strength was identified for the results being achieved from the Three Year Plan, paragraph 6.c. Unit 1 remained in a forced outage that began on April 21, 1992. Unit 2 operated at essentially 100% power for the inspection period. This unit achieved 200 days of continuous operation on December 3. It has operated continuously without a significant event since being restarted on April 29, 1993.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

*K. Ahern, Manager - Operations Support and Work Control
R. Anderson, Vice President, Brunswick Nuclear Project
*G. Barnes, Manager - Operations, Unit 1
M. Bradley, Manager - Brunswick Project Assessment
J. Cowan, Plant Manager, Unit 1
R. Godley, Supervisor - Regulatory Compliance
*R. Grazio, Manager - Brunswick Engineering Support Section
*J. Heffley, Manager - Maintenance, Unit 2
G. Hicks, Manager - Training
*C. Hinnant, Director of Site Operations
P. Leslie, Manager - Security
*W. Levis, Manager - Regulatory Affairs
R. Lopriore, Manager - Maintenance, Unit 1
*G. Miller, Manager - Technical Support
*C. Robertson, Manager - Environmental & Radiological Control
J. Titrington, Manager - Operations, Unit 2
*C. Warren, Plant Manager - Unit 2
G. Warriner, Manager - Control and Administration
*E. Willett, Manager - Project Management

Other licensee employees contacted included construction craftsmen, engineers, technicians, operators, office personnel and security force members.

*Attended the exit interview.

Acronyms and initialisms used in the report are listed in the last paragraph.

2. Operations

a. Operational Safety Verification (71707)

The inspectors verified that Unit 1 and Unit 2 were operated in compliance with Technical Specifications and other regulatory requirements by direct observations of activities, facility tours, discussions with personnel, reviewing of records and independent verification of safety system status.

The inspectors verified that control room manning requirements of 10 CFR 50.54 and the Technical Specifications were met. Control operator, shift supervisor, clearance, STA, daily and standing instructions and jumper/bypass logs were reviewed to obtain information concerning operating trends and out of service safety systems to ensure that there were no conflicts with Technical Specification Limiting Conditions for Operations. Direct observations of control room panels and instrumentation and

recorded traces important to safety were conducted to verify operability and that operating parameters were within Technical Specification limits. The inspectors observed shift turnovers to verify that system status continuity was maintained. The inspectors also verified the status of selected control room annunciators.

Operability of a selected Engineered Safety Feature division was verified weekly by ensuring that: each accessible valve in the flow path was in its correct position; each power supply and breaker was closed for components that must activate upon initiation signal; the RHR subsystem cross-tie valve for Unit 2 was closed with the power removed from the valve operator; there was no leakage of major components; there was proper lubrication and cooling water available; and conditions did not exist which could prevent fulfillment of the system's functional requirements. Instrumentation essential to system actuation or performance was verified operable by observing on-scale indication and proper instrument valve lineup, if accessible.

b. Loss of Emergency Bus E4

On November 22, 1993, at approximately 3:36 PM, Unit 2 experienced a loss of normal power from the balance of plant bus 2C to emergency bus E4. The loss of power to emergency bus E4 resulted in the following system/component actuations: loss of RPS bus 2B and a Group 1 logic B trip (1/2 Group 1); auto start and loading of DG No. 4 to bus E4; Groups 2, 3, 6, and 10 isolations; Reactor Building ventilation isolation; start of both SBGT trains on ventilation isolation and tripping of the CRD pump 2B and SJAE 2B train. The licensee entered AOPs 2.1, 14.0, 36.1, and 37.0 in response to and recovery from this event. The cause of the loss of power to bus E4 was unknown at the time of the event.

While in the process of restoring the Reactor Water Cleanup (RWCU) system, a steam leak detection system temperature alarm was received and RWCU isolated. The licensee investigated the alarm and found secondary containment temperatures normal. The cause of the RWCU isolation was unknown. This system isolation was also a reportable event in accordance with the requirements of 10 CFR 50.72, and was included in the "Red Phone" notification to the NRC associated the above power loss and ESF actuations.

The loss of the normal power supply to bus E4 placed the licensee in a 72 hour LCO in accordance with TS 3.8.1.1 due to the loss of one offsite distribution network to the emergency buses. The LCO required an offsite circuit breaker surveillance within 2 hours and once every 12 succeeding hours and required testing of the other 3 DGs within 24 hours. The inspector verified that the licensee satisfactorily completed the required testing within the TS time limits.

Immediately following the event, the licensee initiated an investigation to determine the root cause and to develop a recovery plan. The investigation included the following actions: removal and testing of the slave and master breakers that had tripped; troubleshooting the control circuitry associated with the slave and master breakers; telephone contact with ABB (ABB had recently refurbished the breakers in question) for any related industry experience; verification of the loads on bus E4 prior to the trip, to determine if they could have been the cause determining if a grid voltage fluctuation had occurred; questioning of personnel in the area of the master and slave breaker control switches for a possible bumped switch and determining if a radio had been keyed in the vicinity of the breakers; and verification of breaker relay set points. This complex and comprehensive investigation process was unable to determine the cause of the event. All investigated areas were found to be normal and the event cause could not be determined.

Following the completion of the investigation process, the licensee began restoration activities. This included: replacement of both the master and slave breakers with newly refurbished breakers; complete meggering of the associated control circuitry; calibration of the over current relays for both the master and slave breakers; meggering of the motor driven fire pump which was operating on the E4 bus and installation of Brush recorders to monitor the breakers during the reenergization process. At approximately 6:17 PM, on November 23, 1993, the licensee closed the master/slave breakers from the bus 2C to bus E4. No abnormal indications or problems were noted during this process. The inspector observed selected portions of the above processes and will monitor the operation of this equipment for future failures that may be related to this event.

The licensee conducted a detailed investigation of the event and was unable to determine the cause. The inspector found these activities to be of sufficient depth and detail to uncover any existing deficiency.

c. Core Flow Inaccuracies

During a recent Unit 2 downpower for recirculation pump motor generator set brush replacements, the unit operated for a period of time with a single recirculation pump in operation. During this single loop operation, operators identified that core flow increased when a recirculation pump was secured. This was not anticipated and differed from other indications such as power and core plate differential pressure which decreased as expected. Flow instrumentation is designed to account for the loss of flow through the idle jet pumps through the use of subtracting circuitry which is actuated when one recirculation pump is not in operation.

