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

Report Nos.: 50-280/88-41 and 50-281/88-41

Licensee: Virginia Electric and Power Company
Richmond, Virginia 23261

Docket Nos.: 50-280 and 50-281 License Nos.: DPR-32 and DPR-37

Facility Name: Surry 1 and 2

Inspection Conducted: October 2 through November 5, 1988

Inspectors: Mike Scott for 11/30/88
W. E. Holland, Senior Resident Inspector Date Signed

Mike Scott for 11/30/88
L. E. Nicholson, Resident Inspector Date Signed

Accompanying Inspector: P. Fillion

Approved by: F. S. Cantrell 11/30/88
F. S. Cantrell, 2A Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine resident inspection was conducted on site in the areas of licensee actions on previous enforcement matters, plant operations, plant maintenance, plant surveillance, licensee event report review, and design changes and modifications.

Results: One apparent violation was identified (280,281/88-41-01) for failure to take appropriate corrective actions for identified deficiencies was noted as follows:

Failure to promptly identify a deviation to the shift supervisor and prepare a deviation report on August 29, 1988, that potential gas binding may adversely effect the operability of the high head safety injection pumps (paragraph 5).

Failure to adequately evaluate the adverse condition documented in station deviation S1-87-946 from November 20, 1987, to April 11, 1988, with regards to control room chiller capacity (paragraph 3).

Failure to identify the potential control room envelope ventilation problem at the time information was available to question the capability of the system (paragraph 7).

Failure to take appropriate corrective actions for an NRC identified violation with regard to inventory of special nuclear material which was discussed in inspection report 280,281/87-10 (paragraph 3).

These examples listed above indicate a weakness in past implementation of the corrective action program at the Surry Power Station.

In addition, an inspector followup item was identified in paragraph 5 for followup on licensee evaluation of technical issues identified during review of loss of Pressure Relief Tank (PRT) water event (280,281/88-41-02).

REPORT DETAILS

1. Person Contacted

Licensee Employees

J. Bailey, Superintendent of Operations
*D. Benson, Station Manager
*R. Bilyeu, Licensing Engineer
H. Blake, Superintendent of Site Services
*R. Blount, Superintendent of Technical Services
*E. Grecheck, Assistant Station Manager
*G. Miller, Licensing Coordinator, Surry
H. Miller, Assistant Station Manager
*J. Ogren, Superintendent of Maintenance
*J. Price, Site Quality Assurance Manager
S. Sarver, Superintendent of Health Physics

*Attended exit meeting.

Other licensee employees contacted included control room operators, shift technical advisors, shift supervisors and other plant personnel.

The NRC Region II Section Chief, F. Cantrell, visited the Surry Power Station on October 12, 13, and 14, 1988. Mr. Cantrell's tours included the low level intake structure, service water system walkdown, auxiliary building, and the Unit 2 containment.

On October 18, 1988, the NRC NRR Director for Project Directorate II-2, H. Berkow, visited the Surry Power Station to review the current status of issues and to tour the station. Mr. Berkow was accompanied by the current Surry Project Manager, C. Patel; and also was accompanied by the oncoming Surry Project Manager, B. Buckley. In addition, Mr. W. Troskoski of the NRC Executive Director's Staff visited the Surry Power Station on the same day and was briefed by the Resident Staff.

On October 20, 1988, one of the Commissioners of the Nuclear Regulatory Commission, Kenneth M. Carr, visited the Surry Power Station for a familiarization tour, to meet with licensee management and staff, and to review current station status. Commissioner Carr was accompanied by the following personnel:

M. L. Ernst, Acting Regional Administrator
M. Federline, Technical Assistant to the Commissioner
B. Wilson, Branch Chief, DRP, Region II
NRC Resident Inspectors

The Commissioner attended the morning management status meeting; met with the resident inspectors; was given a presentation of several areas by licensee management; and was taken on a tour of the service water system, battery and switchgear rooms, control room, emergency diesel generator rooms, and the independent spent fuel storage installation.

2. Plant Status

Unit 1

Unit 1 began the reporting period at cold shutdown with preparations being made to defuel the reactor in order to find and replace leaking fuel assemblies. Fuel offload had commenced. However, at the end of the inspection period, one fuel assembly in location G-6 in the reactor vessel became stuck and efforts were continuing to free the assembly from the manipulator crane gripper when the period ended.

Unit 2

Unit 2 began the reporting period at cold shutdown in day 21 of a scheduled 81 day refueling/maintenance outage. Installation of new recirculation spray heat exchangers was completed and preparations were being made to defuel the reactor when the inspection period ended.

