Progress Energy

FEB 08 2008

Serial: HNP-08-017
10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/LICENSE NO. NPF-63
RFO13 90-DAY INSERVICE INSPECTION (ISI) SUMMARY REPORT

Reference: Letter from D. H. Corlett to the Nuclear Regulatory Commission (Serial: HNP-06-081), "90-Day Inservice Inspection (ISI) Summary Report," dated August 10, 2006

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a, Carolina Power and Light Company, doing business as Progress Energy Carolinas, Inc., submitted the above Refueling Outage 13 (RFO13) Inservice Inspection (ISI) Summary Report for the Harris Nuclear Plant (HNP). Attached is a revised Section 1 (Summary), consisting of six pages total (original submittal pages un-numbered). Please note that the two corrected paragraphs are indicated by revision bars on the fourth page and clarify the results of the two mechanical snubber functional tests.

This revision is provided in accordance with 10 CFR 50.9(a), which requires information provided to the Commission by a licensee be complete and accurate in all material respects. The corrected information does not have a significant implication for public health and safety or common defense and security per 10 CFR 50.9(b).

This document contains no new regulatory commitments.

Please refer any question regarding this submittal to me at (919) 362-3137.

Sincerely,

David Corlett

D. H. Corlett
Supervisor – Licensing/Regulatory Programs
Harris Nuclear Plant

DHC/kms

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

A047

Attachment

cc: Mr. P. B. O'Bryan, NRC Sr. Resident Inspector
Mr. J. M. Given, Jr., NC Department of Labor, Boiler Safety Bureau
Mr. V. M. McCree, NRC Acting Regional Administrator, Region II
Ms. M. G. Vaaler, NRC Project Manager

SUMMARY

The Shearon Harris Nuclear Power Plant, Unit 1, Ten-Year Inservice Inspection (ISI) Plan was developed to comply with 10CFR50.55a(g), which implements, by reference, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition with no Addenda.

The following summary report is being submitted pursuant to the reporting requirements of ASME Section XI, Subarticle IWA-6220(c).

This summary report contains the results of the Inservice Inspection (ISI) examinations performed on selected Class 1 and 2 pressure retaining components and their supports at Harris Nuclear Plant (HNP) from November 17, 2004 to May 16, 2006. The Refueling Outage No. 13 (RFO-13) examinations were performed during the third period in the second inspection interval. The typical designation is 2-3-13, (second interval, third period, thirteenth refueling outage). RFO-13 is the second outage of the third inspection period and the final outage of the second interval.

The components selected for examination were from the HNP ISI Program Plan and consisted of piping welds, vessel welds, component welds, component supports, and integral attachments.

ASME Section XI examinations were completed as scheduled in the HNP ISI Program Plan. The minimum and maximum percentage requirements for examinations completed meet the requirements of ASME Section XI, Tables IWB-2412-1 and IWC-2412-1.

NDE of Welds Summary

Nondestructive examinations were performed using Eddy Current (ET), Magnetic Particle (MT), Liquid Penetrant (PT), Ultrasonic (UT), and Visual (VT) techniques. Performance Demonstration Initiative (PDI) ultrasonic techniques were utilized where required. The results of the examinations were satisfactory with no rejectable indications noted. Indications identified during examinations were evaluated and documented as applicable on each weld examination data report. Code Case N-460 was invoked where the examination achieved greater than 90% coverage, but less than 100% code coverage. Those welds achieving greater than 90% coverage are considered complete. Weld examinations are summarized in Section 3 of this report.

NDE of Welds Summary (continued)

Automated UT, ET, and VT examinations of the reactor vessel welds were performed in accordance with ASME Section XI, 1995 Edition, 1996 Addenda, Appendix VIII as modified by 10 CFR 50.55a. The results of the examinations performed were satisfactory with no rejectable indications noted. Indications identified during examinations were evaluated and documented as applicable on each weld examination data report. In conjunction with the reactor examination, the reactor vessel nozzle-to-pipe dissimilar metal welds were examined using both UT and ET techniques. The dissimilar metal welds were examined with demonstrated techniques, and the results of the examinations were satisfactory with no indications detected.

