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

Report Nos. 50-321/95-21 and 50-366/95-21

Licensee: Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

Docket Nos. 50-321 and 50-366 License Nos. DPR-57 and NPF-5

Facility Name: Edwin I. Hatch Nuclear Plant Units 1 and 2

Inspection Conducted: October 2-6, and 10-13, 1995

Inspector: James L. Coley Jr. 11-7-95
James L. Coley Jr. Reactor Inspector Date Signed

Approved by: Paul Fredrickson for. 11-8-95
David M. Verrelli, Acting Chief Date Signed
Special Inspection Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of inservice inspection (observation of the 2nd ten-year interval ultrasonic examination of the Unit 2 reactor pressure vessel and the in-vessel visual inspection of the reactor pressure vessel internals), facility modifications (installation of the Unit 2 core shroud stabilizer), review of Unit 2 ASME Class 1 & 2 piping radiographic film, and review of previously open inspection findings.

Results:

In the areas inspected, one violation was identified (50-366/95-21-01, Inadequate Control of Special Processes, paragraph 2.a). No deviations were identified. The 2nd ten-year interval inservice inspection (ISI) ultrasonic examination activities for the Unit 2 reactor vessel were progressing in a timely manner.

The in-vessel visual inspection (IVVI) of the reactor vessel internals revealed that the core spray sparger experienced cracking on three additional brackets during the past operating cycle. This makes a total of seven brackets out of a population of 12 with cracks in the weld heat affected zone. The IVVI also revealed a 1/2" long indication adjacent to a fillet weld (Weld No. 17) on the core spray downcomer supply piping at 10°. Structural

ENCLOSURE 2

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evaluations were performed for each of the observed conditions and the evaluations concluded that neither condition represented a safety concern at this time.

General Electric Nuclear Energy (GENE) work activities associated with the installation of the core shroud stabilizer were proceeding slowly. The primary cause of the loss of critical path time was the unexpected failure of the EDM core actuator to hold and withdraw the plug from the partially drilled and partially EDM cut core shroud ledge at the 315° azimuth. GE engineers had to design and test an additional tool to remove the plug. The inspector's audit of the core shroud stabilizer installation activities revealed that the installation was proceeding in accordance with GE's approved fabrication procedures and documentation of the work activities were in accordance with the GENE Quality Assurance Manual.

Radiographs for twenty-three ASME Class 1 and Class 2 welds were also reviewed and found to be satisfactory.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*K. Breitenbach, Georgia Power Company (GPC) Supervisor, Engineering
*G. Brinson, GPC Supervisor, Quality Control
*O. Fraser, Southern Nuclear Operating Company (SNC) Safety Analysis & Engineering Review
*J. Garvin, SNC Nuclear Specialist
*J. Hammonds, GPC Regulatory Compliance Supervisor
*R. Healey, SNC Senior Nuclear Specialist
*A. Maze, SNC Nondestructive Examination Projects Supervisor
*T. Moore, GPC Plant Operations Assistant General Manager
*J. Payne, GPC Senior Engineer
*D. Read, GPC Assistant General Manager Support
*L. Summer, GPC General Manager
*S. Tipps, GPC Nuclear Safety and Compliance Manager
*T. Wells, SNC Senior Engineer, Nuclear Maintenance Support
*D. Willyard, GPC Senior Engineer

Other licensee employees contacted during this inspection included engineers, craftsmen, technicians, and administrative personnel.

