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

Report No.: 50-400/87-16

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

Docket No.: 50-400 License No.: NPF-63

Facility Name: Harris 1

Inspection Conducted: April 6 - 10, 1987

Inspector: *Frank Jape* for P. T. Burnett — 5/11/87 Date Signed

Approved by: *Frank Jape* — 5/11/87 Date Signed
F. Jape, Chief
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope: This routine, unannounced inspection addressed the review of startup tests completed at 50 and 75 percent of rated power and licensee response to IE Information Notices.

Results: No violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *R. A. Watson, Vice President, Harris Nuclear Project
   J. L. Wills, Plant General Manager
   *H. W. Bowles, Director, Onsite Nuclear Safety
   J. M. Collins, Manager, Operations
   R. J. Duncan, Test Program Development Engineer, Technical Support
   *G. L. Forehand, Director Quality Assurance/Quality Control
   *J. L. Harness, Assistant Plant General Manager, Operations
   *A. J. Howe, Regulatory Compliance
   *C. L. McKensie, Principal Quality Assurance Engineer
   *C. E. Rose, Jr., Quality Assurance Supervisor
   *J. R. Sipp, Manager E&RC
   *J. H. Smith, Operations Support Supervisor
   R. B. Van Metre, Manager, Technical Support
   *W. R. Wilson, Principal Engineer, Technical Support
   R. R. Wojonarowski, Reactor Engineering Leader, Technical Support

   Other licensee employees contacted included shift foremen, startup engineers, control room operators, and office personnel.

   NRC Resident Inspectors

   G. F. Maxwell, Senior Resident Inspector
   S. P. Burris, Resident Inspector

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on April 10,1987, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings. No dissenting comments were received from the licensee. Proprietary material was not reviewed in the course of the inspection.

3. Licensee Action on Previous Enforcement Matters (92702)

   (Closed) Violation 86-96-01: Inadequate procedure for measuring reactor coolant system leakrate. The inspector reviewed the revised procedure OST-1026, and analysed surveillances performed under it using micro computer program RCSLK9, from the NRC Independent Measurements Program. Six completed copies of OST-1026, Reactor Coolant System Leakage Evaluation, which were performed in early March, 1987, were reviewed, and

the results compared with calculations using RCLSK9. (System temperatures ranged from 345 to 571 F, all at 2235 psig.) Agreement for both gross and unidentified leakage was within 0.2 gpm in all cases. In those cases in which there was no net change in reactor coolant system average temperature there were no differences in the gross leakage calculations. The small differences that arose when there were changes in average temperature derived at least in part from the licensee correcting for both system volume and coolant density changes, while RCSLK9 accounts only for the latter. Whether the vessel and piping actually respond to changes in coolant temperature in the time frame of the test is open to question.

4. Unresolved Items

No unresolved items were identified during this inspection.

5. Fifty Percent Power Tests (72608)

a. 9105-S-01, Calibration of Steam and Feedwater Flow Instrumentation at Power - 50% (Retest) was performed March 13, 1987 and accepted on April 7, 1987. The level 2 acceptance criterion for the feedwater flow transmitter signals was that they agree within 5% of full scale d/p of the special test instruments. This is a large number, of the order of 61 inchs-water, and about one-third the reading at 50% power. However, the acceptance criterion in 9107-S-03 (90% power data) is no discrepancy in excess of 0.5% full scale d/p (approximately 6.1 inches - water) special test instrument accuracy (1.4 inches - water). This will translate to about 1% instrument error at full power, which, if attained will be acceptable.

b. 9105-S-05, Core Performance at 50% Power, was completed on February 21, 1987, and the results were approved on March 3, 1987. The heat flux hot channel factor and nuclear enthalpy rise hot channel factor each satisfied its technical specification limit at 50 and 75% power, thus justifying escalation of power to the 75% testing plateau. The maximum quadrant power tilt ratio was 1.007, well below the limit of 1.02. The INCORE-calculated average reaction rate error was 4.7%, again, well below the level 2 acceptance criterion of 10%. The reaction rate error is the difference between predicted power production in an assembly and the measured value. Determination of control rod position by use of the incore nuclear instruments was also demonstrated to agree within 12 steps with main control board indications.

The review included the following procedures, which were performed in support of this test:

(1) FMP-101 (Revision 2), In-Core Thermocouple and Flux Mapping,

(2) EST-710 (Revision 3), Hot Channel Factor Tests,

(3) EST-722 (Revision 0), Control Rod Position Determination Via Incore Instrumentation, and

(4) EPT-052 (Revision 0), Power Range Heat Balance Via Precision Calorimetric.

c. 9105-S-06, Thermal Power Measurement and Statepoint Data Acquisition at 50% Power (Retest 2) was performed on March 13, 1987 and the results accepted on April 7, 1987. The test had no level 1 acceptance criteria and was judged successful when acceptable data were obtained, submitted for use in other tests, and thermal power calculated.

d. 9105-S-07, NIS Overlap Verification, Power Range Calibration and Setpoint Adjustment - 50% was completed on February 23, 1987 and approved on March 3, 1987. Adjustment of all four power range nuclear instruments was within two percent of the indicated thermal power as determined by test procedure 9105-S-06. Plots of chamber current versus power at zero, 30, and 50% thermal power all appeared to be linear.

e. 9105-S-09, Power Coefficient at 50% Power, was performed on February 22, 1987, and the results were accepted on March 3, 1987. Xenon stability during the test was confirmed by use of the computer code EXSPACK, which indicated that prior to starting the test xenon was changing by about 6pcm/hr. The total reactivity worth of xenon was about 2450 pcm.