During RFO-13, the reactor vessel head penetrations were examined to satisfy the requirements in Section IV.C.(5)(b) of the First Revised NRC Order EA-03-009 dated February 20, 2004 and to provide a baseline for future examinations. The examinations were conducted in accordance with written procedures and techniques that were demonstrated through the Electric Power Research Institute (EPRI) Technical Report (TR) 1007831, *Materials Reliability Program: Demonstrations of Vendor Equipment and Procedures for the Inspection of Control Rod Drive Mechanism Head Penetrations (MRP-89)* dated September 2003, and Westinghouse internal programs. The volume of the sixty-five penetration nozzles were examined using UT and ET techniques, the interference zone was inspected with 0° UT techniques to assess whether flaws were present. In addition, the ID surface was examined with a supplemental ET technique. The reactor vessel head vent line and the associated J-groove weld were examined using ET techniques. No evidence of primary water stress corrosion cracking (PWSCC) was identified by these examinations. The results of these examinations have been submitted with HNP's 60-day report (HNP-06-080) dated July 14, 2006.

Bare metal visual (BMV) examinations of all dissimilar metal piping butt welds were completed as identified in the EPRI TR 1010087, *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139)* dated August 2005. The locations of the BMV examinations included the six Inconel 82/182 piping welds associated with the pressurizer and the six Inconel 82/182 reactor vessel nozzle-to-pipe welds. These examinations revealed no evidence of boric acid leakage.

BMV examinations of the bottom mounted instrumentation (BMI) of the reactor vessel lower head were performed in RFO-13. Similar examinations were performed during previous refueling outages in accordance with NRC Bulletin 2003-02. Again, no relevant conditions indicative of boric acid leakage were found.

NDE of Welds Summary (continued)

Personnel performing the ISI examinations used isometric drawings in conjunction with tagged valves, component supports, points of connection, vessels, wall or floor penetrations, and floor elevations to locate and identify components subject to examination. Permanent component identification reference points were established and marked on the component in accordance with ASME Section XI, Article III-4000 during Preservice Inspection (PSI). The NDE examination reports, personnel certifications, examination equipment, NDE procedures, calibration standards and the ISI isometrics are listed in the RFO-13 Outage Examination Results Report. This report is on file and available at the plant site for review upon request.

Component Support and Snubber Examination Summary

ASME Section XI component support examinations were completed as scheduled in the HNP ISI Program Plan for outage 2-3-13 (RFO-13). ASME Section XI component support exams are complete as scheduled for the current ISI interval. Examination results were satisfactory except for one pipe support. This support was found with an excessive gap between the pipe clamp and the integrally-welded attached pipe lug. The design called for a 1/16" gap, but the as-found gap was approximately 1/8" to 1/4". Engineering evaluated this condition and determined that the support would still perform its required function. This condition was documented in the corrective action program. The condition was not a result of in-service wear, and the gap was corrected by installing a shim plate.

In RFO-13, the Technical Specifications (TS) Augmented Snubber Examination and Testing program included visual examination of 282 snubbers. Four visual examination failures were observed on the Steam Generator (SG) Blowdown system.

Specifically, two snubbers located on the "C" SG normal blowdown path were observed with inadequate settings for their design thermal movement. Engineering evaluated the "as-found" condition and determined that the supports would still perform their required functions because the potential reduction in movement would be limited to approximately 1/32" and would not result in adverse pipe stresses. This condition was corrected by resetting these snubbers to ensure adequate design thermal movement. No other snubbers were found with this condition on the affected pipe. The other two visual examination failures involved two hydraulic SG snubbers, which were observed with low fluid level. The source of the fluid loss was corrected, and the fluid was restored to normal level. In addition to the observations above, two snubbers were found with arc strikes. These snubbers were removed, functionally tested, and found to meet their acceptance criteria. The visual examinations were performed in accordance with the 10-year snubber examination interval described in ASME Operation and Maintenance (OM) Code Case OMN-13, as implemented by applicable plant procedures.