Other Organizations

*C. Bressier, General Electric Nuclear Energy, Site Manager

U. S. Nuclear Regulatory Commission

J. Canady, Resident Inspector
E. Christnot, Resident Inspector
*B. Holbrook, Senior Resident Inspector

*Attended exit interview

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

2. Inservice Inspection (ISI) Observation of Work and Work Activities (73753) Unit 2

The inspector reviewed documents and records, and observed work activities as indicated below, to determine whether ISI was being conducted in accordance with applicable procedures, regulatory requirements, and licensee commitments. The applicable code for ISI is the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 1980 Edition with Addenda thru the Winter 1981. Unit 2 is presently in its cycle 2R12 refueling outage which is the last outage of the third period of the second 10-year inspection interval. This inspection the licensee had contracted General Electric Nuclear Energy to perform the remaining ISI for this inspection interval which consisted primarily of the 10-year ultrasonic examination of the reactor pressure vessel welds and the in-vessel visual inspection of the reactor pressure vessel internals.

a. GERIS Automated Ultrasonic Examination of Unit 2 Reactor Pressure Vessel Welds From the Vessel OD

The inspector reviewed GE's Procedure No. UT-HAT-702V0 Rev. 1, entitled: Procedure for GERIS 2000 Ultrasonic OD Examination of RPV Assembly Welds, observed data acquisition activities, observed the GERIS system 12-hour calibration checks, observed scanner operation, examined evaluations conducted by GE's analyst of recorded examination data, and reviewed personnel and equipment certifications records. Portions of the data acquisition examinations of the welds listed below were observed by the inspector:

Weld ID	Weld Type	Scan Direction
2C4 @ 270°	Circumferential	Bott. of Weld Looking Up
2C5 @ 474"	Circumferential	Bott. of Weld Looking Up
2C1B.10	Circumferential	Bott. of Weld Looking Up
2C5 @ 500"	Circumferential	Top of Weld Looking Down
2C3CL.1 @242"	Vertical	Left Side of Weld
2C3CL.2 @253"	Vertical	Left Side of Weld

During the above examinations the inspector noted the following examination discrepancies:

(1) Paragraph T-434.1.4 of Section V Article IV to the ASME Code requires that, the surface finish on the surfaces of the calibration blocks shall be representative of the surface finishes of the component. GPC's calibration blocks were not painted and their surfaces were smooth. The surface finish of the reactor pressure vessel was painted with various thicknesses of paint and some areas of the paint appeared to have sand and other particles mixed in the paint.

(2) Paragraph T-432.1 of Section V Article IV to the ASME Code requires in-part that, calibration shall include the complete ultrasonic system. The original calibration must be performed on the basic calibration block and calibration checks shall also include the entire examination system. GE however, was using two different sets of transducer cables. One set of cables were used for examination of the reactor pressure vessel and other set of cables were used for calibration. The cables used for examination of the reactor pressure vessel had never been part of the system

calibration. GE had performed a comparison check of the cables prior to the examination, but had not demonstrated the results of the alternate method to Authorized Nuclear Inspector as required by ASME Section XI, Paragraph IWA-2240.

(3) Paragraph IWA-2240 of Section XI to the ASME Code allows alternative examination methods other than those delineated in the Code to be used provided the Authorized Nuclear Inspector is satisfied that the results are demonstrated to be equivalent or superior to those of the specified method. ASME Section V Article 4 paragraph T-433.2 for calibration 12 hour checks states that: "If any point on the distance-amplitude correction (DAC) curve has decreased 20% or 2db of its amplitude, all data sheets since the last calibration or calibration check shall be marked void." GE's procedure referenced an alternative method from the ASME Code DAC recheck calibration, which would have allowed a decrease in sound amplitude of 50% or 6 db based on features which GE considers improvements in their ultrasonic system. The alternate method GE proposed however, had not been demonstrated to the Authorized Code Inspector to be equivalent or superior to the Code approved method as required by ASME Section XI, paragraph IWA-2240.

However, the inspector was concern that an error band of 6 db or 50% of the signal amplitude could be indicative of transducer or cable connector problems which may not be compensated for by adding gain to the system. The licensee responded to the inspector's concern by auditing all of GE's calibration re-checks. This review revealed that all of the data taken on the Unit 2 reactor pressure vessel was well within the Code allowed 2 db error band. As a result of their findings Southern Nuclear Company (SNC) had GE's examination procedure revised to reflect the Code allowed 2 db loss of gain requirement.