The initial thermal power was determined using procedure OST-1004, Power Range Heat Balance-Daily Interval.

f. 9105-S-11, Loss of Feedwater Heaters Test at 50% Power, was performed on March 29, 1987 and the results accepted on April 7, 1987. There were no level 1 acceptance criteria. The level 2 criterion was that no measurement of feedwater temperature drop more than 44 F. The three recorded measurements of temperature drop were 11, 5, and 11F; satisfying the criterion.

g. 9105-S-14, Reactor Coolant System Flow Measurement at 50% Power, was performed on February 22, 1987 and the results were accepted on February 25, 1987. The measured flow was 309420 gpm, which satisfied the acceptance criterion that flow be greater than 298948 gpm, which includes allowances for error. Data sheet 10.3 and acceptance criterion 7.2.1 indicate the flow rate is 309970 gpm, in contrast to the value above from section 1 of the procedure. The data used were obtained from test procedure 6105-S-06. Data sheet 10.1 demonstrates that the average reactor coolant system (RCS) temperature and steam generator and pressurizer levels were stable over the period of the test. The final determination of reactor coolant system flow will be performed using a precision heat balance once the reactor achieves 100% rated thermal power.

h. 9105-S-18, Preliminary Incore/Excore Calibration, was performed on February 22, 1987 and accepted on March 3, 1987. Performance at this power level precedes the Technical Specification requirement to perform at 75% power. Hence, the use of only two quarter-core flux maps at only two different axial flux differences is acceptable. This test provided an opportunity to make practical use of the engineering and maintenance surveillance tests (ESTs and MSTs, respectively) that were to be applied in the required surveillance.

No violations or deviations were identified.

6. Seventy-five Percent Power Tests (72616)

Several 70% power tests were left to be performed at the time of this inspection. Most of the completed tests were in the review cycle. However, data from the incore-excore calibration test at 75% power were available, and the inspector performed independent calculations of the least squares fit of axial offset to chamber current for each of the eight chambers. Exact agreement on zero-offset, full-power current, slope, and correlation coefficient was obtained in every case. Five data pairs were used in each calculation and all correlation coefficients were greater than 0.995.

No violations or deviations were identified.

7. Licensee Followup of IE Information Notices (92703)

IE INFORMATION NOTICE NO. 86-14: PWR AUXILIARY FEEDWATER PUMP TURBINE CONTROL PROBLEMS and IE INFORMATION NOTICE NO. 86-14, SUPPLEMENT 1: OVERSPEED TRIPS OF AFW, HPCI, AND RCIC TURBINES have been considered by the licensee. The review of the original notice was completed on April 16, 1986 and the review of the supplement was completed on January 19, 1987. Based upon discussions with representatives of Woodward Governor and utilities experiencing mechanical overspeed trips from oil pressure in the governor, licensee engineers concluded that no similar mechanism existed at Harris. They did, however, identify a potential for condensate-buildup-induced overspeed trips and made specific recommendations for corrective action. The modification portion of the corrective action had not been implemented at the time of this inspection.

Overspeed trips of the turbine driven AFW pump were experienced during the remote shutdown and loss of offsite power tests. Initially the cause was assigned to a loose electrical speed probe connection. Maintenance findings and post-maintenace testing appeared to justify that conclusion. Subsequently, additional overspeed trips were experienced, and they were ascribed to condensate buildup in the steam supply lines. Currently, the condensate is being drained several times per shift in response to annunciator alarms, and the recent experience has sensitized the operators to the need for prompt response to the alarms.

Corrective modifications are currently being planned. The need for modifications to eliminate condensation in the steam supply line to the pump turbine stems from the current practice of keeping the steam supply valve closed. This results in unheated line length of over 100 feet in which steam leakage past the supply valve can condense without being drained by automatic action. If the supply were controlled at the turbine governor valve, there would be less than three feet of line to heat on pump start, and condensation in the remainder of the steam line would be forced through the condensing pot drains by steam pressure. Apparently, the decision to control steam at the supply valve was made to avoid performing high energy line break analysis to the 100 plus feet of steam line between the supply valve and the governor valve. Subsequent discussions with members of the Plant Systems Branch, NRR confirmed that classifying the steam line as a non-high energy line in the current mode of operation was acceptable.

IE INFORMATION NOTICE NO. 87-05: MISWIRING IN WESTINGHOUSE ROD CONTROL SYSTEM was reviewed by the licensee, who prepared a test procedure to determine if the problem existed at Harris. That test, EPT-041T, was performed on March 30, 1987. A review of the completed test procedure confirmed that it was responsive to the concern expressed in the notice and that the wiring error did not exist in the Harris 1 rod control system.

No violations or deviations were identified.