Review of data from a previous downpower with single loop operation indicated the same instrument response. In light of this information, the licensee conservatively issued a Standing Instruction which required operators to manually scram the reactor following the loss of a recirculation pump. This change was implemented since core flow could not be accurately determined, and operation in the prohibited unstable regions of the power to flow map could possibly occur.

The unit operated under the standing instruction for several days until a detailed engineering review could be completed and more substantial guidance could be provided. Adverse Operating Procedure (AOP) 0-AOP-4.3, Recirculation Pump Trip, was a procedure common to both units. Following discovery and troubleshooting of this issue, the licensee developed and issued 2-AOP-4.3, Unit 2 Recirculation Pump Trip, on November 1, 1993. This new procedure was developed to incorporate the engineering determination that total core flow could be calculated based on core plate differential pressure (DP). Since Unit 1 was shut down, 0-AOP-4.3 was reissued as 1-AOP-4.3 with no modifications. A revision to 1-AOP-4.3 will be issued incorporating the same information if the problem is not corrected prior to Unit 1 restart. The inspector verified that all operating shifts had been trained on the new Unit 2 procedure.

The inspector reviewed the revised procedure and found that it provides the operators with an approved methodology to determine core flow as a function of reactor power and core plate DP. Once core plate DP is determined, a quick decision on further operation can be made based on reactor power and core plate DP. The AOP provides values for power and DP below which a manual scram is required. These values ensure sufficient core flow exists to avoid operation in Thermal Hydraulic Instability (THI) Regions A and B. If confirmation exists that operation is not within the THI Regions A and B, the AOP instructs the operator to monitor for symptoms of THI. The AOP provides a graphic representation of indicated core plate DP and power which can be used when reactor power is between 40 and 100% to determine an estimated percentage of total core flow. This graph has been placed on the RTGB as an operator aid to determine if a manual scram or immediate exit from operations in Region C is required based on the determined operating point. The AOP also provides guidance for operations at less than 40% reactor power. This consists of either reducing power to less than 25%, or increasing core flow to greater than 45%, thus avoiding operations in the THI regions.

The inspector verified that all shifts had been trained in the revised procedure and discussed the above actions with the operators who had attended the training session. The inspector found these actions to be conservative and appropriate until the unit can be shut down and the problem corrected. The licensee is still investigating the root cause of this instrument inaccuracy.

The inspector will continue to follow this investigation and corrective actions taken by the licensee to resolve this issue.

d. Inattention to Detail

The licensee experienced a series of minor events during the past two months which could be attributed to inattentiveness. None of the events were serious, but combined, could indicate a negative trend. There were seven injuries during this inspection period with four occurring the same day. The most serious injury occurred when a system engineer fell 14 feet in the drywell.

There were also four operators' errors during this period. These included a surveillance test that was not suspended when a level switch failed resulting in minor flooding in the RCIC room (Inspection Report 50-325,324/93-41). Two required surveillances had nearly missed being performed in the required LCO time and were identified just prior to the expiration of the action statement. On November 12, 1993, a reactor operator, while performing a control rod operability check, during which each rod is inserted from step 48 to step 46 and withdrawn back to step 48, became distracted, and failed to withdraw the rod to step 48. He thought he had withdrawn the rod, signed the step as completed, and completed PT 14.1. The operator ran a computer check (OD-7) at the completion of the test to verify rod positions and in his review of this data failed to detect the mispositioned control rod. The error was discovered by the nuclear engineer approximately 12 hours later. This event was documented in ACR 93-357. The operator was counseled and volunteered to discuss the event and its causes with his peers. The inspector observed this presentation and noted that he emphasized the need to reduce and control distractions.

The licensee also reinstituted verification of control rod position after movement by a qualified second individual. This verification had been previously implemented after a prior Unit 2 mispositioned control rod event (Inspection Report 50-325,324/93-16). After being in use for a few months this check had been canceled at the operators' request. Prior to the event the operators were required to use Option 1 of the OD-7 to verify the position of all rods at the completion of the PT. The licensee has concluded that the operators will use Option 2 in the future since it provides a listing of rods that are not at Step 48.

The inspector discussed his concerns regarding the apparent increase in event frequency with the licensee. ACR 93-356 was generated as a management concern regarding the number of errors by operations.

In addition to the above, several examples of inadequate clearances occurred during this time period which resulted in:

- Several control rods being inserted during a scram test while work was being performed in the reactor vessel due to an inadequate clearance. (Inspection Report 325,324/93-41 Violation 93-41-02)

- A clearance installing a gagging device on SW valve and failing to remove the gag when the clearance was canceled. (ACR 93-327)

- Four AOG valves being found with open indication while under a clearance which required them to be closed. (ACR 93-340)

- A clearance being prepared and hung on the incorrect motor heater. (ACR 93-363)

- A clearance being issued which indicated that several valves should be closed while an existing special instructions indicated they should be red tagged open. (ACR 93-370)

The inattentiveness events have not been limited to operations. There were also at least six maintenance events identified during the previous inspection period. These included a welder being issued two different types of weld rods at the same time and the uncoupling of an air actuator during maintenance with air pressure still applied.

As a result of the above, Operations management briefed all the shifts on the dangers of complacency and the need to be more attentive. The inspector attended three of these briefings and observed that the presentations covered the above events and management concerns. The rod mispositioning event of November 12 elevated the complacency concerns. On November 18, a site-wide "stand-down" was conducted for one hour. All work was stopped and sub-unit managers and above held safety meetings and discussed the complacency issue and actions which should be taken by their groups. Recent inattentiveness events were discussed to illustrate the problems. The duration of these briefings ranged from 15 minutes to one hour and no questions were asked. The inspector found the quality of the presentations to be mixed. Site management has recognized that long, relative problem-free runs and low activity outages have historically resulted in inattentiveness events. Currently, Unit 1 outage activities have dramatically decreased and Unit 2 has been on line for about 200 days. The need to take pre-emptive action to heighten staff awareness of complacency was not recognized and implemented in a timely manner by site management. They did take action to reverse

the complacency trend by having the "stand-down". The inspector found the licensee's actions to be marginally adequate.

