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

Report Nos.: 50-348/92-11 and 50-364/92-11

Licensee: Southern Nuclear Operating
Company, Inc.
600 North 18th Street
Birmingham, AL 35291-0400

Docket Nos.: 50-348 and 50-364 License Nos.: NPF-2 and NPF-8

Facility Name: Farley 1 and 2

Inspection Conducted: March 30 - A, and April 13-17, 1992

Inspector: [signature] 5/18/92
for N. Economos Date Signed

Approved by: [signature] 5/18/92
J. J. Blake, Chief Date Signed
Materials and Processes Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, unannounced inspection was conducted in order to observe ongoing, ten (10) year inservice inspection (ISI) of Unit 2, reactor pressure vessel (RPV) welds and associated nozzles; Eddy Current (EC), examination of Steam Generator (S/G) tubes, including authorized repairs by laser welding of sleeves and plugging as appropriate. The design change package for the elimination of the RTD bypass system was reviewed and related welds were inspected for code and regulatory compliance as applicable. Results of Feedwater nozzle examinations were reviewed and compared with previous examination data which revealed no changes had taken place.

Results:

In the areas inspected, violations or deviations were not identified. The ten (10) year ISI of Unit-2 reactor vessel welds and associated nozzles was successfully performed. Two adjacent indications, which were code acceptable were found in the weld of outlet nozzle seventeen (17). Tubes plugged because of EC inspection

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and/or laser welding associated problems included, eleven (11) in S/G "A", twenty-one (21) in S/G "B" and ten (10) in S/G "C". The number of tubes sleeved included eighteen (18) in S/G "A" and eleven (11) in S/G "C". The licensee chose not to sleeve tubes in S/G "B" because of logistics associated with technical support of laser welding equipment. UT examination of feedwater nozzles showed no evidence of crack indications in suspect areas. The RTD bypass system elimination was performed by Westinghouse (W) as a design change modification. Examination of the RPV welds and steam generator tubes were conducted by well trained and qualified personnel using state of the art equipment. Technical procedures and administrative controls were consistent with code and regulatory requirements and were adequately enforced. Surveillance of activities was maintained and documented on a daily basis by the licensee and W, QA personnel. Field problems were handled by well qualified technical personnel stationed on site while the activities were in progress. Licensee management was very responsive to a radiograph sensitivity concern raised by the inspector. Management took prompt and positive action to resolve this matter satisfactorily.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   C. Barefield, Engineer Plant Modifications
   W. Bayne, QA Supervisor
   *S. Casey, Supervisor, System Performance and Engineering
   *R. Coleman, Manager Plant Modifications
   J. Fitzzerald, ISI Coordinator
   *D. Hartline, ISI/IST Supervisor
   *R. Hill, Assistant General Manager Support
   *D. Morey, General Manager
   *C. Wesbilt, Manager Operations
   T. Sherrel, Level II Examiner Radiography
   *M. Stinson, Assistant General Manager Operations
   R. Woodfin, Maintenance and Operations Support Supervisor
   B. Yance, Manager Systems Performance

   Contractor Personnel

   Westinghouse (W), Nuclear Services Division

   R. Bedard, Outage Management Supervisor
   L. Kozak, Senior Engineer Laser Welding
   D. Kurek, Level III UT Examiner
   W. Stock, Level III EC Examiner
   D. Thompson, Onsite QA Engineer