3. Licensee Action on Previous Enforcement Matters (92702)

(Closed) Unresolved Item (URI) 280,281/88-12-01, Engineering Evaluation and Report of Control/Relay Room Chillers. This item was identified in inspection report 280,281/88-12 during a region-based inspection of the performance of the control room and emergency switchgear room chiller system. The inspector noted that station deviation S1-87-946, dated November 20, 1987, identified the fact that the subject chiller units did not meet the 90 ton capacity specified in the UFSAR. No formal evaluation or 10 CFR 50.59 review was performed until the inspector raised the issue during the week of April 11, 1988. The subsequent analysis, dated April 19, 1988, justified continued operation as long as service water temperature remained below 70 degrees F. The licensee has upgraded the chillers to meet the design capacity. However, as discussed in paragraph 7 of this report, the entire system has since been determined to be degraded to the point of not being able to meet performance specifiction. The failure to adequately evaluate the adverse condition documented in station deviation S1-87-946 is an additional example of violation 280,281/88-41-01 which is discussed in paragraphs 5 and 7 (control room ventillation) of this report. This unresolved item is therefore closed.

(Closed) Unresolved item (URI) 280,281/88-18-01, Review of procedures for configuration control of piping blanks. This item was identified in inspection report 280,281/88-18. In that report the inspector had identified a concern with regards to control of piping blanks which are routinely removed by operations in order to establish temporary flowpaths for evolutions involving mid-nozzle operation during outages. Since that time, the inspector was provided with the results of a review which was conducted at the direction of the operations superintendent in his area.

That review concluded that, for the most part, configuration control was being adequately maintained when piping blanks were removed to support operations evolutions. However, certain procedures did not adequately address removal and/or reinstallation of blank flanges to support operations or maintenance activities. Some of these procedures were:

- Operations Procedure (OP)-19.2, Containment Vacuum System - Refueling Operations

- Maintenance Operating Procedure (MOP)-5.6, 5.7, 5.8; Reactor Coolant System Loop Fill

- MOP-8.1, 8.4, 8.6; Return to Service of Charging Pump A, B, C

- MOP-14.1, 14.2; Remove RHR Pumps from Service

- MOP-14.3, 14.4; Return RHR Pumps to Service

Corrective actions for the operations procedures were entered on the commitment tracking system by the Operations Superintendent on September 7, 1988.

The inspector reviewed the licensee actions in the operations area and determined that they were appropriate. The inspector also reviewed the problem area with other department supervisory personnel and concluded that a similar problem does not appear to exist in other station areas.

Technical Specification 6.4 requires that detailed written procedures with appropriate check-off lists shall be provided for normal startup, shutdown, and operation of a unit and of all systems and components involving nuclear safety of the station. Failure to provide adequate procedure for installation or removal of piping blanks on safety-related systems is a violation of Technical Specification 6.4.

The inspectors reviewed the findings and the corrective actions performed prior to the end of the inspection period and determined that they are acceptable. After discussion between the inspectors and NRC regional management, it was concluded that adequate corrective actions to prevent recurrence have been taken by the licensee prior to the end of the inspection period.

(Open) Violation 280,281/87-10-01, Failure to conduct an annual physical inventory for all special nuclear material. The violation was identified during a NRC inspection in May, 1987, and was discussed in inspection report 280,281/87-10. Licensee response, in part, to the violation on July 17, 1987, agreed that the violation was correct and stated that corrective steps which will be taken to avoid further violations included: (1) preparation of a new procedure to perform the required inventories, (2) conduct of a physical inventory in accordance with the procedure on July 8, 1987, (3) conduct of the next physical inventory on September 30, 1987, and (4) conduct of future inventories on a semi-annual basis to be concurrent with the DOE reporting requirements.

During this inspection period, the inspector was informed that the licensee commitment to perform items 3 and 4 above had not been accomplished as required. This condition was identified during an audit by the licensee's quality assurance organization. The failure to perform these requirements was identified in the licensee's corrective action program by a deviation report (S1-88-1035) written on October 6, 1988. This deviation report resulted in additional licensee review of the requirements of the regulations and concluded that their initial response to the violation was inadequate. Their conclusions were that (1) detector location was not clearly identified during past audits, (2) the most recently performed periodic audit procedures were signed off as complete, when in fact they were incomplete, and (3) the most recent deviated procedure which was prepared on October 6, 1988, to establish an accurate physical inventory of all detectors could not be accomplished due to personnel being unwilling to sign off verification steps based on memory.

The inspector reviewed the preceding conclusions with licensee management and was informed that they would be sending a revised response to the violation identified in inspection report 280,281/87-10. Failure to take appropriate corrective actions for an NRC identified violation is identified as an additional example of violation 280,281/88-41-01 which is discussed in paragraph 5 of this report.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. No new unresolved items are identified in this inspection report.

5. Plant Operations

Operational Safety Verification (71707)

The inspectors conducted daily inspections in the following areas: control room staffing, access, and operator behavior; operator adherence to approved procedures, technical specifications, and limiting conditions for operations; examination of panels containing instrumentation and other reactor protection system elements to determine that required channels are operable; and review of control room operator logs, operating orders, plant deviation reports, tagout logs, jumper logs, and tags on components to verify compliance with approved procedures.

The inspectors conducted weekly inspections in the following areas: verification of operability of selected Engineered Safety Feature (ESF) systems by valve alignment, breaker positions, condition of equipment or component(s), and operability of instrumentation and support items essential to system actuation or performance.

Plant tours included observation of general plant/equipment conditions, fire protection and preventative measures, control of activities in progress, radiation protection controls, physical security controls, plant housekeeping conditions/cleanliness, and missile hazards. The inspectors routinely monitor the temperature of the auxiliary feedwater pump discharge piping to ensure steam binding is prevented.