The above three examples of violations to ASME Code requirements should have been resolved by the licensee during their review and approval of GE's procedure or during audits of GE's initial calibrations. Therefore, these violations of Code requirements are identified as Violation No. 50-366/95-21-01, Inadequate Control Of Special Processes.

During the examination process the inspector also noted that GE was not performing a near surface examination (70° scan or scanning with an OD creeper transducer) to insure that the maximum amount of weld metal and base material would be scanned. GE stated that, in lieu of performing the near surface scan they would report the first 1/4" of metal thickness starting from the outside surface of the reactor vessel as a scan limitation area due to the transducer near field effects. Since, indications affected by a transducer's near field can be missed or sized incorrectly regardless of the transducer angle used. The inspector reviewed two GE documents which supported GE's position that the 1/4" material limitation was sufficient depth in the metal to allow indications to be detected and sized correctly. The two documents reviewed were (1) GE's ASME Code Sizing and Detection Capabilities of the GERIS 2000 OD Ultrasonic Imaging System, and (2) General Electric Company Report for the Detection and Sizing Capability Test for Regulatory Guide 1.150, Revision 1.

In addition to the above examination activities the inspector observed GE analyst evaluate data for portions of the following welds:

<u>Weld Identification</u>	<u>Direction of Scan</u>
2C3BL.4 Vertical Weld	Left Side of Weld
2C5.B.2 Horizonal Weld	Bott. of Weld Looking Up
2C5 @ 500" Horizonal Weld	Top of Weld Looking Down

The inspector also reviewed examiner certification records for all nondestructive examination processes and reviewed the GERIS ultrasonic equipment calibration and certification records.

b. In-Vessel Visual Examination of the Reactor Vessel Internals

The inspector observed GE's Level III Visual Examiner review and evaluate video tapes of the visual examination of the reactor vessel internals. This inspection included the required core shroud visual examinations delineated by design and fabrication documents for installation of the core shroud stabilizers. Visual inspections were performed in accordance with Southern Nuclear Company's Visual Examination Procedure No. VTH-750, Revision 6. The following components were reviewed by the inspector:

<u>Components Examined</u>

90° to 10° on Core Spray Supply Piping
Core Spray Downcomer at 10°
90° to 170° on Core Spray Supply Piping
Core Spray Downcomer at 170°
Core Spray Sparger Brackets at 30°, 120°, and 150°
24" of Vertical Weld V-6 on Bottom Side of Weld H-4 from the ID
24" of Vertical Weld V-6 on Bottom Side of Weld H-4 from the OD
H-9 Weld at Shroud Ledge Between Jet Pumps 12 & 13 (@225°)
24" of Vertical Weld V-5 on Bottom Side of Weld H-4 from the ID
24" of Vertical Weld V-5 on Bottom Side of Weld H-4 from the OD
H-9 Weld at Shroud Ledge Between Jet Pumps 2 & 3 (@ 45°)
6" of Vertical Weld V-3 on Top Side of Weld H-4 ID
6" of Vertical Weld V-3 on Top Side of Weld H-4 OD
H-9 Weld at Shroud Ledge Between Jet Pumps 8 & 9 (@135°)
6" of Vertical Weld V-4 on Top Side of Weld H-4 ID

6" of Vertical Weld V-4 on Top Side of Weld H-4 OD
H-9 Weld at Shroud Ledge Between Jet Pumps 18 & 19 (@315°)

The above examinations revealed two new areas of concern:

(1) A crack-like indication 0.5 inch in length was identified in the heat affected zone of fillet weld No. 17 on the 10° Core Spray Downcomer. A structural margin evaluation was performed by GE (Letter G-GPC-5-120, Transmittal of GE Report: GENE-523-A110-1095 dated October 12, 1995) which concluded that Hatch Unit 2 can safely operate with up to a 263° circumferential crack and that no operational changes or restrictions are required at this time.