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

   NRC Resident Inspectors

   *G. Maxwell, Senior Resident Inspector
   *M. Morgan, Resident Inspector

   *Attended exit interview

2. Inservice Inspection (73753) Unit - 2

   a. Reactor Pressure Vessel Examination

The ultrasonic examination (UT) of the reactor pressure vessel (PRV) conducted during this outage provided for the completion of first interval examination requirements that had been deferred to this outage and initiated examination activities for the second ten year interval. Both examinations were conducted under ASME Section XI (83,S83) Code requirements and the Farley Technical Specification Section 4.05; Regulatory Guide 1.150 Revision 1, was applicable to the extent defined by the licensee's position submitted to the Commission at an earlier date. The examination was being conducted by Westinghouse, Nuclear Service Division (W), (NSD). The controlling examination procedure was FMP-O-NDE-157.19, Rev. 3 Remote Inservice Inspection of Reactor Vessels for Farley Nuclear Power Plants. This procedure provided general requirements for straight and angle beam immersion UT examinations of long and circumferential welds, nozzle safe end welds, heat affected zones etc. Examinations were conducted from the RPV inner surface, inner radius surfaces, inside nozzle bores and from the flange seal surface. Requirements for calibration and examination parameters for the above mentioned welds and associated base metal areas, including location of each scan with reference to the vessel axis and datums, number of scan increments and the incremental progression between scans, were defined in the Examination Program Plan, Rev. 0, dated March 12, 1992.

Examinations were conducted with 1- 1/2" diam. 2.25MHz transducers with nominal refracted angles including: 0°L, 45°S, 60°S, 70°L, 12°L and 16°L. Specific arrays of these transducers were mounted on two separate plates. One was identified as the ten (10) year plate and the other, the 40 month plate which was in reference to the second interval inspection. The inspector observed, parts of near surface examinations of [illegible] belt line region weld Nos. 2, 3, 5 and, the examination of the nozzle to shell weld on outlet nozzle No. 17 in "A" Loop. Examination of this nozzle with the 0° transducer revealed an indication located on the nozzle side of the weld. Preliminary sizing showed that it had an amplitude of about 100% DAC, a length of about one inch and a through wall dimension of about 1.79 inches. This same indication was reexamined with a 1.5MHz, 0°L focusing transducer which showed that the indication was actually two overlapping indications with lengths of about 0.36" and 0.72". Following evaluation/analysis by level III examiners, the indication was found to be acceptable per applicable section of the Code including IWB-3512, IWB-3200 and IWB-3122.1. A review of previous PSI/ISI showed this indication was detected but not recorded during the January 16, 1985 RPV examination.

Within the areas of inspection, deviations or violations were not identified.

b. Eddy Current Examination (ET) of S/G Tubes

Through discussions with cognizant licensee personnel and by review of documents presented, the inspector ascertained that the ET examination program for this refueling outage included the following activities:

- 100% bobbin examination of all 3 S/G(s)
- Motorized rotating pancake coil examination of:
  - All distorted indications (DI's) and pluggable tubes with >40% through wall indications
  - 100% of hot leg (HL) roll transitions
  - 100% of all indications with a peak to peak bobbin voltage >1.0 volt and <3.6 volts
  - Confirmation and characterization of flaws
  - Verification that flaw is within the bounds of TSP
  - Augmented Inspection Program (100 intersections)
- 100% of row 1 and 2 U bends with special Zetec probe
- Baseline examination of sleeves with crosswound probe

The examination was being conducted by W, following applicable code requirements as well as general and specific guidelines (APR-2 Rev. 2), issued for this inspection by W and approved by the licensee. In addition to the above guidelines, the NRC issued Amendment 87, to technical specifications 4.4.6.4.a.6 Plugging and Repair Limit, and Bases 3/4.4.6 to address interim plugging and repair limit at tube support plate (TSP), intersections for the ninth operating cycle. Changes to the TS which addressed plugging and/or repair limits are summarized below:

- Degradation within the bounds of the TSP with a bobbin voltage ≤ to 1.0 volt will be allowed to remain in service.

Degradation within the bounds of the TSP with a bobbin voltage > 1.0 volt will be repaired or plugged except as noted below.

° Indications of potential degradation within the bounds of the TSP with a bobbin voltage > 1.0 volt, but ≤ to 3.6 volts may remain in service if an RPC probe inspection does not detect degradation.

° Indications of degradation with a bobbin voltage > 3.6 volts will be plugged or repaired.