The inspector has reviewed the above issues and considers those involving maintenance and operations to be isolated examples with minor significance. Those involving clearance issues are further examples of Violation 93-41-02 and the corrective actions for this violation should address the continued clearance problems.

e. Operator Logs

The inspector has identified weaknesses in the licensee's log keeping in previous inspection reports. Progress has been made but there are indications that additional improvement is needed. On November 12, 1993, a Unit 2 reactor operator (RO) failed to return control rod 06-23 to its proper position (step 48) during the performance of PT14.1, Control Rod Operability Check. A review of both the Unit 2 RO and SRO logs did not indicate that a mispositioned control rod event had occurred. The RO log contained an entry on November 13 at 5:00 AM which stated that PT14.1 was completed satisfactory except for four control rods which would be tested after maintenance. The next relevant entry was at 5:42 PM on the same day which stated that control rod 06-23 was withdrawn from position 46 to position 48 in accordance with the nuclear engineer's instruction. The Unit/SRO log has only one entry related to this event, the movement of control rod 06-23 from position 46 to position 48 in accordance with the nuclear engineer's instructions.

A review of both logs found that the mispositioned control rod had not been entered. OI-71, Operations Shift Logs, Revision 1, Section 5.1.1, requires that logs be written with sufficient detail to enable the reconstruction of events. This event was documented in an ACR but had not been entered in the unit log as required by OI-71. This is also contrary to the requirements of Technical Specification 6.8.1 and is a Violation, Inadequate Control Room Logs (50-324/93-52-01). The inspector is also concerned that management or supervisory reviews of the logs and this event did not identify the failure to log this event. This is another example of the lack of attention to detail previously covered in this report.

f. Engineered Safety Feature System Walkdown (71710)

On December 1, 1993, the inspector completed an electrical and valve lineup walkdown of the Unit 1 Residual Heat Removal (RHR) System. The inspector verified that the component positions were in accordance with Operating Procedure 1-OP-17, Residual Heat Removal Operating Procedure, Revision 50. During the walkdown, the inspector noted that a component identification tag was missing from a valve. This information was given to the Unit 1

Senior Control Operator (SCO) who initiated appropriate corrective action.

The inspector identified that RHR Keepfill Station Outlet Isolation Valve E11-F098 is required by 1-OP-17 to be locked in the open position. The inspector found the valve open with no locking device installed. The inspector informed the Unit 1 SCO of this finding. While investigating this issue, the SCO identified that Periodic Test 0-PT-8.2.2.b, LPCI/RHR System Operability Test Loop B, had recently been performed on the system. The valve in question, was required to be open and left in the open position by 0-PT-8.2.2.b. The SCO documented the conflicting requirements of these two procedures in Adverse Condition Report (ACR) 93-381, which was initiated to resolve this item.

While performing the electrical component lineup portion of the system walkdown, the inspector identified several circuit breakers that were not in agreement with 1-OP-17. The three breakers in question were for valve motor heater circuits and were required to be in the on position. The installed component identification tags on the breakers identified them as spares and the breakers were in the off position. The inspector questioned the Unit 1 SCO about this item. The SCO investigated the issue and identified that the breakers and their respective valves had been spared by Plant Modification, PM 84-195. The licensee is continuing to investigate this issue to determine if the PM has been closed. The SCO initiated a Procedure Action Request (PAR) and ACR 93-382 to investigate, document, and correct the issue. This investigation had not been completed at the close of the inspection period.

During the performance of this walkdown, the inspector found that the system was properly aligned and in agreement with the operating procedure except as noted above. The inspector also verified that the control room indications were in agreement by performing a control board walkdown. The inspector verified the correct revision of the operating procedure was available and in use in the control room. He found that the material condition of the system was good, all components were clean, exhibited no leakage, were properly labeled except as noted, appeared to be in good working order, and had been recently painted. The inspector noted the recently improved decontamination efforts of the pump rooms, heat exchanger rooms, pumps, heat exchangers, and other components. These actions have significantly improved system appearance and accessibility.

g. Cold Weather Preparations (71714)

The inspector reviewed licensee preparations for cold weather. Operating Instruction, OI-43, Freeze Protection and Cold Weather Bill, Revision 010, provides specific actions which must be taken

at various ambient temperature. The licensee performs preventative maintenance on plant protection and heat tracing systems in accordance with Preventative Maintenance Instruction, OPM-HT001, Revision 5. The preventative maintenance was performed on Unit 1 under PM route 1EYL046 and completed on October 11, 1993. Unit 2 cold weather protection was performed under PM route 93-NGR004 and was completed on October 15, 1993. The inspector reviewed both completed job orders. The inspector verified through a review of operator log sheets that OI-43 was implemented as required. The inspector concluded that the licensee's cold weather preparations were adequate and had been implemented in a timely manner.

h. Licensee Action on Previous Findings (92701, 92702)

(Closed) URI 325/324/93-27-04, Licensee Terminated Two Auxiliary Operators (AOs) for Falsification of Records. On June 18, 1993, two Aos were terminated for incorrectly documenting on their rounds sheets the amount of time they had spent in the Reactor Building. In addition, one of the AOs has listed, on 10 occasions, readings for times when he was not in the Reactor Building.

The licensee performed a review of AO rounds from January 4 through March 27, 1992. The review was a detailed comparison of AO rounds documentation with the Security computer logs of vital area access. The review of 1362 vital area entries resulted in 33 discrepancies. Twenty-six discrepancies involved the diesel four day fuel tank area. These discrepancies were randomly distributed throughout the period and occurred on all shifts. The licensee concluded that the procedure requirement to enter this area was not clearly understood by all AOs. No instrument readings are required to be taken in this area. The licensee was able to resolve all but three discrepancies and these all involved the same individual. The licensee concluded that falsification did not occur based primarily on the otherwise commendable performance of the individual (Inspection Report 50-325,324/93-22).