(2) The IVVI also revealed that, three additional core spray sparger brackets had experienced cracking in the weld heat affected zone (HAZ) during the past operating cycle. A total of seven brackets from a population of twelve now have experienced cracking in the weld HAZ. The inspector requested to see a structural evaluation of this condition. On October 23, 1995, Southern Nuclear Operating Company telecopied the inspector a copy of GENE-523-A115-1095, "Structural Evaluation of the Hatch 2 Core Spray Sparger Bracket Indications." In this report GE evaluated the indications as indications in the core shroud since the cracking was not observed in the bracket weld, but in the HAZ on the shroud side of the weld. This evaluation indicated that the observed indications were well below the allowable flaw sizes, Thus, the structural integrity of the core shroud was assured for the next operation cycle. The report also considered the possibility of bracket failure in combination and concluded that it was not a safety concern.

Within the areas examined, No violations except the violation identified in paragraph 2.a above were identified. No deviations were identified.

3. Facility Modifications - Installation of the Unit 2 Core Shroud Stabilizer (37701)

On July 25, 1994, the Nuclear Regulatory Commission issued Generic Letter (GL) 94-03 to address the potential for cracking in core shrouds and to request licensees to take certain actions. By letter dated August 24, 1994, Georgia Power Company (GPC) responded to GL 94-03. GPC indicated their plans to install a permanent preemptive repair of the shroud in both Hatch units. A permanent repair was subsequently installed on Unit 1 during the fall 1994 refueling outage. The repair encompassed the entire set of circumferential welds in the core shroud and involved the installation of four tie-rod assemblies in the annulus region around the core shroud. GPC submitted the details of the planned repair for the Unit 2 core shroud to NRC on July 3, 1995. Supplemental information in response to the NRC staff's request for additional information dated August 17, 1995 was provided by GPC on August 25, 1995.

The function of the Unit 2 core shroud repair is to structurally replace all circumferential welds from the H1 weld at the top of the core shroud to the H8 weld at the bottom of the core shroud. The Unit 2 core shroud contains a total of nine circumferential girth welds. These welds are labeled H1 through H5, H6A, H6B, H7 and H8. The only significant cracking of BWR core shrouds has been associated with these welds.

The core shroud repair is designed to restrain the core shroud head, the top guide support ring, and the core support plate, and to limit upward displacement of the core shroud to acceptable levels during normal, upset and postulated accident conditions. The modification has been designed as an alternative to the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code pursuant to 10 CFR 50.55a(a)(3)(i). The repair design provides structural integrity for, and takes the place of, all circumferential welds subject to cracking in the core shroud. The repair is designed for the remaining life of the plant and any possible extension beyond the current operating license. The repair is also designed to accommodate uprated power conditions corresponding to 105% rated power (2558 MWt).

The core shroud repair design consist of four tie-rod stabilizer assemblies installed 90° apart in the core shroud/reactor vessel annulus. Each assembly consists of a tie-rod, an upper bracket, upper stabilizers, a lower spring, a middle support assembly, and collet mount connected by a solid rod. The assemblies, which are designed and fabricated as safety-related components, are used to maintain the alignment of the core shroud assuming all circumferential welds are cracked 360° through-wall.

At the top of the shroud, each stabilizer assembly fits into a slot that is machined partially into the top shroud flange just below the shroud head. The stabilizer upper bracket is inserted into this slot and extends downward to below weld H3 providing support for the upper stabilizer. The tie-rod passes through a hole in the upper bracket and is held against the upper bracket with a nut. The tie-rod extends downward approximately 151 inches to the lower spring. At the middle of the tie-rod, a support is installed between the tie-rod and the RPV to minimize the potential for vibration, and provide a limit to the potential motion of the shroud between welds H4 and H5. The bottom of the tie-rod threads into the lower spring which has a clevis at its bottom that is attached to a collet connector with a pin. The collet connects to the shroud support through a hole that is machined in the shroud support.