At the time of this inspection, scheduled examinations were completed and W was in the process of analyzing suspect indications for plugging purposes. The total number of tubes identified with rejectable indications during this outage included 27 in S/G "A", 20 in S/G "B" and 20 in S/G "C". The total number of tubes plugged prior to and including this outage are as follows:

	S/G "A"	S/G "B"	S/G "C"
Total Number tubes plugged	310	187	312
% of Tubes Plugged	9.15%	5.54%	9.20%

c. Steam Generator Tube Repair, Sleeving

By letter dated October 22, 1990, the NRC issued Amendments No. 85 and No. 78 to the Farley Units 1 and 2 Technical Specifications (TS), authorizing the use of a laser welded sleeves for the repair of steam generator tubes per WCAP-12672. The laser welded tube sleeve and sleeving process has been described in WCAP-13115 and as such has been reviewed and approved by the NRC. This process allows installation of thermally treated Alloy 690 sleeves in both the hot and cold legs of steam generator tubes at tubesheet and support plate elevations.

° Tubes subject to sleeving are those exhibiting through wall degradation (TWD) which exceeds plant TS plugging limits. Installation of laser welded sleeves at support plates intersections has been addressed previously in safety evaluation report SECL-90-3336 and is licensed for use in the Farley S/G(s). Installation of laser welded tubesheet sleeves, which span from the end of the tube at the bottom surface of the tubesheet, to a point above the secondary side of the tubesheet, has been addressed in safety evaluation report, SECL-92-013, Rev. 2. Steam Generator Tube sleeving. In summary the subject sleeves are secured by first performing a hydraulic expansion of the upper and lower portions of the sleeve within the S/G tube. At the lower joint, a mechanical hard roll expansion is performed to provide both structural and leak resistance characteristics in this area. The hydraulically expanded region near the top of the sleeve is subsequently, laser welded to the S/G tube to provide structural support and leak tight integrity. In addition to the hydraulic expansion and mechanical hard roll performed at the lower end of the sleeve, a predetermined location of the sleeve in the clad region of the tubesheet is subsequently welded using the laser weld process. Following weld completion, welds located in the firespan region will be subjected to postwelded heat treatment to enhance resistant to primary water stress corrosion cracking. The weld joints in the firespan regions are UT examined to verify weld integrity. Welds that fail to meet UT acceptance criteria are generally plugged. Seal welds on the lower end of the sleeve, within the tube sheet, are subjected to remote visual inspection. Sleeves which were installed and examined successfully were ET examined with bobbin coil to provide baseline data for future reference.

Applicable code(s), which controlled design installation and testing requirements, included ASME Boiler and Pressure Vessel Code, Sections III (68 through S70), XI (83S83) and IX Edition and Addenda in affect at time of qualification, and Code Case N-395 Laser Welding Section III. Div. 1.

Technical procedures used to control and document site activities which were reviewed for content and technical adequacy were as follows:

FNP-O-SPP-GW-001 Rev. 10	General Welding Standard for Repairs, Replacements and Modifications

STD-FP-1990-5147	Laser Weld System, ROSA III Laser Welded Sleeving, 0.875" ODx0.050" Wall S/G Tubing
STD-FP-1990-5148 Rev. 1	Tube Cleaning System, Rosa III Laser Welded Sleeving 0.875" ODx0.050" Wall S/G Tubes
STD-FP-1990-5152 Rev. 1	Visual Inspection System, Rosa III, Laser Welded Sleeving 0.875" ODx0.050" Wall S/G Tubes
STD-FP-1990-5153 Rev. 1	Heat Treatment System, Rosa III, Laser Welded Sleeving .875" ODx0.50" Wall S/G Tubes
STD-FP-1990-5154 Rev. 1	Laser Welded Sleeving System Set-UP and Verification at Site
STD-FP-1992-5867 Rev. 0	Laser System Set-Up and Functional Test at Site
OOA-APR-S-1 Rev. 2	Site QA Surveillance Plan - Farley No. 2 Laser Weld Sleeving

By the end of this inspection on April 17, 1992, the licensee had identified 19 tubes in S/G "A" and 11 tubes in S/G "C" for sleeving. Because tube degradation at Farley is confined to the tubesheet region and at the first and second tube support plate (TSP), intersections, the licensee decided to install sleeves in each of the two TSP intersections and one in the tubesheet for each tube flagged for sleeving for a total of three sleeves per tube. This meant that in each sleeved tube there were five freespan laser welded joints and one seal weld within the tubesheet.