The inspectors conducted biweekly inspections in the following areas: verification review and walkdown of safety-related tagout(s) in effect; review of sampling program (e.g., primary and secondary coolant samples, boric acid tank samples, plant liquid and gaseous samples); observation of control room shift turnover; review of implementation of the plant problem identification system; verification of selected portions of containment isolation lineup(s); and verification that notices to workers are posted as required by 10 CFR 19.

Certain tours were conducted on backshifts or weekends. Backshift or weekend tours were conducted on October 3, 5, 6, 8, 9, 10, 11, 12, 13, 14, 15, 17, 18, 19, 20, 22, 23, 24, 29; and November 3, 4, and 5. Inspections included areas in the Units 1 and 2 cable vaults, vital battery rooms, steam safeguards areas, emergency switchgear rooms, diesel generator rooms, control room, auxiliary building, Units 1 and 2 containments, cable penetration areas, independent spent fuel storage facility, low level intake structure, and the safeguards valve pit and pump pit areas. Reactor coolant system leak rates were reviewed to ensure that detected or suspected leakage from the system was recorded, investigated, and evaluated; and that appropriate actions were taken, if required. The inspectors routinely independently calculated RCS leak rates using the NRC Independent Measurements Leak Rate Program (RCSLK9). On a regular basis, radiation work permits (RWPs) were reviewed and specific work activities were monitored to assure they were being conducted per the RWPs. Selected radiation protection instruments were periodically checked, and equipment operability and calibration frequency were verified.

In the course of monthly activities, the inspectors included a review of the licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities to include: protected and vital areas access controls; searching of personnel, packages and vehicles; badge issuance and retrieval; escorting of visitors; and patrols and compensatory posts.

POTENTIAL FAILURE OF SPENT FUEL POOL DOOR SEAL

The inspectors monitored the licensee actions regarding the discovery that a potential exists for a failure of the spent fuel pool door seal. The licensee identified, via station deviation report S1-88-1027, that during certain evolutions the only barrier that prevents emptying the spent fuel inventory into containment is an inflatable seal around the transfer canal door. Prior to moving activated fuel, the licensee opens the fuel transfer tube gate valve and moves a dummy fuel assembly through the containment fuel pool penetration. It is during this evolution that the water in the spent fuel pool is retained only by an inflated seal around the transfer canal door. A loss of instrument air that inflates the door seal could also render the transfer system inoperable and prevent closing the transfer tube gate valve.

The licensee was in the process of testing the fuel transfer system when an operator noted an air leak in the air supply to the inflatable door seal. Investigation revealed a spot light had been placed adjacent to the hose from the air system regulator to the seal, and it had burnt a small hole in the airhose. The airhose was temporarily repaired using tape, and the gate valve was closed until a permanent repair could be performed. The door seal remained inflated during this event with no loss of water past the seal. However, at the same time this condition occurred, an operator was preparing to enter the refueling canal to aid in positioning the dummy fuel assembly. In addition, as documented in station deviation S1-88-1012, the fuel transfer system conveyor air motor became mechanically bound. This air motor is supplied from the same instrument air supply as the door seal. A work order was issued and the motor was subsequently repaired. After repair, the checkout evolutions involving transfer of the dummy assembly were completed.

The above event caused the senior reactor operator in charge of refueling to evaluate the scenario and submit a station deviation identifying the overall concern. Given the worst case of a fuel transfer system failure with the dummy assembly stuck in the tube, coupled with a failure of the door seal to retain the spent fuel pool water, the spent fuel pool would drain to within 13 inches of the top of the fuel assemblies. This water would drain inside containment and flood the basement since the reactor cavity seal is not normally installed during this evolution. The licensee mandated that the fuel transfer gate valve remain closed until an investigation into the event could be performed. This investigation, dated October 10, 1988, concluded the following:

- Loss of instrument air to the seal could result in a significant leakrate into the spent fuel building transfer canal and containment.

- The gate valve (22 inches diameter) could be closed during canal door seal leakage, if no obstructions blocked the closure.

- With a total loss of air and the transfer cart protruding through the transfer tube, it may not be possible to reinflate the seal or isolate the transfer tube (close the gate valve).

- It would take approximately 85 minutes to drain to the top of the weir (approximately 13 inches over the fuel assemblies) with no makeup.

- Since fuel pool cooling would be lost, maximum heatup rate would be less than 4.5 degrees F/hour, which would allow approximately 24 hours before bulk pool boiling would occur.

- Assuming 13 inches of shielding above the top of the fuel assemblies, a dose rate of approximately 50 R/hour at the edge of the spent fuel pool would be expected.

The report recommendations included a requirement that the reactor cavity seal assembly be installed prior to opening the transfer tube gate valve for testing the conveyor system. This would limit a drain down of the spent fuel pool to approximately 14 feet over the top of the spent fuel racks, thus providing adequate shielding to perform recovery operations in the fuel building.