The tie-rod stabilizer assemblies were designed using the ASME Code Section III, 1989 Edition, subsections NB and NG as a guide. The original ASME Code Section III (1968 Edition and addenda through Summer 1970) for the design and construction of the RPV did not contain design requirements for core support structures. The additional loads placed on the RPV by the stabilizer assemblies were evaluated to the original design code. The shroud stabilizer repair/replacement was performed in accordance with Article IWA-7000 of ASME XI, 1980 Edition with Addenda through Winter 1981. Although, the core shroud was not initially supplied as a ASME Code component, Section XI requires ISI of core support structures. This required replacement is also different than most replacements, in that, the stabilizers are not a direct replacement. Instead the structural functions of the shroud and shroud support horizontal welds are replaced by new components.

The inspector reviewed design documents, the maintenance work order (MWO), GE's Quality Assurance Manual, fabrication procedures, work travelers, and observed work activities to determine whether the core shroud stabilizer, was being installed in accordance with licensee approved documents, regulatory requirements, licensee commitments, and applicable codes.

The following documents were reviewed by the inspector:

- Georgia Power Edwin I. Hatch Nuclear Plant-Unit 2 (Docket 50-366 [Letter S/N HL-4877]), Core Shroud Stabilizer Design Submittal

- NRC Letter dated 8-17-95 Request for Additional Information Regarding Core Shroud Modification for Hatch Nuclear Plant, Unit 2 (TAC No.92783)

- GENE Specification 25A5718, Rev. 0, Shroud Repair Hardware Design Specification - Hatch Unit 2, May 1995 (Attachment 2)

- GENE Specification 25A5717, Rev. 1, Shroud Stabilizers Code Design Specification - Hatch Unit 2, June 1995 (Attachment 3)

- NRC Safety Evaluation for Core Shroud Stabilizer Design - Edwin I. Hatch Nuclear Plant Unit 2 (TAC No. M92783)

- GENE Shroud Modification Procedure No. HA-SM/SHD-001, Rev. 1, Installation of Tooling and Hardware, (with Field Revision Request [FRR] No. 1D8H2-FRR-005 dated 10-09-95)

- GENE Shroud Modification Procedure No.HA-SM/EDM-001, Rev. 0, Check-out and Set-up of Electrical Discharge Machining (EDM) Equipment, (with FRR No. 1D8H-FRR-001 and FRR No. 1D8H2-FRR-003)

- GENE Shroud Modification Procedure No. HA-SM/EDM-002, Rev. 0, Operating Procedure for the Electrical Discharge Machining (EDM) Equipment, (with FRR No. 1D8H2-FRR-002 and FRR No. 1D8H2-004)

- GENE Quality Assurance Manual for Modification, Maintenance, Repair, or Replacement Projects - QAM-001, Rev. 4

- GENE Field Disposition Instruction (FDI) No. HT2-0121-12900, Rev. 3, Shroud Repair Program, (MPL-B11-A001)

- Shroud Repair Maintenance Work Order No. MWO 2-95-2853
- Design Change Report No. 94-052
- Work Process Sheet No. 2H94-052-M001, Section XI Repair/Replacement Evaluation/Documentation Report
- GENE Shroud Modification Procedure No. HA-SM/UT-002, Rev. 0, Ultrasonic Thickness Procedure
- GENE Shroud Modification Liquid Penetrant Examination Procedure No. HA-SM/PT-001, Rev. 1
- GENE Traveler No. HA-EDM/Test Burn, Rev. 0, EDM Equipment Setup and Test Burn
- GENE Traveler No. HA-Install-45, Rev. 1, which included the following:
    - Special Process Control Sheet (SPCS) No. EDM - 45-01, Collet Bolt Hole 1" Deep Burn
    - SPCS No. EDM - 45-02, Collet Bolt Hole 9" Deep Burn
    - SPCS No. EDM - 45-03, Collet Bolt Hole Spot Face Burn
    - SPCS No. EDM - 45-04, Shroud Flange Burn
    - SPCS No. EDM - 45-05, Jet Pump Restrainer Bracket Ear Burn
- GENE Traveler No. HA-Install-135, Rev. 1 (SPCS's similar to above)
- GENE Traveler No. HA-Install-225, Rev. 1 (SPCS's similar to above)
- GENE Traveler No. HA-Install-315, Rev. 1 (SPCS's similar to above)
- GENE Traveler No. HA-Measure-45, Rev. 1
- GENE Traveler No. HA-Measure-135, Rev. 1
- GENE Traveler No. HA-Measure-225, Rev. 1
- GENE Traveler No. HA-Measure-315, Rev. 1
- GENE Traveler No. HA-Wedge-45, Rev. 1
- GENE Traveler No. HA-Wedge-135, Rev. 0
- GENE Traveler No. HA-Wedge-225, Rev. 1
- GENE Traveler No. HA-Wedge-315, Rev. 1