The inspector discussed with the W cognizant senior engineer various aspects of the laser welding process including its development, qualification of procedures and field application. In summary, the weld procedure was qualified to be used without filler metal. Nitrogen was used as shielding gas. The laser beam was produced with the use of a Lumonics laser unit, model JK706 and a fiber optic beam delivery system which generates a pulse beam with a frequency of about 14 Hertz. Heat input/energy was controlled by the limits of voltage and total weld time. Weld speed was between 6.50 and 6.77 ipm and it took two passes to complete the weld. Weld fabrication was accomplished by remote control through a computer which was programmed to check all the variables prior to welding. Weld size varies between 15 mils minimum, and 20 mils which is the optimum. Applicable welding procedure specifications were 74361 Rev. 0, used to qualify the freespan weld joint, and 74362 Rev. 1, used for the seal weld in the tubesheet. The inspector reviewed both documents and their respective procedure qualification records for compliance with the applicable code and the aforementioned Code Case. The inspector observed several of these activities including, installation, welding and ultrasonic testing of several sleeve welds. At the completion of this inspection the inspector ascertained that, out of the 35 tubes sleeved in S/G(s) "A" and "C", a total of six tubes were rejected by UT due to rejected welds. One of these was found in S/G "A" and the remaining five were in S/G "C". This rather high rejection rate was due, in part, to the fact that rejected welds could not be repaired at this time which meant that the tube had to be plugged. In addition to the aforementioned work effort, the inspector reviewed material certifications, receipt inspections and releases from stores records for the sleeves and plugs used during this outage. In a similar manner the inspector reviewed personnel certifications for NDE technicians and welding operators used at this time. Records of QA surveillances performed by W and the licensee were reviewed and found to be satisfactory. Third party inspection (ANII) was performed by Factory Mutual.

Within the areas inspected deviations or violations were not identified.

3. Steam Generator Feedwater Nozzle Inspections

Recently cracking of piping at the S/G Feedwater nozzle reducer to pipe welds was identified in a W PWR plant. The cracking was attributed to thermally induced fatigue, caused by introducing relatively cold feedwater into the main feedwater pipe upstream of the S/G. This cracking problem was identified in 1979, and NRC Bulletins 79-13 was issued to require inspection and replacement of defective feedwater piping components. Because of the recently identified problem, the inspector discussed the status of feedwater nozzle inspections at Farley. Through these discussions the inspector ascertained that the feedwater nozzle reducer to pipe weld in each of the three S/G(s) was ultrasonically examined during this outage. The examination results showed that no reportable indications were identified. A review of related records showed the examinations were performed in accordance with procedure UT-F-480 Rev. 4 which was written to comply with ASME Code Sections XI and V (83S83). Each weld was examined with a 45° and 60° shear wave transducer from the upstream side only because of nozzle geometry configuration. Weld root geometry was observed and recorded with a 60° scan in each of the three welds. This condition was documented as being intermittent over the entire length of the weld. Instrument calibrations, material and personnel certifications were reviewed and found to be satisfactory.

Within the areas inspected deviations or violations were not identified.