Early in 1993, the licensee reviewed a second series of data for the period of May through September, 1992, to assess the adequacy of their corrective actions from the earlier review. This indicated problems with two AOs and revealed a pattern of recorded entry times that were not in agreement with the Security computer data. The actual time the two AOs spent in the Reactor Building was less than the time the licensee considered reasonable to perform AO Reactor Building rounds. In June, 1993, the licensee took a third sample which was limited to the two AOs in question. The same pattern was observed. The licensee's investigation revealed that no Technical Specification required information has been falsified. The licensee terminated both AOs after investigation of the survey data was completed.

Procedures OI-03.3, Auxiliary Operator Daily Surveillance Report, Revision 26, and OI-03.4, Daily Check Sheets, Revision 46, are the procedures which govern AO rounds and were in effect June, 1993. OI-03.3 covers Technical Specification related tasks and OI-3.4 addresses the balance of AO tasks.

10 CFR 50.9 requires that information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commissions's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects. The AOs' entry of inaccurate Reactor Building stay times is a Violation of 10 CFR 50.9, Failure to Keep Accurate Records (50-325,324/93-52-02). This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation meet the criteria specified in Section VII.B of the enforcement policy.

3. Maintenance

a. Maintenance Observation (62703)

The inspectors observed maintenance activities, interviewed personnel, and reviewed records to verify that work was conducted in accordance with approved procedures, Technical Specifications, and applicable industry codes and standards. The inspectors also verified that: redundant components were operable; administrative controls were followed; tagouts were adequate; personnel were qualified; correct replacements parts were used; quality control hold points were adequate and observed; adequate post-maintenance testing was performed; and independent verification requirements were implemented. The inspectors independently verified that equipment was properly returned to service after maintenance.

Outstanding work requests were reviewed to ensure that the licensee gave priority to safety-related maintenance. The inspectors observed/reviewed portions of the following maintenance activities:

PM 93-038 Repair of the Unit 1 Core Support Shroud

The inspector observed the video tapes of the EDM machining and observed the installation of the support blocks. The installation sequence of the support blocks is:

1) a block is lowered into the water and positioned
2) the bottom bolts are installed and torqued
3) the upper bolts are installed
4) the dimension for the shim between the shroud and the support block is taken
5) the shim is custom machined for thickness
6) shim is installed

7) upper bolts are torqued
8) retainers installed and tack welded

On November 26 at 2:20 a.m., the first support block was placed in the reactor vessel at location 75 degrees. The second block is located at azimuth 105 degrees and the third at azimuth 45 degrees. Three blocks had been installed at the end of the inspection period. The inspector observed the installation of the first two blocks and noted that the licensee experienced some difficulty aligning these blocks.

The inspector observed the licensee experiencing difficulty in installing one of the upper bolts at location 105 degrees. The upper bolt is a tee bolt with a 2.630 inch shank and a thread diameter of 2.590 inches. The shank is machined with a square shoulder and there is a space between the shank and the first thread for the shim. The bolt weighs approximately 30 pounds. The shoulder of the shank was catching the edge of the hole in the block. This prevented the upper bolt from being inserted. It took several hours for this evolution to be successful.

On December 1, the inspector observed the licensee's attempt to place the final machine shim for location 105 degrees right on an upper bolt. They were not successful and tried to fit the shim on several other bolts with a success rate of 50%. The shim and bolt drawings were checked for tolerance stackup and the worst case had a 0.002 inch clearance. The drawing dimension for the shim throat was 2.598 (plus or minus 0.003) inches. The throat of the shim in question was 0.002 inches undersized. The inspector questioned if the shim was being receipt inspected after the final machining. The licensee's contractor, GE, informed the inspector that the shim blank was receipt inspected at the original machine shop except for the final thickness which took place at a local machine shop where the blank was machined to a final thickness. The inspector questioned the logic of not receipt inspecting the entire shim after the final machining process. GE performed a complete receipt inspection after both shims for location 45 degrees were machined and determined that the throat dimension shrunk approximately 0.005 inches during the final machining process. GE modified their procedure to require complete receipt inspection of each shim after final machining.

The inspector asked about the disposition of the undersized shim and was informed that an NCR would be written to accept-as-is. An attempt would be made to fit install the shim and scrap it if it did not fit. The inspector discussed his concerns with the Unit 1 Maintenance Manager and the Unit 1 Plant Manager who concurred with the inspector's concerns. The shim was sized to fit and liquid penetrant tested after the excess metal had been removed.

The installation of the blocks is taking longer than originally planned. The licensee is currently re-evaluating their schedule to determine the impact installation of the shroud modification and the replacement of the jet pump hold down beams may have. A revised startup schedule is expected to be issued in early December.

Overall the licensee's efforts in implementing this modification to date have been good. They have provided round-the-clock oversight and support for GE. GE has been responsive in modifying tooling and solving problems as they occur. This is a first of a kind effort and much of it is research and development. The tooling required to install the modification is new and unique.

b. Surveillance (61726)

The inspector observed performance of PT 6.1 Standby Liquid Control System Monthly Operability Test and noted that the test personnel were prebriefed by Operations prior to starting the tests, the test procedures were the correct revision, available in the field, and used, and personnel were knowledgeable on the test. A qualified AO supervised an AO under training in this test. It was performed well with satisfactory results.

c. Review of LERs (92700)

(Closed) LER 2-93-02, Inadvertent ESF Actuation During Surveillance Testing When a Test Lead Contacted Ground. This event occurred during an extended outage. An I&C technician, while conducting this test, inadvertently allowed a test lead to contact ground resulting in a blown fuse and an ESF actuation. An inspection of the equipment found no damage, the fuse was replaced, and the equipment was returned to service. The involved technician was counselled and a real-time training session was conducted for the I&C surveillance crews to make them aware of the potential for this event. These actions appear to be satisfactory for this event.