The inspector's review of the above documents revealed that they were adequate for the work in-process. The inspector also verified that fabrication instructions were documented in accordance with GE's Quality Assurance Manual 001, Rev. 4, and that in-vessel visual inspections of the shroud had been conducted satisfactory. In addition, the load test for the new plug gripper which GE had designed to remove the drilled plug at the 315° azimuth location was verified satisfactory.

Within the areas examined, no violation or deviation was identified.

4. Review of Radiographic Film for ASME Class 1 & 2 Piping Welds (57090) Unit 2

The inspector examined the radiographic film and associated records for the welds listed below to determine whether they had been processed, examined, evaluated, disposition, and were being maintained in accordance with the licensee's approved radiographic procedure and the 1986 Edition of Section V to the ASME B&PV Code. Radiographic film for the following Welds were reviewed.

M W O	Weld No	Class	Pipe Size
2-95-1390	2E11FW-8	2	3" Dia. x .265" Thk.
2-95-1390	2E11FW-1	2	3" Dia. x .265" Thk.
2-95-1390	2E11FW-6	2	3" Dia. x .265" Thk.
2-95-1390	2E11FW-7	2	3" Dia. x .265" Thk.
2-95-2632	2E51FW-1	2	4.5" Dia. x .400" Thk.
2-95-2632	2E51FW-3	2	4" Dia. x .400" Thk.
2-95-2632	2E51FW-4	2	4" Dia. x .400" Thk.
2-95-2632	2E51FW-2	2	4" Dia. x .400" Thk.
2-95-2522	2E21FW-1	2	3.5" Dia. x .265" Thk.
2-95-2301	2E11FW-3	2	6" Dia. x .595" Thk.
2-95-2300	2E11FW-4	2	6" Dia. x .600" Thk.
2-95-2595	2P73FW-3	1	4.5" Dia. x .425" Thk.
2-95-2595	2P73FW-1	1	2.375" Dia. x .268" Thk.
2-95-2595	2P73FW-2	1	4.5" Dia. x .430" Thk.
2-95-2595	2P73FW-5	1	4.5" Dia. x .475" Thk.
2-95-2595	2P73FW-6	1	2.375" Dia. x .275" Thk.
2-95-2522	2E21FW-2	2	3.5" Dia. x .265" Thk.
2-95-2522	2E21FW-3	2	3.5" Dia. x .265" Thk.
2-95-2522	2E21FW-4	2	3.5" Dia. x .265" Thk.
2-95-2522	2E21FW-7	2	3.5" Dia. x .265" Thk.
2-95-1390	2E11FW-3	2	3.5" Dia. x .265" Thk.
2-95-1390	2E11FW-5	2	3.5" Dia. x .265" Thk.
2-95-2522	2E21FW-5	2	3.5" Dia. x .265" Thk.
2-95-2522	2E21FW-6	2	3.5" Dia. x .265" Thk.

The inspector's review of the above radiographs revealed that good radiographic quality had been achieved. The film had been interpreted, evaluated, and disposition correctly.