4. Resistance Temperature Detectors (RTD), Bypass Elimination, (Unit 2) (37700B)

During this refueling outage, the licensee replaced the existing RTD Bypass System with one utilizing new fast response RTDs designed to eliminate the bypass piping valves, flow elements, snubbers, whip restraints, hangers, insulation and the fully immersed RTDs associated with the existing RTD bypass system. Existing hot leg flow scoops, one per loop, were abandoned in place and a pipe cap was welded over the flow scoop piping to provide RCS integrity. The modification was performed under Production Change Request (PCR), No. 88-2-5260, Rev. 1 and related Production Change Notice (PCN), No. P-88-2-5260. In addition to the RTD bypass pipe elimination, this modification proposed the replacement of drain valve Q2B13V002B in reactor coolant loop 2 drain piping. This replacement was performed to allow unrestricted sensing of RCS level by LT-2965. Bechtel, who provided design support and documentation update, proposed that the existing Kerotest y-pattern, metal diaphragm valve be replaced with a Velan bellows seal globe valve with a lower flow coefficient (Cv), than the existing valve i.e., Cv 67 vs Cv 25. A W evaluation performed to determine the impact of the revised Cv, on system performance concluded that the lower flow coefficient, does not represent an unreviewed safety question as defined in 10 CFR 50.59. However the conclusion also stated that the modification imposed certain limitation on RCS draindown, addressed in evaluation (No. SECL-91-374), which must be implemented in the RCS draindown operating procedures. The inspector reviewed applicable maintenance work request (MWR), issued to direct and control field activities including the removal of existing piping in each loop and the installation of thermowells, basses and nozzle caps. Applicable MWRs included MWR-256317, -256318, -256320, -256321 and -256322. The aforementioned hardware were fabricated and inspected to meet the requirements of ASME Code Section III Class 1, 1983 Edition. Site installation and inspections were controlled by requirements of ASME Code Section XI, (83S83).

Quality records including material certifications, receipts inspection records and W quality releases for the above mentioned components were reviewed and found to be satisfactory. Field welding and nondestructive examinations were performed in accordance with the following procedures, written to meet the aforementioned applicable codes. These procedures were reviewed for content, technical adequacy and compliance with the above mentioned code case the reviewed procedures were as follows:

WEP-251, Rev. C	Welding Austenitic Stainless Steel by GTAW Method
WEP-200	Welding Austenitic Stainless Steel by SMAW Method

Supporting welding procedure qualification record reviewed included PCI-63 Rev. 0.

NED-240, Rev. 1 Liquid Penetrant Examination

NED-530, Rev. 0 Ultrasonic Thickness Measurements

20.A.131-1983, Rev. 0 Radiographic Examination of Welds

At the time of this inspection field installation and testing activities were completed. The inspector, inspected the weld on each of the three pipe caps for workmanship characteristics and appearance. In addition, the inspector reviewed radiographs of each of the three welds, identified as 2-A-1, 2-B-1 and 2-C-1. Weld thickness as documented on the radiographer's reader sheet was approximately 0.438 inches while the pipe diameter was indicated as three (3) inches. The inspector reviewed the film for each of the stations taken to cover the circumference of each weld, for film, radiographic quality and code acceptance as described in the above mentioned RT procedure. By this review the inspector ascertained that while film quality was satisfactory, the radiographic technique used, did not produce sufficient sensitivity to adequately display the required 4T penetrameter hole in some of the films reviewed for welds 2-B-1 and 2-C-1 and particularly in those for weld 2-C-1. In this weld, the inspector determined that only through the use of a magnifying glass and by knowing the location of the essential 4T hole, could one rate the presence of its image between very faint to none existent. This apparent lack of sensitivity gained additional significance because it made it difficult to evaluate the degree of root concavity that was identified in several locations over the length of weld 2-C-1. The inspector met with management and expressed his concerns over the adequacy of film sensitivity in these radiographs and especially for weld 2-C-1. Because the RPV cavity and associated piping was now flooded, it appeared that reshooting the subject weld(s) was not too practical and therefore, other NDE methods were discussed including UT or film enhancement. Following completion of this inspection the licensee notified the inspector that sufficient air had been trapped in the pretruding pipe stub(s) to allow the subject weld(s) to remain above water and be reradiographed satisfactorily. In addition, the licensee stated that the weld(s) were ultrasonically examined and found to be satisfactory. This prompt and positive response to the inspector's concern over a condition that presented possible safety implications demonstrates management's positive involvement towards a conservative and satisfactory resolution of significant technical issues. The reshots will be reviewed on a routine basis during future inspection.

Within the areas inspected deviations or violations were not identified.

5. Exit Interview

The inspection scope and findings were summarized on April 17, 1992, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection results listed below. No dissenting comments were received from the licensee. Proprietary information is not contained in this report.