Report Details

Summary of Plant Status

Unit 1 began the inspection period at full power. On August 27, the unit tripped from full power due to a rod control system failure (Sections 01.2 and 02.3). The reactor was restarted on August 28, and the unit returned to commercial service on August 29. The unit returned to full power on August 30 and remained at or near full power for the remainder of the inspection period.

Unit 2 began the inspection period at full power. On August 16, the unit began a coast down to a refueling outage. Late on September 7, a unit shutdown was commenced from 85 percent power, and early on September 8, the unit was removed from commercial service and the reactor was shut down for a scheduled refueling outage. On September 16, reactor defueling was completed, and the reactor remained defueled for the remainder of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Daily Plant Status Reviews (71707)

The inspectors conducted frequent control room tours to verify proper staffing, operator attentiveness, and adherence to approved procedures. The inspectors attended daily plant status meetings to maintain awareness of overall facility operations and reviewed operator logs to verify operational safety and compliance with Technical Specifications (TSs). Instrumentation and safety system lineups were periodically reviewed from control room indications to assess operability. Frequent plant tours were conducted to observe equipment status and housekeeping. Deviation Reports (DRs) were reviewed to assure that potential safety concerns were properly reported and resolved. The inspectors found that daily operations were generally conducted in accordance with regulatory requirements and plant procedures.

O1.2 Prompt On-Site Response to Events (93702)

a. Inspection Scope

On August 27, the licensee notified the inspectors concerning a Unit 1 reactor trip. The unit tripped from full power while operators were performing 1-OP-17.1, Control Rod Operability, Revision 17. The inspectors reviewed control room conditions and operations following the trip. Additionally, the inspectors attended the licensee's post-trip review and reviewed trip data to independently verify that safety system and operator performance was as expected throughout the event.

b. Observations and Findings

The inspectors found that the automatic trip was generated from the Reactor Protective System (RPS) when a nuclear instrument negative rate condition was detected. The trip occurred as operators began to move the B control rod bank inward in accordance with 1-OP-17.1. The inspectors observed that operator response following the trip was good, and procedures for responding to the trip were appropriately implemented. Although the inspectors observed that a large number of people were present in the control room soon after the trip, communications remained formal and the extra personnel were not distracting to the operators.

A review of trip data found that the trip signal was valid and was caused by one or more dropped control rods. All safety systems performed as designed for plant conditions during the trip. During the trip recovery, main condenser vacuum was lost. Operators were required to use the main steam atmospheric dump valves to control reactor coolant system temperature. Plant startup and subsequent corrective actions are discussed in Sections O1.3, O2.1 and M2.1.

c. Conclusions

The inspectors concluded that safety system and operator response to the Unit 1 reactor trip was proper.

O1.3 Unit 1 Restart Following Reactor Trip (71707)

a. Inspection Scope

On August 28, after a reactor trip the day before, the inspectors reviewed the licensee's corrective actions for the reactor trip's root cause. Additionally, the inspectors observed operators performing Unit 1 restart activities to ensure that operational activities were conducted in accordance with plant procedures and TS requirements.

b. Observations and Findings

The inspectors observed that the licensee's investigations into the root cause for the Unit 1 reactor trip found that a failure in the rod control system had caused a drop of one group of the B control rod bank when the rods were moved when performing 1-OP-17.1. Repair efforts identified a discrepancy in the signal from the rod control system logic cabinet to the power cabinet supplying the group An interface card was replaced, and the problem was corrected. However, the problem could not be duplicated by placing the suspect card back into the system. The investigation postulated that the suspect card had either intermittently failed or had broken the circuit at the card's edge connector. After replacing the suspect card, technicians completed current order traces for the entire system, and no other problems were found. The inspectors reviewed the current (amperage) order traces before and after the card changeout and verified the problem's identification and correction. Additionally, the licensee conservatively decided to perform rod drop timing tests prior to unit restart, although test performance was not required by TS or the licensee's response to NRC Bulletin 96-01, Control Rod Insertion Problems. The inspectors observed that tests were successfully completed (Section M1.2). Formal root cause evaluations were continuing at the inspection period's end. Additional reviews of restart issues are discussed in Section O2.1.