Within the areas examined, no violation or deviation was identified.

5. Licensee Actions on Previous Inspection Findings (92701 & 92702)

(Closed) Unresolved Item No. 50-321,366/92-025-02, Improper Certification of Welding Material

During a review of welding material certification records, an inspector noted that, for two heats (69A315 and 31375) of ER70S-X welding wire, the certified material test reports (CMIRs) did not provide tensile and impact test results in the heat treated condition. Tensile and impact property results were provided only in the "as welded" condition. The licensee conducted an audit of all welding materials on site and found that these were the only two heats of welding material that were not properly tested. The licensee's audit also revealed that neither of the two heats of welding wire had been used in heat treated applications which would have required tensile and impact tests in the heat treated condition. The cause of this discrepancy was found to be due to personnel error on the part of the quality control (QC) receiving inspector. Additional training was given to all QC inspectors to ensure that they were fully cognizant with post weld heat treatment requirements. Since neither heat of welding material had been used in a heat treated applications and no other example of inadequate inspection could be found this item is considered closed.

(Closed) Inspector Followup Item No. 50-321,366/94-025-02, ISI Procedure Weaknesses

This item involved two procedural problems, the first problem was that Southern Nuclear Company's (SNC) Procedure No. UT-HAT-212VO recommended using an OD creeping wave transducer on pipe welds which had weld overlay weld repairs on them, but did not require it. Therefore, the OD creeping wave transducer was not being used. The second problem was the procedure did not have or reference instructions for recording indications. SNC's audit of the inspector's findings revealed other procedures with similar weaknesses. The licensee's ultrasonic test (UT) procedures for the examination of weld overlay repairs now require that OD creeping wave transducer be used. In addition, all UT procedures have been revised to reference SNC Procedure No. AUX-H-301 for recording indications.

(Open) Unresolved Item No. 50-321,366/94-025-01, Review of Previous Ultrasonic Examination Data

An inspector's examination of UT overlay weld data for Weld 1B31-1RC-12BR-B-3 revealed inconsistencies in the data recorded for this weld during different refueling outages. The differences had not been documented or dispositioned. As a result of this finding SNC re-reviewed all previous overlay UT data to determine whether this was an isolated instance or if any other overlay welds had significant differences in examination results. SNC's review revealed that one other weld on Unit 1 (No. 1B31-1RC-22AM-1) had differences in recorded data between outages. The differences noted were that until 1994, examination results showed two indications at or near the examination volume, but not at the overlay interface. In 1994 no indications were recorded.

Corrective actions taken by SNC has consisted of revising UT examination procedures to require the evaluator to compare all examinations to previous examination data and document and disposition all significant differences in the results. The licensee however, has not re-examined Weld No. 1B31-1RC-22AM-1 because Unit 1 has been operating since the finding was identified. This item will remain open until 1B31-1RC-22AM-1 is re-examined.

(Closed) Unresolved Item No. 50-321/92-021-01, Inadequate Control of Primary System Pressure and Temperature

On August 27, 1992, during recovery from the scram transient and as a result of a lack of forced circulation, the reactor coolant became thermally stratified in the vessel. That is, relatively cold makeup water settled to the bottom head region of the reactor pressure vessel while the upper region remained at saturation temperature. The operators noted that the vessel bottom head metal temperature was less than that allowed by Unit 1 Technical Specifications figure 3.6-2. However, due to a procedure error, the operators were led to monitor the vessel metal temperature at a point above the bottom head region. An orderly cooldown and vessel pressure reduction was commenced. On August 28, 1992, due to the pressure reduction, the reactor vessel metal temperature was back within the pressure/temperature limits of figure 3.6-2. With the reactor pressure at approximately 100 psig and the water in the reactor still stratified, one residual heat removal (RHR) system pump was started in the shutdown cooling (SDC) mode to continue the cooldown which began earlier. When the system was placed in the SDC mode, relatively hot coolant was transferred into vessel bottom head area resulting in the vessel bottom head drain temperature increasing at a rate greater than the Unit 1 Technical Specification limit of 100 degrees Fahrenheit per hour. The licensee attributed the cause of these events to stratification of the reactor coolant within the reactor vessel. A contributing factor was a less than adequate procedure. Immediate corrective actions included revising procedures and training.