(Closed) LER 2-90-16, Reactor Scram on Turbine Stop Valve Fast Closure Caused by Reactor High Water Level When Fuses Failed in the Feedwater Control System Circuitry. The fuse blew while a functional test was in progress in the feedwater control cabinet H12-P612. All systems responded as designed and the safety significance of this event was minimal. After circuit analysis, the fuses were replaced and an event recorder was connected to monitor the circuitry. The fuses were sent to Harris E&E center for analysis. After intensive testing and evaluation by engineering and a consultant, it was determined that the fuses were not defective and had been exposed to an overcurrent condition which resulted in their failure. The licensee was unable to determine the cause of this overcurrent condition. In addition to the above, the inspector verified that the licensee

had completed the activities associated with additional operator training on the SULCV and TDRFP operation stated in the LER.

d. Licensee Action on Previous Inspection Findings (92701, 92702)

(Closed) CP&L Item B-9, Unit 1 B Control Rod Drive Water Pump. The licensee has completed testing and repair of the 1B control rod drive pump. These activities were previously identified and discussed in Inspection Report 50-325,324/93-41. The results of the vendor testing and dimensional analysis performed on the pump casing were inconclusive in determining the cause of the high discharge pressure. Based on these results, a decision was made to trim the pump impeller to obtain the desired pump performance characteristics.

The pump was returned to the site and reinstalled in the system. Testing on the reinstalled pump was performed satisfactorily on October 31, 1993. The licensee verified that pump discharge pressure was within the required operating parameters. The modification to the pump impeller has been documented in the pump technical manual for future reference. The inspector discussed results of the vendor work, vendor testing, and site testing with the system engineer. Based on these discussions and the results of the testing the inspector considers this issue closed and acceptable for Unit 1 restart.

4. Engineering Support

a. Jet Pump Holddown Beam Failures

Another BWR facility recently experienced a failure of a jet pump hold down beam. The beam fractured and severed at the interface of the forged and machined surfaces. Because of the above, the licensee decided to perform a visual inspection of the holddown beams to determine if they had a similar condition. On November 5, a modified VT-1 inspection was conducted. A review of the inspection video tapes by Level III inspectors revealed the jet pump hold down beams were not cracked at the interface. UT examination did reveal some cracking in the threaded area.

The licensee has decided to replace the Unit 1 jet pump hold-down beams during the current outage and the Unit 2 hold-down beams during the next refueling outage which is scheduled to start March 1994. The blanks for 40 beams have been ordered and the licensee is currently negotiating a fabrication schedule. The impact of the hold-down beam changeout cannot be evaluated until manufacturing schedules are finalized. The inspectors will follow this item in subsequent reports as additional information becomes available.

b. Review of LERs (92700)

(Closed) LER 2-91-019, LLRT Failure of Two Main Steam Isolation Valves. LLRTs performed during a 1991 refueling outage found leakage past the seal on main steam line "C" and "D" inboard and outboard MSIVs. This had resulted from apparent foreign material in the seat or poor seating surfaces. All of the valves were repaired satisfactorily during the outage. An analysis indicated that the 10 CFR 100 dose limits could have potentially been exceeded during a design bases event. However, since the percent open was not known, the degree to which the radiological limits could have been approached could not be determined. The licensee believes that the improved parts used in the above repair will reduce the potential for recurrence of this event.

(Closed) LER 1-90-13, Containment Atmosphere Dilution System Design Deficiencies. NRC Inspection Report 325,324/90-26 identified that the Containment Atmosphere Dilution (CAD) system did not met the design requirements for electrical separation and single failure criteria. These deficiencies had existed since the initial design and installation of the system.

The licensee developed and implemented additional procedures which verified that the system could be operated manually. These additional procedures were necessary to allow continued system operation until a permanent design change could be developed and implemented. The above LER provided detailed information concerning this issue and established the following corrective actions:

- The electrical power supplies would be changed to achieve redundant CAD system nitrogen vaporizer trains, Train A would be powered from a Division I source and Train B would be supplied from Division II. This work to resolve the single failure criteria issue was completed under PM 90-63 in January 1993.

- The system's ability to operate in accordance with the emergency operating procedures would be enhanced by modifying the Hardened Wetwell Vent path to provide an additional venting pathway capable of being powered by either Division I or Division II AC sources. This installation was completed on March 29, 1993 and September 17, 1993, under PMs 91-1 and 92-73, for Units 1 and 2 respectively.

- The electrical separation of power and control cables in cabinets XU-53 and XU-56 would be resolved by the installation of physical separation barriers between Division I and Division II cables. These barriers were installed on Unit 1 under PM 90-65, completed on October 5, 1993. The Unit 2 work is scheduled to be

completed under PM 90-66 during the March 1994 refueling outage.

The inspector reviewed the above PMs, discussed them with the appropriate engineering personnel, and performed a walkdown of selected portions of the modifications. During this review, the inspector did not identify any additional discrepancies or deviations. Based on this review and the inclusion of the remaining work in the upcoming Unit 2 refueling outage, this issue is considered closed.

c. Licensee Action on Previous Inspection Findings (92701, 92702)

(Open) IFI 93-33-02, Cracked Core Support Shroud. On November 7, the licensee contractor started electric discharge machining of the holes in the shroud to permit installation of the support blocks. The process machined the lower pair of holes and then machined the upper pair of holes for each block. Three alignment fixtures were used so that six holes could be machined simultaneously. Approximately 26 hours were required to machine each pair of holes. The material was removed as swarf. It was collected through an underwater vacuum system and filtered through a pair of 10 micron filters and a pair of 1 micron filters. Two filter banks were used to facilitate filter changeout. The licensee initially set a limit of 50R/hr for filter replacement and later reduced this to 30R/hr. Approximately 550 filters were used during machining of the 48 holes and the vacuum cleanup of the vessel. On November 24, the licensee completed the machining of the holes.

On November 18, the final shroud inspection results were completed and incorporated into Procedure O-SP-91-064/PT90.5. Engineering Evaluation Report 93-0536, Evaluation of Unit 1 Core Shroud Indications and Operability Assessment of Units 1 and 2, Revision 1, and Plant Modification 93-038, Unit 1 Core Shroud Modification, Revision 0, were submitted to NRR on November 18, 1993. NRR will review the data and issue a safety evaluation of the licensee's core support shroud analysis and repair.