The inspectors observed operators performing the Unit 1 reactor startup and found that operators properly used appropriate procedures and were cautious and methodical during startup operations. No significant problems were encountered during the startup, and control room communications and formality were good. The inspectors noted that appropriate supervisory and management personnel, as well as Oversight personnel, were present to review startup activities.

c. Conclusions

The inspectors concluded that the licensee's actions to correct the cause of the Unit 1 trip and operator performance during reactor startup were appropriate. A conservative decision was made to perform rod drop timing tests prior to unit restart.

O1.4 Tropical Storm Fran Response (71707)

a. Inspection Scope

On September 5 and 6, the inspectors reviewed the licensee's response to severe weather from Tropical Storm Fran. The inspectors compared the licensee's response to plans for severe weather contained in the Hurricane Response Plan, the site Emergency Plan, and in 0-AP-41, Severe Weather Conditions, Revision 12.

b. Observations and Findings

The inspectors found that the licensee adequately reviewed the status of plant equipment and activities in anticipation of the severe weather. Prior to the storm's arrival, outside areas were inspected for loose equipment and debris. Outside efforts appropriately focused on areas temporarily staged with equipment in preparation for the Unit 2 refueling outage. Additionally, although entry conditions were not met, the Hurricane Response Plan was implemented as a precaution to ensure that all appropriate actions were taken.

During the storm, the inspectors found that the licensee appropriately monitored the storm's progress and remained alert for possible effects on the facility. 0-AP-41 was entered as required during tornado watches, and actions were taken to protect the facility from moderate winds and heavy rains. As a result of local power outages caused by the storm, power was lost to numerous emergency sirens (Section P1). No site damage was sustained due to the storm.

c. Conclusions

The inspectors concluded that the licensee properly prepared for and responded to severe weather caused by Tropical Storm Fran.

01.5 Unit 2 Shutdown for Refueling (71707)

a. Inspection Scope

On September 8, the inspectors observed operators placing Unit 2 in MODE 3 from MODE 2 in preparation for refueling. The inspectors reviewed activities to ensure operations were conducted in accordance with plant procedures, TS requirements, and commitments made in response to NRC Bulletin 96-01, Control Rod Insertion Problems.

b. Observations and Findings

The inspectors found that operators effectively controlled unit shutdown activities. Operating procedures were followed, and TS requirements were complied with during the mode changes. Communications during shutdown evolutions were good.

The inspectors observed that the unit was shutdown by manually tripping all control rods into the core, and control rod drop time tests were performed immediately following the shutdown. On September 16, the inspectors also observed rod drag tests being performed on approximately ten control rods in the spent fuel pit (Section M1.2). These actions met licensee's commitments made in response to NRC Bulletin 96-01. No problems were identified during the rod control system testing.

c. Conclusions

The inspectors concluded that operator control of plant shutdown activities was good, and the licensee complied with commitments made in response to NRC Bulletin 96-01, Control Rod Insertion Problems.

01.6 Control of Reactor Coolant System (RCS) Drain Down (71707)

On September 10, the inspectors observed RCS drain down activities including observation of the pre-job briefing and actual drain down to the 74 inches above mid-loop level. A thorough briefing was conducted which included an overview of the procedure, a review of drain down curves, expectations of instrumentation during drain down, communications, cautions, and specific personnel assignments. The licensee exhibited good sensitivity to risks associated with drain down. This was evidenced by excellent involvement between Reactor Operators and Senior Reactor Operators (SROs) and by careful procedural compliance. Several Shift Technical Advisors were also involved and accurately predicted when instrumentation would come on scale. The prediction for when level would be observed in the sight glass was within approximately one minute and for the control room instrument, which comes on scale at 95 inches above mid-loop, was within ten minutes. Two good initiatives were also noted. One was a rolling lit sign highlighting the time to boiling. Also, the licensee developed a recorder to do an inventory balance by measuring various tank levels involved in the drain down. The recorder did not calculate the total inventory as expected, but did record the data to do so. Also the recorder did not print out digital information as originally intended. The licensee indicated they intended to continue this initiative and modify the recorder as necessary to perform the calculation and to print the calculation's results. In summary, the RCS drain down was well coordinated and personnel exhibited a good sensitivity to risk.