The inspectors continued to review the licensee corrective actions regarding this postulated scenario. An action plan, dated October 12, 1988, was developed by the licensee to implement both short term and long range corrective actions. Identification of the potential problem by the refueling SRO indicates a increased sensitivity to safety issues that is commendable. The extensive investigation into this event after a station deviation was submitted was also commendable. However, it is should be noted that the failure of the reactor cavity seal on May 17, 1988, had not resulted in a generic review of similar seal configurations at the station until the spent fuel pool seal potential problem was highlighted by the above occurrence.

LEAKAGE OF PRT WATER INSIDE CONTAINMENT

The inspectors investigated the circumstances that resulted in approximately 250 gallons of water leaking from a pressurizer safety valve flange on Unit 2. This event occurred on October 4, 1988, and was identified in station deviation report S2-88-520. The licensee had been venting the reactor for several days by maintaining the reactor coolant system (RCS) at mid-nozzle and degassing through an empty pressurizer, into the pressure relief tank (PRT), and out the process ventilation system. The pressurizer code safety valves were removed and a temporary cleanliness cover (gasket material) was placed over the flange openings. The pressurizer power-operated relief valves were being maintained open providing a vent path from the pressurizer to the PRT. The event was initiated when the control room operator attempted to vent and depressurize the safety injection accumulators utilizing operating procedure 2-OP-7.7.4, Venting Safety Injection Accumulators. Step 5.1.3.4 of the above procedure opens valve HCV-2936 and vents the accumulators into the process vent system. It was at this point that the control room operator noticed fluctuations in the RCS standpipe level and an increase in PRT pressure. The PRT level was noted to decrease from 13 to 10 percent during the event, thus translating to a loss of inventory from the PRT of approximately 250 gallons. The operator secured the accumulator vent and noted that the RCS standpipe level indication returned to normal. It was also reported that water was flowing from the pressurizer safety valve flanges that would indicate that the PRT water was being displaced back toward the pressurizer through the discharge lines from the primary relief valves which had been removed for maintenance.

The inspector reviewed this event with the station staff performing an investigation into the incident. The event team also researched the possibility of losing RCS inventory out the incore thimble guide tubes that are disconnected at the seal table with their low pressure seals installed. This scenario was deemed to be the worst case condition since it was determined that enough pressure could develop in the reactor coolant system to blow the low pressure seals on the thimble guide tubes and therefore constitute a cold leg loss of inventory.

The above event signifies the complexity of problems when using a shared system, such as the process vent system, to perform concurrent functions. The licensee event team identified several weaknesses with procedures, system interfaces, valve lineups, and level indications. It was also recommended that an alternate method of venting the accumulators, such as to the containment atmosphere, be evaluated. The licensee reported this event via the INPO Network System on October 13, 1988.

The inspector reviewed the results of the investigation and agreed that significant questions remain to be answered regarding the impact of this event. These questions include possible overpressurization of piping, inaccurate level indication, inaccurate sparger location, and the effects of the boric acid that was spilled on adjacent components. The licensee was researching the above items when the inspection period ended. Therefore, this will be identified as an inspector followup item for followup on licensee evaluation of technical issues identified during review of loss of PRT water event (280; 281/88-41-02).

INSIDE RECIRCULATION SPRAY PUMP DEGRADATION - UNIT 2

On October 7, 1988, the licensee made a 10 CFR 50.72 call to the NRC informing us of degradation which has been discovered during disassembly and inspections of the Unit 2 inside recirculation spray pumps. The disassembly and inspections were scheduled during the current Unit 2 outage due to similar degradation of the same pumps on Unit 1 which was identified in June 1988. Additional followup of this area is discussed in paragraph 6 of this report.

POTENTIAL GAS BINDING OF HIGH PRESSURE SAFETY INJECTION PUMPS

On October 12, 1988, the licensee reported, pursuant to the requirements of 10 CFR 50.72, that an evaluation had identified a gas accumulation in the suction of the high pressure safety injection pump (HPSI) that could possibly gas bind the pump during a loss-of-coolant accident. The report further stated that vents will be installed on the high points during the present outages. The cause of this was attributed to a design deficiency of the system.

The resident inspectors, accompanied by their section chief, toured the piping spaces and noted the high points in question. The licensee discovered this situation during a review in response to NRC Information Notice IN 88-23, dated May 12, 1988. The internal response to this IN was assigned to a system engineer who requested that the site nondestructive test group ultrasonically inspect selected high points for absence of water. The results of the ultrasonic testing were compiled and transmitted to the station engineer on August 23, 1988, indicating actual voids did exist in the piping. The engineer concluded in a memorandum to his supervision, dated August 29, 1988, that "the operability of the HPSI pumps during an emergency is in question".

The above memorandum stated that the worst known pipe voiding at the time of the examination was in the area of Unit 2 LCV-2115B and LCV-2115D. These valves open to provide a flowpath from the refueling water storage tank (RWST) to the suction of the HPSI pumps. The approximate volume of this void was stated to be 3.8 cubic feet, thus considerably greater than the 2.2 cubic feet that could be safely passed as stated in a telephone conversation by the pump vendor (Byron Jackson). The memorandum went on to state that the installation of vents is recommended to ensure that the HPSI pumps are provided with a full suction in all situations.