On November 30, a meeting was held in Washington, D.C., to discuss the submittal. The results of this meeting will be documented in the meeting summary.

(Closed) NRC Bulletin 93-03, Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs. The licensee has completed the installation of the equipment modification for Unit 1 and completed all work which can be completed on Unit 2 during power operation. They have also successfully completed the hydrostatic pressure testing of the installed instrument lines on Unit 1. The only remaining items are the reference leg backfill sensitivity tests which will be conducted at zero and full reactor power. The testing procedure packages associated with this

testing have been approved for Unit 2 activities and are in the approval cycle for Unit 1. This testing is scheduled to be performed during unit start-up.

(Closed) P2191-07, Cracking Of Sulzer Bingham Recirculation Pump Shafts. This item was previously addressed in Inspection Report 325,324/93-41. The licensee provided the inspector a copy of the report produced by ENPROTECH detailing the results of the Structural Integrity Test performed on the shafts of the Unit 1 recirculation pumps. The report concluded that no indications of a flaw in the pump shaft structural integrity system were detected in either the A or B pump shafts. Based on these results, this issue is considered satisfactory for Unit 1 restart and is closed.

(Closed) PM 89-01 Digital Feedwater Control System. On October 29, 1993, the licensee completed all applicable portions of the preoperability testing associated with the installation of PM 89-01. This PM installed the Digital Feedwater Control System on Unit 1. Prior inspections of this activity were documented in Inspection Report 50-325,324/93-41.

Completion of the final preoperability testing was delayed due to some needed software changes. The inspector followed these activities and witnessed portions of the final loop calibration testing. All required testing with two exceptions was completed and the system was turned over to operations for start-up testing on October 29, 1993. Two test exceptions were noted for activities which are required to be performed during power ascension and following a reactor scram. The inspector has reviewed the testing results and found them to be acceptable. The inspector found no other discrepancies and considers this item closed and acceptable for Unit 1 restart.

5. Plant Support

a. Verification of Plant HP Policies and Security Areas

The inspectors verified by general observations that: the security organization was properly manned and security personnel were capable of performing their assigned functions; persons and packages were checked prior to entry into the PA; vehicles were properly authorized, searched and escorted within the PA; persons within the PA displayed photo identification badges; personnel in vital areas were authorized; effective compensatory measures were employed when required; and security's response to threats or alarms was adequate.

The inspectors also observed plant housekeeping controls, verified position of certain containment isolation valves, checked clearances and verified the operability of onsite and offsite emergency power sources.

b. Declaration of An Unusual Event

At 10:11 a.m. on November 8, the control room received a report of an injured individual in the Unit 1 drywell. The individual had fallen down a ladder from the 52 foot to the 38 foot elevation. Health Physics and onsite Emergency Medical Technicians were dispatched to the scene. Initial reports indicated that the individual had sustained a possible broken arm, back injuries, internal bleeding and numerous cuts and contusions. A call for an ambulance was made at 10:27 a.m. and the individual was transported off site at 11:11 a.m. An Unusual Event was declared in accordance with Section 12.1.1 of Plant Emergency Procedure 2.1, Initial Emergency Actions, Rev. 37. The licensee conservatively treated the individual as potentially contaminated and immediately transported him to the Dosher Hospital in Southport for medical attention.

At 11:52 a.m., the injured individual had been frisked at Dosher Hospital and determined not to be contaminated. The individual's slightly contaminated protective clothes were packaged and returned to the site. At 11:57 a.m., the one hour ENS report was made to the NRC. By 12:17 p.m., all contaminated clothing had been returned to the site, the hospital was declared clean, and the UE was terminated. The individual's injuries were numerous contusions, several cuts and a broken scapula (shoulder blade). The individual was treated and released from the hospital later that afternoon and returned to work the following day.

The licensee's response to this event was conservative and timely. The required reports were made to the NRC well within the established time limits.

c. Decontamination Activities

The licensee completed the decontamination of the RWCU phase separator tank room on November 19. This room had high radiation levels and several inches of resin encrusted on the floor for over ten years. On November 10, decontamination activities started using robots to vacuum resin from the floor. The separator tank was flushed to an oversized cask. These efforts reduced the radiation levels on the outside of the tank from 30 to 3 R/hr and levels on the floor from 800 Mr/hr to 200-500 Mr/hr.

There had been speculation that the phase separator tank had overflowed through an open manway cover on the tank top onto the floor where it had dried. An analysis of the floor resin determined that it was of a different type than that used in the RWCU system. The licensee concluded that the resin had apparently backed up through a floor drain. Video views of the top of the tank did not reveal resin residue, but did show the open manway

covers. The lack of resin on the tank supported the licensee's conclusion of the resin source. The licensee plans to reinstall the manway covers in January after radiation levels have declined.

On November 16, the inspector observed a room inspection after decontamination. The licensee used a robot equipped with mechanical pinchers and a video camera. He noted that paint on the floor was intact and exhibited little damage. All of the support plates were corroded. He discussed his observations with NED. Their initial evaluation has determined that the corrosion was acceptable. The inspector concluded that the licensee was innovative and did an effective job of decontaminating the phase separator tank room. This was an area which had been contaminated for many years and is a good example of the licensee's efforts to reduce contaminated areas.

d. Radioactive Waste Shipments

The licensee made 270 shipments of radioactive material between January 1 and November 5, 1993, to various locations. These shipments included spent fuel for storage, depleted resin, used tools, protective clothing, radioactive waste for disposal, and material that needed to be tested. The inspector reviewed the shipping records of 108 radioactive waste shipments made from July 1 to November 5, 1993, including five spent fuel shipments to the Shearon Harris Nuclear Plant. The inspector found no discrepancies in the records reviewed. The inspectors found that the trucks used to transport the waste shipments were inspected by the licensee and observed discrepancies had been documented. A discussion with Region II DRSS personnel found that the licensee had made over a thousand shipments of radioactive material since the last NRC identified violation in this area. On October 26, 1993, Region II DRSS personnel contacted a representative of South Carolina's Radiological Health Program about the licensee's radioactive waste shipments to Barnwell. This official stated that the licensee had experienced three events of record between June 1, 1987, and March 23, 1990, and none had occurred since that date. The inspector's review determined that all of these events were minor. He concluded that the licensee's performance in the area of radioactive material shipments continued to be acceptable based on a record review and discussions with Regional personnel responsible for this area.