O1.7 Unit 2 Refueling Activities (71707)

a. Inspection Scope

On September 14, the inspectors observed operators performing Unit 2 refueling activities to ensure that plant operations were conducted in accordance with procedures and TS requirements.

b. Observations and Findings

During core off load, the inspectors verified that operators in containment properly used procedures 2-OP-4.1, Controlling Procedure for Refueling, Revision 32-P2; 2-OP-4.13, Fuel Transfer System, Revision 1; and 2-OP-4.15, Manipulator Crane, Revision 2. Applicable TS requirements were reviewed and found to be complied with for operable equipment and for containment integrity. The inspectors observed that communications were properly established between all stations and that evolutions were supervised by a licensed SRO in containment. Also, the inspectors observed that fuel movements were being continuously tracked by a reactor engineer in the control room.

c. Conclusions

The inspectors concluded that refueling activities were properly conducted by operators.

O2 Operational Status of Facilities and Equipment (71707)

O2.1 Unit 1 Startup Reviews

a. Inspection Scope

On August 28, the inspectors attended a Station Nuclear Safety and Operating Committee (SNSOC) meeting to ascertain if problems following the August 27 Unit 1 trip were being adequately reviewed and resolved.

b. Observations and Findings

Prior to discussing the Unit 1 startup, the inspectors observed the SNSOC discussing a proposed revision to security procedure SPIP-12, Refueling/Major Maintenance Measures. The revision proposed changes to the requirements for security badge display to allow wearing badges inside Protective Clothing (PC) in containment. The inspectors observed that a long discussion was generated by the proposed revision. The discussion eventually led to members of the SNSOC proposing that the procedure should be further revised to delete the requirement for displaying badges in other situations and/or throughout the plant entirely. The inspectors observed that throughout the long discussions, neither the individual presenting the revision nor the SNSOC members displayed a knowledge of the actual requirements for badge display contained in the Physical Security Plan (Revision 0, dated April 1, 1996). After the discussions appeared to be moving towards referring the revision back to security for additional reviews, the inspectors informed the SNSOC members that several of their proposals did not meet the requirements of the Physical Security Plan and that they had failed to recognize this due to their apparent lack of knowledge of the plan's requirements.

The inspectors then observed the SNSOC consideration of the Unit 1 Post Trip Review Report. The report was presented to the SNSOC for approval following its completion by Shift Technical Advisors. As presented, the inspectors observed that the report accurately analyzed plant performance during the trip and corrective actions for the trip's cause. However, the report's analyses of secondary plant equipment problems encountered during the trip led to much discussion in the SNSOC meeting. In discussing these failures, the SNSOC determined that the report inadequately stated that a loss of secondary vacuum would be addressed through the DR closeout process and inaccurately stated that a reheat flow control valve, 1-MS-FCV-104C, had been repaired. The discussions then centered on deciding what corrective actions were necessary for the loss of secondary vacuum and for the failure of 1-MS-FCV-104C to shut following the trip. The SNSOC decided not to approve the report until additional corrective actions were taken for the loss of vacuum and until a safety evaluation was completed for 1-MS-FCV-104C being left in a degraded condition. Later that same day, the additional corrective actions were completed, and the report was approved by the SNSOC.