The exact size of the gas voids cannot be determined from the data obtained. The engineer stated that he suspended any further examinations as soon as he was convinced that the pumps could not safely pass the known amount of gas. The licensee has decided not to pursue a determination of an acceptable amount of voiding and instead is proceeding with the installation of high-point vents.

The question of pump operability was raised on August 29 in the engineer's memo and copies were addressed to several tiers of supervision. No deviation report was prepared by the engineer when he identified the potential problem; nor did any of the supervision identify the problem in the licensee's corrective action program. It was not until the work to add the vents was discussed in a scheduling meeting at the assistant station management level on October 12, that the significance of the issue was recognized.

10 CFR 50, Appendix B, Criterion XVI, states that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined, and corrective action taken to preclude repetition. The identification of significant conditions adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to the appropriate levels of management. These requirements are implemented by the licensee's QA Topical Report VEP 1-5A. That report states, in part, that adverse conditions significant to quality, the cause of the conditions, and the corrective action taken are reported to appropriate levels of both offsite and onsite management by the use of a deviation report. The topical report also requires review of each deviation report for reportability of the condition to the NRC. The topical report

requirements are implemented at the station by Surry Power Station Administrative Procedure SUADM-O-12, Operations Department Notifications. This procedure defines a deviation as a significant difference between the expected value or conditions and the actual value or condition. It further requires that the shift supervisor shall be informed of all deviations, or nonconformances which may be deviations and also requires the individual identifying the deviation to complete a deviation report. The failure to promptly identify a deviation to the shift supervisor and prepare a deviation report on August 29, 1988, that potential gas binding may adversely effect the operability of the high head safety injection pumps is identified as a violation (280,281/88-41-01).

Station management expressed their concern that the above item was identified on August 29, 1988, and that it was not properly evaluated at that time. Unit 1 continued to operate until September 14, when it was shutdown for EDG concerns, and Unit 2 operated until the refueling outage that began September 10. This problem could potentially have resulted in all the high head safety injection pumps being unable to perform as required during an accident.

The inspectors routinely review each station deviation report and have noted a large increase in the numbers of reports submitted. The station deviation report submitted on the spent fuel pool door does indicate an increased awareness to identify and act on safety concerns. The licensee has taken some interim corrective actions, such as reading each station deviation report in the daily management meeting. This process will continue until a formal program can be developed to adequately identify and evaluate safety concerns. This station has historically addressed discrepancies and concerns in a somewhat informal manner, with no formal mechanism in place, for example, to generate written justification for continued operations (JCOs). The inspectors are continuing to evaluate the effectiveness of a program to identify and evaluate safety concerns as they arise.

CONTAINMENT SPRAY NOZZLE BLOCKAGE

On October 19, 1988, a station deviation report (S1-88-1157) was submitted identifying several spray nozzles in the Unit 1 containment that were covered with tape (tape wrapped around the nozzle). The shift technical advisor subsequently performed an inspection of all the spray nozzles and documented the results via a memorandum dated October 19, 1988. The results identified eight nozzles covered with tape and several nozzles oriented incorrectly. The inspectors discussed the results of this inspection with the station staff and monitored the station evaluation and corrective actions.

STATION TAGGING PROGRAM

On October 24, the inspectors discussed with the superintendent of operations the findings and reply to a licensee quality assurance audit (Report S88-24) regarding the station tagging program. Based on concerns from a previous INPO audit, management requested the Quality Assurance (QA) audit that was conducted during the Spring 1988, Unit 1 refueling outage. The QA audit report identified a total of nine findings and ten observations relating to failures to follow procedures, inadequate procedures, and a lack of attention to detail. The specific details of this audit were presented to the inspectors by the Quality Assurance Manager.

The findings and responses seem to indicate a tagging system that is basically sound, but confusing at times in that it does not adequately handle the abnormal tagging situations that arise during a major outage. The concept of using blanket tagouts to isolate entire systems and include all the work under this single tagging order was introduced prior to the Spring 1988, refueling outage.

Although the audit results did not identify major problems with tagging and isolations in the field, it is evident that some specific details are needed in the area of blanket tagouts to correct the confusion. The licensee agreed with this observation and stated that plans are being implemented to install a computerized tagging system that would improve on the method of establishing isolation. The inspectors will continue to monitor the licensee actions on this subject as part of the routine inspection program.

Within the areas inspected, one apparent violation was identified.

6. Maintenance Inspections (62703)

During the reporting period, the inspectors reviewed maintenance activities to assure compliance with the appropriate procedures.

INSPECTION AND REPAIR OF INSIDE RECIRCULATION SPRAY PUMPS 2-RS-P-1A & B

During this inspection period, the inspectors monitored the work associated with the disassembly, inspections, and repair of the Unit 2 inside recirculation spray (IRS) pumps 1A and 1B. The inspectors observed selected disassembly evolutions of the pump casings in the shop and noted the following:

- 2-RS-P-1A had indication of damage due to wear ring rotation of the casing wear ring and the first stage impeller upper wear ring between the fixed ring support area and the rings, respectively. Also, the first stage impeller could not be removed from the shaft as designed and showed rotational movement between the shaft keyway and the impeller keyway. Additional internal parts including shaft sleeve snap rings, impeller lock collar bolts, and lock wire were either loose or missing.