e. Review of LERs (92700)

(Closed) LER 2-93-01, Technical Specification Requirements Exceeded When Auxiliary Reactor Building Roof Vent Radiation Monitor was Inadvertently De-energized. This event occurred because when the normal monitor was out of service a portable unit which was used for compensatory measurement lost power. This was because the unit was powered through a series of temporary extension cords. This compensatory sampling was established at

3:40 a.m. and the cord was found disconnected at 5:07 p.m. on the same date. This event would not have occurred if the extension cords had been properly labeled as required by plant procedures. The corrective actions for this event included revision of the applicable E&RC procedures to emphasize the need for proper labeling of temporary power cords and training of E&RC personnel on these and other associated procedural requirements. The inspector verified that these actions had been completed and they appear to be adequate to prevent recurrence of this event.

6. Other Areas

a. Evaluation of Licensee Self Assessment (40500)

The inspectors attended Plant Nuclear Safety Committee meetings conducted on November 4, 10, and December 2. They verified that the meetings were conducted in accordance with Technical Specification requirements regarding quorum membership, review process, frequency and personnel qualifications. Meeting minutes of the meetings not attended were reviewed to confirm that decisions and recommendations were reflected in the minutes and followup of corrective actions was completed.

There were no concerns identified relative to the PNSC meetings attended. The resolution of safety issues presented during these meetings was considered to be acceptable. The December 2 meeting did not contain agenda items but was used to inform the PNSC of the revised procedural approach now being used to perform maintenance on control rod drive hydraulic control unit (HCU) actuators. This procedure was revised as the result of the inadvertent insertion of several control rods in Unit 1 in October (Inspection Report 50-325,324/93-41). Using an informational meeting to keep the PNSC appraised of the corrective actions on an area of concern is considered to be a positive communications approach.

b. Meetings with Local Officials (94600)

The Senior Resident Inspector (SRI) conducted several informational meetings with local officials at towns near the plant to provide an update on the NRC's organization, mission, and responsibilities. He also provided a summary of the plant status, business telephone numbers of appropriate NRC contacts, and a brief resume of the NRC resident inspectors. While making arrangements for these meetings, the inspector offered to make a presentation to the town and/or county board of alderman/ commissioners, or meet with officials selected by the municipal governing body.

The SRI and his section chief met with the Southport Mayor and Aldermen on November 9 at 7:30 PM. After a short presentation, several questions involving the current status of the plant were answered.

On November 16, the SRI made a presentation to the City of Long Beach Mayor and Town Council members. No questions were asked at the conclusion of the presentation.

On November 17, the SRI held a telephone conversation with the Town Manager for Bald Head Island. This resort community is primarily comprised of absentee property owners. The manager stated that since very little had changed since our last meeting in 1991, he did not feel a meeting was necessary. The inspector then telefaxed him a copy of an updated telephone number for the appropriate NRC contacts.

On November 19, the SRI met with the Mayor of Caswell Beach. The Mayor stated that he had discussed the inspector's offer to make a presentation to the Town Council and they had decided that they were very familiar with the current plant status and the NRC mission since he also worked at the plant. Copies of the telephone numbers for appropriate NRC contacts were provided for the mayor and council members along with an open offer to provide a briefing or answer any questions that may arise in the future.

The SRI, accompanied by a Resident Inspector, met with the County Manager for Brunswick County on November 19 at the County Government Facility in Bolivia. The same information provided to other officials was also provided with additional copies of the telephone listing for all County Commissioners. The inspector also offered to meet with these officials at a later date if they so desired.

The SRI met with the New Hanover County Manager, the Chairman of the New Hanover County Commissioners, and the Emergency Preparedness Chairman on December 3 at the County Government Offices in Wilmington. He made a brief presentation using the outline previously stated, answered questions, and provided them with an updated listing for appropriate NRC contacts. An invitation was extended to meet with the remaining County Commissioners at a later date if desired.

The SRI also met with the Wilmington City Mayor and City Manager on December 3, and discussed the proposed agenda for a meeting schedule with the City Council on January 8, 1994.

The inspector has appointments for presentations to three other local towns/cities in December and January. He is also currently attempting to arrange meetings with the two remaining towns in the next few months. These items will be reported in succeeding reports as they are completed.

c. Three Year Plan (40500)

The inspector reviewed the licensee's Three Year Plan Monthly Progress Report for October, 1993. Key initiatives accomplished during October included: the implementation of a centralized training process (TY202), the definition of the scope of improvements for the Cooling Water Reliability Program (TY505), the distribution of a monthly backlog report for monitoring and trending backlog reductions (TY304), and the prioritization of plant systems for preventive maintenance evaluations (TY501). The implementation phase of the Unit 1 Digital Feedwater System, the Reactor Vessel Water Level Instrumentation Upgrade, and the Fish Diversion Screen Upgrade were completed. The design phase of the Diesel Generator Service Water Supply and Discharge Piping Replacement was also completed. Overall, 11 of 16 activities scheduled for completion were completed. Additionally, four activities were completed ahead of schedule and three previously late items were completed. The five items that were not completed were initiation tasks. Four tasks on the Cooling Water Reliability Initiative, TY 505, did not start as scheduled due to higher priority on the Unit 1 Core Shroud. These tasks are now scheduled to start in December.

The inspector selected Three Year Plan item TY 204 for a more in depth review to determine the current status on this item. TY 204 is an initiative in the area of management effectiveness and human performance. This initiative will strive to establish a culture in which world-class performance is the highest priority and continuous improvement will become a way of life. The specific objectives of this item are: to communicate the new Nuclear Generation Group (NGG) culture and business direction and to continue the focus on work environment improvements; to align the plant's organizational structure and staffing to support the NGG vision for world-class performance; and to effectively plan the plant's human resource requirements, including the use of CP&L and contractor personnel.