The inspectors found that the SNSOC adequately ensured that problems were being resolved prior to unit startup. However, the inspectors questioned station management concerning the appropriateness of the SNSOC directly making decisions about corrective actions to be taken for equipment problems. The inspectors expressed concern that such active decision making was more appropriate for station management discussions than for the SNSOC, which was intended to be a safety review authority. Licensee management stated that they believed the deliberations took the track they did because the Post Trip Review Report should have been more thoroughly reviewed prior to being presented to the SNSOC. Management stated that they expected that such problems would be identified and resolved by the Supervisor, Station Nuclear Safety (SNS) prior to the SNSOC presentation. The inspectors also noted that the Post Trip Review Report presented to the SNSOC had not been presented to the Superintendent, Operations for concurrence signature prior to its submission to the SNSOC. In the SNSOC meeting, both the Supervisor, SNS, and the Superintendent, Operations were seeing the report for the first time as SNSOC members.

Additionally, the inspectors found that VPAP-1404, Reactor Control, Revision 0, Section 6.8.1.a, stated that the decision to re-start a reactor after a trip shall be made in accordance with VPAP-2804, Start-Up Assessment. However, VPAP-2804 had never been issued although four years had elapsed since the time that VPAP-1404 had been issued (June 1992). This fact was noted in a supplemental page in VPAP-1404 which stated that until VPAP-2804 was issued, the "Start-up Assessment Review Package distributed by Nuclear Safety and Licensing" shall be used. The inspectors noted that the Start-up Assessment Review Package was a document normally used for startup following refueling outages and was not used for startups following reactor trips. The inspectors discussed this issue with licensee management who directed station personnel to complete VPAP-2804 as soon as possible.

c. Conclusions

The inspectors concluded that the SNSOC considered a security procedure revision without having an accurate knowledge of the applicable Physical Security Plan requirements. The SNSOC ensured that adequate corrective actions were completed prior to restart of Unit 1 following a reactor trip. However, a Post Trip Review Report was not fully reviewed prior to being presented to the SNSOC which required the SNSOC to depart from its normal review functions and plan and direct additional corrective actions. An administrative procedure formally delineating the startup assessment process following a reactor trip did not exist.

O2.2 Unit 2 Containment Conditions

On September 9, the inspectors entered Unit 2 containment shortly following the shutdown to ascertain the status of equipment following the long operating cycle. The inspectors found that material conditions and cleanliness were good. A few small system leaks were identified. The inspectors noted that these small leaks had already been identified by licensee personnel performing RCS leak identification surveillance tests. The inspectors concluded that safety systems in the Unit 2 containment were in good overall condition immediately following shutdown.

On September 18 and 19, the inspectors reviewed activities inside Unit 2 containment. These activities did not promote good housekeeping practices and potentially represented a challenge to personnel assigned to cleanup containment prior to restart. Specifically, the inspectors observed that cloth rags were on the pressurizer head in places where insulation had been removed; materials such as plastic bottle tops and work gloves were on top of supports and raceways below deck gratings; and, tools and miscellaneous debris were laid on supports or dropped on the floor. The inspectors observed no designated trash receptacles in use in areas where work was being performed on the lower two levels of containment. Housekeeping conditions and practices were discussed with plant management.

O2.3 NRC Notifications

a. Inspection Scope

The inspectors reviewed the following licensee notifications to the NRC to ascertain if the required reports were adequate, timely and proper for the events.

b. Observations and Findings

On August 27, the NRC was notified as required by 10 CFR 50.72 concerning RPS and engineered safety feature actuations generated when Unit 1 tripped from full power. The inspectors found that the licensee's reporting actions were appropriate. Additional inspection activities and findings are discussed in Sections O1.2, O1.3, O2.1 and M2.1.

On September 6, the NRC was notified as required by 10 CFR 50.72 concerning the notification of off-site authorities. Specifically, the licensee notified surrounding counties and individuals of flood warnings downstream of the Lake Anna Dam due to heavy rains from Tropical Storm Fran. The inspectors found that the licensee's reporting actions were appropriate.

On September 6, the NRC was notified as required by 10 CFR 50.72 concerning a loss of a significant portion of the offsite emergency notification system. Specifically, the licensee notified the NRC and the Commonwealth of Virginia concerning the fact that 28 of 55 emergency sirens were inoperable. The inspectors found that the licensee's reporting actions were appropriate. Additional reviews are discussed in Section P1.