- 2-RS-P-1B had indication of damage due to wear ring rotation of the casing wear ring between its fixed ring support area and the ring. Also, the pump had damage associated with failure of a stabilizing bearing sleeve. Parts of the sleeve appeared to have passed through the pump internals during some period prior to disassembly. Some parts were also found in the pump operating sump after the pump had been removed from its normal location.

The licensee's inspection results generally agreed with that of the inspectors. The pumps were disassembled in the presence of a field service representative from the pump vendor, Bingham. This representative concluded that although the pumps were degraded, they were still operable and capable of producing a pressure and flow; however, no estimate was provided with regard as to how long the pump(s) would run in their "as found" condition.

The inspectors will continue to monitor pump repairs and testing as part of the regular inspection program.

Within the areas inspected, no violations or deviations were identified.

7. Surveillance Inspections (61726)

During the reporting period, the inspectors reviewed various surveillance activities to assure compliance with the appropriate procedures as follows:

- Test prerequisites were met.
- Tests were performed in accordance with approved procedures.
- Test procedures appeared to perform their intended function.
- Adequate coordination existed among personnel involved in the test.
- Test data was properly collected and recorded.

Inspection areas included the following:

TESTING OF THE UNIT 1 REACTOR CAVITY SEAL

On October 15, 1988, the inspectors witnessed testing of the J-Seal portion of the reactor cavity seal assembly for Unit 1. The test was being conducted in accordance with Special Test ST-224, Operability Reactor Cavity J-seals dated October 11, 1988. The purpose of the test was to verify the ability of the reactor cavity J-seals to perform their intended function by comparison of actual leakage with expected values. This test was accomplished by installing the reactor cavity seal assembly in the Unit 1 containment, flooding the cavity to different specified levels above the reactor vessel flange, deflating the inflatable seal at the specified levels, and monitoring for leakage past the J-seals. Potential leakage rates were specified in the procedure past the J-seal with maximum leakage anticipated at 26'- 6" to be approximately 150 gpm.

The inspectors reviewed a copy of the test procedure, the radiation work permit associated with the test, and attended the pretest briefing on the morning of October 15. The inspector witnessed the initial raising of cavity level from inside the Unit 1 containment and independently verified leakage from the J-seals to be less than 0.05 gallons per minute at the first test point (1'- 6"). The inspector then exited containment and observed filling of the cavity to the 16'- 0" level from the control room. The inspector noted that the licensee was still having communications problems due to having to use hand held radios in the noisy containment while wearing respirators. The inspector continued the monitoring of the test from the control room until J-seal leakage at the second test point (16'- 0") was determined. That leak rate was determined to be less than 0.10 gallons per minute. The test was completed on October 17, when the leakage past the J-seal at the third test point (26' - 6") was determined to be less that 0.30 gallons per minute.

After completion of testing and approval of the test results by the station safety committee, the inspectors reviewed the completed test procedure. Several small discrepancies were noted and identified to the licensee. However, the inspectors consider that the completed test procedure did adequately document testing of the cavity seal.

CONTROL ROOM & EMERGENCY SWITCHGEAR ROOM VENTILATION

The inspectors followed testing and evaluation of the ventilation that provides cooling to the control room and emergency switchgear room (ESR). The licensee documented via station deviation report (S1-88-937) dated September 9, 1988, that this ventilation system can no longer maintain normal design room temperatures when operating in the designed configuration. Previous operating experience has proven that the operation of two chillers and both trains of air handling units has been required to maintain acceptable room temperatures. The system was designed to maintain the control room at 75 degrees F and 50% relative humidity during either normal or emergency conditions, with the ESR maintained at 80 degrees F and 40% relative humidity during normal operations, and 87 degrees F and 35% relative humidity during emergency operations.

The system consists of three chiller units with three service water pumps (1-VS-P-1A, B & C) that supply cooling to each chiller condenser. Each chiller has a chilled water pump (1-VS-P-2A, B & C) that circulates chilled water to a set of air handling units. The air handling units utilizes a fan to pass air over the chilled water coils and out a system of ductwork. A normal train consists of a chiller providing water to one of two sets of air handling units. The third chiller functions as a swing chiller to supply either train. Each train in itself should be capable of maintaining design temperatures.

The station deviation report referenced above concluded that the ability of the existing equipment to perform its intended function is indeterminate at this time based on the following:

- The actual chiller capacities were significantly less than the design/procurement documentation. This issue was addressed in inspection report 88-12 as unresolved item 88-12-01. The upgrade of the chiller motors to meet original design capacities was completed in May 1988.

- The emergency switchgear room heat loads have increased by the installation of new electrical equipment over the years. It is apparent that the design change process did not track and account for the additional of heat loads.

- The material condition of the Heat, Ventilation/Air Conditioning (HVAC) equipment has significantly degraded since original installation.