The first task in this involved communication of the vision, mission, values, and focus areas to all plant personnel. This has been accomplished through meetings of the site Vice President and senior management with site personnel, weekly memorandums to all personnel, and various other communication tools. The inspector has observed several of the above meetings and reviewed the other communication tools and have found this to be very effective.

The second, third, and fourth tasks involved a broad based employee survey which covered a wide spectrum of organization and human resource issues including career development, compensation and benefits, communications, customer satisfaction, work process

effectiveness, work motivation/job satisfaction, management and employee relations, performance management, total quality, safety, image, supervisory skills, empowerment, working conditions, and stress.

All regular, full time and part time employees were afforded the opportunity to participate in developing and inputting into this survey. This survey was company wide and 86% of the employees took the survey in July and August of 1993. The survey results have been compiled and presented to department managers. They are now preparing to provide feedback on the survey to employees. They have indicated that improvement plans and new initiatives will be developed as needed to improve weak areas. The licensee's actions to date indicate a strong desire to seek out employee concerns and develop initiatives for improvement.

Tasks 5, 6, 7, and 8 covered a review of the organizational structures and the implementation of needed changes to improve efficiency and effectiveness. These reviews and changes involved several studies on current staffing and projected mission and staffing meeds. These efforts culminated in the announcement of the new NGG organization and position assignments on August 26, 1993.

This new organization reduced corporate staffing and placed increased focus on each plant having responsibility for its own management, engineering, and training. This approach appears to place more emphasis on responsibility for success at the plant level. Although these changes were announced in August, 1993, the full implementation including moving engineering resources to the site, will not occur until 1994. Overall this effort appears to be on track and it has been clearly and effectively communicated to all plant personnel.

Tasks 10, 12, and 14 involve contractor utilization. On this issue, the licensee developed a plan that changed the way contractor resources will be used in the future. This plan calls for an overall reduction in contractor resources. Contractors will still be used in the areas of security, janitorial, clerical, and craft support. They will also be used for project specific tasks, with defined work scope, schedule, deliverables, and at fixed prices and will not be retained beyond the completion of the work scope. If a specific resource is needed on a continuing basis, they may make the position a CP&L employee. The licensee established a goal to significantly reduce contractor dependence in 1993. The inspector has tracked this item and found that the licensee has made significant progress toward achieving the task goal.

The remaining tasks in this initiative involve human resources and organization development. It appears that the licensee is still in the planning and development stages on these tasks. Therefore, the inspector did not attempt to evaluate these tasks.

The licensee has made significant progress on this initiative and visible results are reflected in the improved morale and attitudes of plant personnel. The reorganization of the NGG department has provided a more focused organization with stronger management that appears to have a better opportunity for success.

Licensee and corporate management have clearly shown that they are interested in input from all NGG personnel. It is more readily apparent that each person is aware of the plant mission and higher standards for success have been established. The positive results achieved to date in this area is considered a strength.

The inspector experienced difficulty in reviewing this item due to the lack of status information available in the site files for the Three Year Plan items. After discussions with the manager responsible for this item, all needed information was made available. This manager indicated that the site files would be expanded to include this information.

d. Licensee Action on Previous Inspection Findings (92700)

(Closed) IFI 325,324/90-14-01, Followup on Implementation and Effectiveness of the Integrated Action Plan (IAP). The IAP was developed by the licensee in 1989 to provide a plan of action to correct problems identified by the NRC and licensee audits. It was expected to produce rapid and sustained improvements in plant performance. While some improvements attributable to this plan did occur, it failed to achieve all desired results. A continuing pattern of equipment problems and personnel errors led to a series of special NRC inspections to assess the plant material condition and personnel performance were conducted in early 1992. These inspections were summarized in Inspection Report 325,324/92-12 and found significant equipment and material deficiencies and determined that the IAP had not achieved the desired results. The licensee in response to this report indicated that they would develop a plan to address these issues.

On June 23, 1992, the Region II Administrator requested that CP&L submit a performance improvement plan to address both the short term (pre-startup) and long term (post-startup) issues. The licensee submitted a Corporate Improvement Initiative (CII) and the Brunswick Three Year Plan on November 30, 1992, and December 15, 1992, respectively. These plans incorporated the IAP open items and broadened the scope of many completed IAP items that had not been fully effective. The NRC accepted these plans as a "get well" program for CP&L. The NRC reviewed these plans and conducted several inspections in this area to ensure that the

plans were being effectively implemented. These inspections have found that the plan is well structured and the licensee has been aggressive in the plan implementation. They have additionally achieved excellent results on a majority of the items completed to date. Additional NRC inspections and reviews are currently planned. Since the IAP has been incorporated into the Three Year Plan and is no longer being used, this item is closed.

7. Exit Interview

The inspection scope and findings were summarized on December 3, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection findings in the summary. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Description/Reference Paragraph
93-52-01	VIO: Inadequate Control Room Logs, paragraph 2.e.
93-52-02	NCV: Failure to Keep Accurate Records, paragraph 2.h.

8. Acronyms and Initialisms

ACR	Adverse Condition Report
AO	Auxiliary Operator
AOP	Adverse Operating Procedure
DG	Diesel Generator
DP	Differential Pressure
E&E	Energy and Environment
E&RC	Environmental and Radiation Control
ESF	Engineered Safety Feature
GE	General Electric Company
HCU	Hydraulic Control Unit
HP	Health Physics
I&C	Instrumentation and Control
IAP	Integrated Action Plan
IFI	Inspector Followup Item
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRT	Local Leak Rate Test
MSIV	Main Steam Isolation Valve
NCR	Non Conformance Report
NED	Nuclear Engineering Department
NGG	Nuclear Generation Group
NRC	Nuclear Regulatory Commission
PA	Protected Area
PM	Preventative Maintenance
PNSC	Plant Nuclear Safety Committee
RFP	Reactor Feed Pump
RHR	Residual Heat Removal
RWCU	Reactor Water Cleanup
SBGT	Stand By Gas Treatment

STA	Shift Technical Advisor
SW	Service Water
THI	Thermal Hydraulic Instability
TS	Technical Specification
UE	Unusual Event
URI	Unresolved Item
UT	Ultrasonic Testing