On September 8, the NRC was notified as required by 10 CFR 50.72 concerning the identification of a condition which could have prevented the fulfillment of a system safety function needed for accident mitigation. While shutting down Unit 2 for a refueling outage, operators identified during a routine surveillance test that feedwater check valve, 2-FW-62, would not prevent backflow. The inspectors found that the licensee's reporting actions were appropriate. The licensee was required to submit a Licensee Event Report (LER) for this event, and the inspectors will review the licensee's corrective actions after LER issuance and prior to unit restart.

On September 10, the NRC was notified as required by 10 CFR 50.72 concerning a loss of emergency assessment capability lasting greater than one hour. At 6:52 p.m., the Safety Parameter Display System (SPDS) computer failed and could not be restored within one hour. Maintenance personnel successfully returned the SPDS computer to operation at 10:24 p.m. The inspectors monitored the licensee's reporting and corrective actions and found them to be appropriate for the situation.

On September 19, the NRC was notified concerning the identification of a condition which could have prevented the fulfillment of a system safety function needed for accident mitigation. Due to questions raised by the inspectors, the licensee identified a procedural deficiency that could have resulted in the dose to the control room personnel exceeding Fuel Handling Accident (FHA) analysis assumption. After further review the licensee retracted this notification on October 15, 1996. The inspectors reviewed the licensee's reporting actions and found that they were appropriate. Additional reviews are discussed in Section E2.1.

c. Conclusions

Five NRC notifications required by 10 CFR 50.72 were properly made by the licensee.

O4 Operator Knowledge and Performance (71707)

O4.1 Operator Error During Charging Pump Breaker Manipulation

a. Inspection Scope

On September 10, the inspectors reviewed an event in which all Unit 2 charging pumps were found to be simultaneously inoperable while the unit was in MODE 5.

b. Observations and Findings

At approximately 12:35 p.m. on September 10, operators attempted to start charging pump 2-CH-P-1C and received a breaker disagreement light. At the time, 2-CH-P-1C was considered to be the only pump operable to meet TS requirements. Pump 2-CH-P-1A was available, but its emergency power supply, the 2H Emergency Diesel Generator, was inoperable. Pump 2-CH-P-1B was unavailable due to a tagout. Operators dispatched to the breaker identified that the charging motor power control switch was in "off", and the breaker springs indication showed that the springs were discharged. Operators placed the power control switch in "on", but the springs still did not charge as expected. The pump was declared inoperable and TS 3.1.2.1 and 3.1.2.3 action statements were reviewed and complied with. The inspectors responded to the control room shortly after identification of the problem and verified that TS action statement compliance was maintained throughout the time that all charging pumps were inoperable.

Initial investigations into the problem found that the breaker was racked into the cubicle earlier the same day during an evolution to swap the pump power supply from the H to the J bus. Pump 2-CH-P-1C was declared as the only operable pump after racking in the breaker at 5:25 a.m. The operator performing the evolution completed procedure 0-OP-26.9, 4160-volt Breaker Operation, Revision 9, including initialing steps indicating that the switch was "on" and that the springs indicated charged. In accordance with the procedure, since repositioning the switch was not required, no independent verification of switch position was required. After finding and verifying that the breaker configuration was incorrect, the licensee concluded that the operator erred when signing off the steps. During interviews, the operator informed his supervision that he did not remember the details concerning procedure performance.

The licensee also investigated the reason for the springs failing to charge when the switch was re-positioned to "on". The breaker was found to have an intermittent problem which was preliminarily determined to be caused by a slight misalignment in the internal relay switch crank. The breaker was replaced with a spare breaker which operated satisfactorily. The pump was returned to operable status at 3:07 p.m.

The inspectors found that the operator's error led to the plant operating without any charging pumps. Although the breaker had an intermittent mechanical problem which prevented the springs from charging, the operator error was a missed opportunity to identify the problem. As corrective action, the operator was disciplined in accordance with the licensee's programs, and the procedure was enhanced to require independent verification for the power control switch.