The licensee performed special test ST-220, Control Room Envelope Air Conditioning System, to record data for determining the ability of the system to perform as designed. Prior to performing this test, the air handling units were cleaned to the maximum extent possible. The filters in these units have traditionally been routinely replaced, but the fans and cooling coils were found to be extremely dirty. The construction of these units precludes easy access to the fans and coils for routine cleaning. The service water valves that modulate and short-cycles service water through the chiller condensers were removed and the piping blanked to maximize chiller performance. The licensee stated that these valves were known problems and would be repaired prior to declaring the system fully operable.

The results of the above test were still under development as the inspection period ended. A preliminary evaluation of the raw data was performed by the resident inspector and confirmed the licensee initial findings that the chilled water flow was less than the design value and the air flow through the air handlers did not meet design criteria. The inspector considers the collection of data to be adequate.

On November 2, 1988, the licensee's evaluation of the test data had progressed to a point that it was determined that the capacity of the chiller system is inadequate to maintain design room temperatures during a loss of coolant accident with a loss of offsite power. In addition, the non-safety related central chiller system, which is to be used as a backup during certain high energy line break and Appendix "R" scenarios, appears to be inadequate to perform as required. This information was reported to the NRC as a 4-hour, non-emergency call.

The resident inspectors attended a briefing of station management near the end of the inspection period. In that meeting it became apparent that the capability of the control room envelope ventilation system to perform its design function as specified in the UFSAR was questionable as far back as two years ago. However, the information that the Architect/Engineer (A/E) provided with regard to calculated base heat load was unrealistic which resulted in the present licensee actions to confirm actual system loading. This process did not include identification of the suspect condition by the station corrective action program (writing of a deviation report) until September 9, 1988. Discussions with engineering personnel involved in the ventilation upgrade program indicated that the recent sensitivity to documentation of deviations after the Unit 1 reactor cavity seal event resulted in the preparation of the deviation report. The failure to identify the control room envelope ventilation potential problem at the time information was available to question the capability of the system is an additional example of violation 280,281/88-41-01. The resident inspectors will continue to monitor the effort to correct this problem since it is identified as a requirement for unit restart.

EMERGENCY DIESEL GENERATOR SPECIAL TESTING

On October 22, the inspector witnessed testing of the No. 2 Emergency Diesel Generator (EDG) in accordance with Special Test ST-225, EDG(S) Load Reject Testing. The purpose of the test was to evaluate the emergency diesel engine governor transient response capability by instantaneously reducing load on the generator. The test required that the No. 2 EDG be loaded to approximately 2.75 MWe on the 2H emergency bus and then disconnect the load from the EDG by opening its output breaker to the emergency bus. The test was conducted satisfactorily and resulted in minimal change in speed when the output breaker was opened. No discrepancies were noted.

On October 24 and 25, the inspectors witnessed testing of the number 3 EDG in accordance with Special Test ST-227, Emergency Diesel Generator (EDG) NO. 3 Load Sequence Test. The purpose of the test was to obtain the voltage and frequency response of the isolated emergency diesel generator subjected to load profiles which bound worst case scenarios. The results of these tests will be analyzed to determine the required number of load blocks and the times between each load block to ensure that the diesel can start and accelerate all loads. The design considerations are addressed in paragraph 9 of this report. The test required that the associated emergency bus be loaded with the required pumps and resistive load bank to simulate the initial load block to which the EDG would be subjected. The offsite source breaker to the bus is then opened, and the EDG breaker will connect the EDG to the bus if the EDG is already running; or, if the EDG is not running, the diesel will start, obtain rated speed and voltage, and will then connect to the bus. Additional loads will be added to the bus as directed in the test to evaluate their effect on the diesel. The inspectors witnessed both types of tests. No discrepancies were noted.

Within the areas inspected, one additional example of a violation identified in paragraph 5 was noted regarding the adequacy of control room ventilation system.

8. Licensee Event Report (LER) Review (92700)

The inspectors reviewed the LER's listed below to ascertain whether NRC reporting requirements were being met and to determine appropriateness of the corrective action(s). The inspector's review also included followup on implementation of corrective action and review of licensee documentation that all required corrective action(s) were complete.

(Closed) LER 280/88-05, Inoperable Heat Tracing Due to Inadequate Procedures. The issue involved identification on inoperable technical specification heat trace circuits due to inadequate surveillance and maintenance procedures. The concern of operability of the heat trace was initially identified by the NRC resident inspectors and resulted in a violation. Corrective actions included extensive maintenance and calibration of the heat trace system. Additional corrective actions included operator walkdowns of the associated local panels each shift. The inspectors verified that corrective actions were implemented. This LER is closed.