Component Support and Snubber Examination Summary (continued)

RFO-13 snubber testing activities included functional testing of 37 mechanical snubbers and five large bore hydraulic snubbers in the representative sample plans, and hand-stroke testing of 23 mechanical snubbers in the harsh environment test group. Testing results for the mechanical snubbers in the representative 37 sample plan resulted in two functional test failures. All five hydraulic snubber functional test results were satisfactory. All 23 harsh environment snubbers were observed to stroke freely, so the hand-stroke test results were satisfactory.

The following details are provided for the results of the two functional test failures.

One mechanical snubber failed to properly activate due to a manufacturing defect. The snubber did stroke smoothly with drag values within the acceptance range. One mechanical snubber failed to properly activate due to a manufacturing defect. The snubber did stroke smoothly with drag values within the acceptance range. The cause was a manufacturing defect originating from inadequate crimping of the guide rod. The failed snubber was replaced with a pretested snubber. Per Relief Request 2RG-008 approved on June 18, 1999, scope expansion on a manufacturing failure is not required. Although not required, another snubber of similar size and application was selected and functionally tested. The functional test results for this additional snubber were satisfactory.

The other functional test failure involved a snubber on the main steam (MS) piping exiting the "A" steam generator. The snubber passed the activation functional test, however, it seized during its drag test. The manufacturer evaluation found that the snubber had localized fretting damage along the ball screw shaft where the ball nut contacts the shaft during the normal hot position. The fretting was evaluated and attributed to low amplitude vibration in a horizontal section of pipe, which may also affect the other two vertical MS pipe systems exiting the other steam generators. In accordance with Relief Request 2RG-008 approved on June 18, 1999, a failure mode group (FMG) was established that encompassed six MS piping snubbers, including the snubber that had a functional test failure. The other five snubbers in the FMG were functionally tested with no failures found. The condition was corrected by replacing the failed snubber with a pre-tested snubber.

Component Support Examinations are summarized in Section 4 of this Report.

Steam Generator ISI Summary

Fifty percent (50%) of the open tubing in all three steam generators (SG) was inspected using ET technology (bobbin exam) during RFO 13. Rotating coil examinations were conducted on approximately 20% of the tubing in the outer tube bundle and blowdown lane regions in all three SGs on the top of the secondary tubesheet (hot and cold leg sides). Twenty percent (20%) of the Row 1 U-bends were also inspected with rotating coil examinations.

No degraded tubes were identified from the ET data analysis. The examination results for the SGs are classified as C-1 per the Harris Technical Specifications and EPRI NDE Examination Guidelines. Category C-1 has the following criteria: "Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective".

System Pressure Tests

ASME Class 1, 2 and 3 Pressure Tests were performed during RFO-13 in accordance with ASME Section XI, Articles IWA-5000, IWB-5000 and IWC-5000.

Code Case N-533 was invoked to allow insulation removal at bolted connections while the system was not at operating pressure. Relief Request 2RG-009 was invoked to allow evaluation of leakage at bolted connections in lieu of removing one bolt. Relief Request 2RG-010 was invoked to schedule and examine piping welds with a Risk-Informed Program.

Where leakage was identified, it was either evaluated in accordance with Relief Request 2RG-009, or measures were taken to correct the condition. The Pressure Test records and supporting documentation are on file and available at the plant site for review upon request.

Containment Inspections

No IWE examinations of the ASME Class MC Components were required or scheduled during RFO-13. However, as a prudent measure, examinations were performed between the moisture barrier and approximately 12" up from the moisture barrier on the liner. No recordable conditions were observed during these examinations. In addition, as follow-up corrective action from the previous outage, the 'A' Containment Spray valve chambers were opened and examined. A VT-1 examination was performed on all bolting and nuts of the valve chamber manway. No recordable conditions were observed during this examination.

No IWL examinations were required or scheduled in RFO-13. However, for information, an ongoing examination of the exterior containment concrete is in progress, and the results of this examination will be reported in a future ISI Summary Report.

Repair and Replacement Summary

The repair and replacement summary and the attached ASME Section XI, NIS-2 forms detail the component, system, work order (WO) number, component description and the description of the work order activity. Complete repair and replacement documentation for the specific component is maintained at the plant site as a permanent record and is retrievable through the work request (WR) or WO document package, which is identified on the NIS-2 form. Section 5 of this report provides the repair and replacement documentation.