GE subsequently reviewed each circumstance involved in this event and found them bound by previous analyses. However, GE also performed an event specific finite element analysis to demonstrate compliance with the requirements of 10CFR50 Appendix G, which references the ASME Code, Section XI, Appendix G. In the analysis, the temperature-time and pressure-time traces were applied to a finite element model of the bottom head to determine the pressure and thermal stresses associated with the event. The results show that the actual transient is acceptable per the Appendix G requirements.

During the inspector's examination of this item the following documents were reviewed: GPC LER No. 0-92-23 dated September 25, 1992 and GPC Updated LER No. 1-92-23 dated December 23, 1992; GENE Document No. 523-160-1292/DRF-00551, Rev. 0, entitled: Appendix G Analysis of Bottom Head

Cooldown Transient Hatch, Unit 1, dated December 1992; GENE Document No. NEDC-32319P Class 3, 5/94, entitled: Coolant Stratification Mitigation Evaluations for Edwin I. Hatch Nuclear Plant (Units 1 & 2); GPC Procedure No. 3450-ELL-010-1S, Rev. 21, entitled: Residual Heat Removal System (Unit 1) and GPC Procedure No. 3450-ELL-010-2S, Rev. 18 (Unit 2).

Based on the above review, the inspector concluded that assumptions used by GE in their analysis appear conservative and therefore, satisfactory. In addition, revisions made by the licensee to the above procedures for the Residual Heat Removal System should insure that plant heatup and cooldowns are properly controlled. This item is considered closed.

6. Exit Interview

The inspection scope and results were summarized on October 13, 1995, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results listed below. Although reviewed during this inspection, proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

(Open) Violation No. 50-366/95-21-01, Inadequate Control of Special Processes, paragraph 2.a

(Closed) Unresolved Item Nos. 50-321, 50-366/92-025-02, Improper Certification of Welding Material, paragraph 5

(Closed) Inspector Followup Item Nos. 50-321, 50-366/94-025-02, ISI Procedure Weaknesses, paragraph 5

(Open) Unresolved Item Nos. 50-321, 50-366/94-025-01, Review of Previous Ultrasonic Data, paragraph 5

(Closed) Unresolved Item No. 50-321/92-21-01, Inadequate Control of Primary System Pressure and Temperature, paragraph 5

7. Acronyms and Initialisms

ASME	-	American Society of Mechanical Engineers
B&PV	-	Boiler and Pressure Vessel
BWR	-	Boiling Water Reactor
DAC	-	Distance Amplitude Curve
DB	-	Decibel
DCR	-	Design Change Report
EDM	-	Electrical Discharge Machine
ERT	-	Event Response Team
F	-	Fahrenheit
FRR	-	Field Revision Request
GE	-	General Electric
GENE	-	General Electric Nuclear Energy
GERIS	-	General Electric Remote Inspection System
GL	-	Generic Letter

GPC	-	Georgia Power Company
HAZ	-	Heat Affected Zone
ID	-	Inside Diameter
ISI	-	Inservice Inspection
LER	-	Licensee Event Report
MWO	-	Maintenance Work Order
MWT	-	Mega-Watt Thermal
No.	-	Number
NRC	-	Nuclear Regulatory Commission
OD	-	Outside Diameter
PT	-	Penetrant Testing
Rev.	-	Revision
RHR	-	Residual Heat Removal
RICSIL		Rapid Information Communication Service Information Letter
RPV	-	Reactor Pressure Vessel
SDC	-	Shutdown Cooling
SNC	-	Southern Nuclear Operating Company
SPCS	-	Special Process Control Sheet
UT	-	Ultrasonic Testing