9. Design Changes and Modifications (37700)

During this inspection period, the inspectors selected several design change packages which were being implemented for review. These changes were:

EMERGENCY DIESEL GENERATOR MODIFICATIONS

On October 11-14, 1988, an inspector from the Region II Office was on site to review the purpose of and procedure for special tests to be conducted on the emergency diesel generators. While reviewing Information Notice 85-91, Load Sequencers for Emergency Diesel Generators, the licensee identified that a problem similar to that described in the Notice existed at Surry Power Station. At Surry, should a loss of offsite power (LOOP) occur before or simultaneously with a loss of coolant accident (LOCA), accident mitigation equipment will be sequenced onto the diesel generators (D/G) in blocks that can easily be accepted by the D/G. The resulting consequences is that limiting safeguards or safety injection signals clear the emergency bus and initiate load sequencing. However, should a LOOP occur after a LOCA, the bus is not cleared. When the D/G output breaker closes onto the bus, the D/G sees an instantaneous load change of greater magnitude than it is designed to handle. This conclusion is based on the 12.5 Mega Volts/Amps (MVA) ultimate step increase limit given on the manufacturer's Dead Load Capacity Curve ACD 67-41. Motor start MVA values at full voltage given on the motor data sheets were summed and the total was compared with the 12.5 limit. Motor start Mega Watt (MW) values were determined by multiplying the start MVA by the start power factor (from the motor data sheets). Since this calculation showed that the 12.5 MVA limit was exceeded, it was postulated that both D/Gs on the accident unit may fail for the LOOP after LOCA scenario.

A study was performed by the licensee's Nuclear Engineering Department (NED) to define the problem and propose solutions. After this study was complete, a team of engineers from Stone & Webster Engineering Corporation were brought to the NED office in Innsbrook, Virginia, to independently review the problem definition and proposed solution. They also re-established the relevant original design basis. Meetings with engineers from the generator vendor, Morrison-Knudsen Company, were held to help resolve the problem. Engineers from other utilities were also consulted. Unit 1 was shutdown in September 1988, as a result of the identified problem. Unit No. 2 had been shutdown earlier in September 1988, for a refueling outage.

Essentially the proposed solution was to install timing relays to sequence blocks of load onto the D/G for the LOOP after LOCA scenario. The timing sequence for this scenario must be faster than for the other scenarios in order to meet the accident analysis constraints. An objective was to make the timing sequence as fast as possible to obtain the best possible margin of safety.

The NRC inspector reviewed the 70 percent complete draft modification package at the site. The final package will include a complete safety evaluation wherein the proposed sequencing scheme (to be validated by test) is shown to be consistent with any accident analysis. Computer codes will be utilized in making this determination. The present sequencing scheme for the LOOP before LOCA scenario will not change.

The purpose of the diesel generator tests will be to demonstrate the D/G's ability to accept relatively large instantaneous load increases while maintaining acceptable voltage levels, thus validating the proposed sequencing scheme. The licensee believes that the ultimate short time Kilo Watt (KW) ability of the D/Gs may be limited by the turbo charger performance in the first few minutes of operation. The turbo charger may not achieve maximum efficiency with the relatively cool exhaust gases in the first few minutes of operation. The "Cold Load Capability" test will define this power limiting effect in terms of magnitude and time duration. It is expected that the effect will limit the D/G output to a level below the 30 minute rating of 2950 KW. After the exhaust air warms up the D/G output will increase; and the D/G will be tripped by the operator at 3000 KW. The "Transient" test will demonstrate the D/G's ability to handle the proposed load sequence scheme. The criteria for this test is that the motors will accelerate within the safe start times and that motor starters, or other relays, do not drop out as a result of the voltage dip. A transient analyzer recorder will be utilized to accept and process variables to be monitored. Key output variables will be profiles of generator output voltage and current. Also monitored will be frequency, KW, KVAR, and exciter terminal voltage. Obviously, motor controller center contactor status must be observed. The inspectors observed the "before modification" testing of the No. 2 and No. 3 emergency diesel generators. These test observations are discussed in paragraph 7 of this report.

In conclusion, the licensee has accurately defined the D/G loading problem, and has taken a complete and appropriate approach to its solution. The test procedure is valid, and should achieve the objective. If the test results are as predicted, the test will validate the proposed loading sequence scheme to be installed during the present outages. The inspector's comments as described herein were relayed to the licensee's management in an exit interview conducted on October 14, 1988.

Within the areas inspected, no violations or deviations were identified.

10. Exit Interview

The inspection scope and findings were summarized on November 9, 1988, with those individuals identified by an asterisk in paragraph 1. The following new items were identified by the inspectors during this exit.

One apparent violation (280,281/88-41-01) with four examples for failure to take appropriate corrective actions for identified deficiencies was noted as follows:

- Failure to promptly identify a deviation to the shift supervisor and prepare a deviation report on August 29, 1988, that potential gas binding may adversely effect the operability of the high head safety injection pumps (paragraph 5).
- Failure to adequately evaluate the adverse condition documented in station deviation S1-87-946 from November 20, 1987, to April 11, 1988, with regard to control room chiller capacity (paragraph 3).
- Failure to identify the control room envelope ventilation potential problem at the time information was available to question the capability of the system (paragraph 7).

  Failure to take appropriate corrective actions for a NRC identified violation with regard to inventory of special nuclear material which was discussed in inspection report 280,281/87-10 (paragraph 3).

The apparent example violations listed above indicates a weakness in past implementation of the licensee's corrective action program at the Surry Power Station.

One inspector followup item was identified in paragraph 5 for followup on licensee evaluation of technical issues identified during review of loss of PRT water event (280,281/88-41-02).

The licensee acknowledged the inspection findings with no dissenting comments. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.