The inspectors reviewed the procedural requirements for the evolution. Unit 2 TS 6.8.1 requires that written procedures be established implemented and maintained, including by reference to Appendix A of NRC Regulatory Guide 1.33, Revision 2, procedures for operation of the chemical and volume control systems. This requirement was implemented in part by 0-OP-26.9, Section 5.2.5, which required operators racking in 4160-volt breakers to verify the power control switch in "on" and to verify that the breaker springs are charged. Contrary to this requirement, the operator did not verify the power control switch in "on" or the breaker springs charged. As a result, the only available Unit 2 charging pump was unknowingly inoperable for approximately seven hours. This licensee-identified and corrected violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-339/96009-01).

c. Conclusions

An NCV was identified for an operator's failure to follow procedures when racking in a charging pump breaker. As a result, the only available Unit 2 charging pump was unknowingly inoperable for approximately seven hours during shutdown conditions.

O7 Quality Assurance in Operations (40500)

O7.1 Oversight Meeting

On September 18, the inspectors met with Oversight personnel. Issues discussed included Oversight activities and findings since previous meetings. Copies of recent audits were provided for review. The inspectors also observed Oversight personnel observing plant activities on numerous other occasions during the inspection period and met briefly with them to discuss their observations. The inspectors concluded that the Oversight organization was continuing to assess station performance effectively.

08 Miscellaneous Operations Issues (92901, 92700)

08.1 <u>(Closed) Unresolved Item (URI) 50-339/96007-01 (EA 96-292): Review Compliance With 10 CFR 50.54k Requirements For Operator Presence at Unit Controls</u>

a. <u>Inspection Scope</u>

On August 5, the inspectors were informed by licensee managers that a licensed operator had inadvertently left the Unit 2 controls area for a short time period. During this inspection period, the inspectors reviewed the event's details, the licensee's administrative controls over Operator at the Controls (OATC) movements, and compliance with regulatory requirements.

b. <u>Observations and Findings</u>

The inspectors ascertained the following event details through interviews with operators and management:

- On August 5, at approximately 3:30 p.m., the Backboards Operator properly relieved the OATC for a meal break. The individual carried the Backboards Operator's portable phone to the OATC area.

- Approximately two minutes after relief, the OATC made a personal phone call using the Backboards Operator's portable phone.

- During the OATC's call, the other party requested a phone number for calling back, and the OATC proceeded to the portable phone base unit at the Backboards Operator's desk to retrieve the phone number. The Backboards Operator's desk was a few feet outside the OATC's designated work area. It was estimated that the individual was away from the OATC's work area for less than five seconds.

- The OATC's actions were observed by an observer who had entered the control room to gather information related to a licensee self-assessment activity. The observer informed the Unit SRO concerning the OATC's actions. The Unit SRO had not directly observed the OATC's actions due to involvement in discussions with maintenance personnel.

- After the problem was identified, the OATC was relieved of the OATC and Backboards Operator duties pending further investigations. Deficiency Report N-96-140 was initiated to identify and track corrective action.

- The next day, August 6, the individual was directed to report for Fitness for Duty (FFD) chemical testing. FFD chemical testing results were later found to be negative. The individual's access to the protected area and qualification for licensed duties were temporarily suspended pending further corrective and disciplinary action.

On August 20 and 21, the inspectors discussed these findings with licensee management. The inspectors were informed that the licensee concluded that the individual's actions were negligent of duty and were probably influenced by outside personal problems. Individual disciplinary actions were taken in accordance with the licensee's established program.

Additionally, the inspectors discussed with licensee management administrative requirements and policies regarding conducting personal business while at the controls. Procedure OPAP-0007, Control Room Activities, Revision 6, Paragraph 6.2.3, stated, "Only activities essential to supporting station operation should be conducted in the control room." Further, Paragraph 6.2.4 stated, "Non-job-related discussions should be minimized to prevent any possible interference with conduct of the shift or monitoring of station parameters." No clear requirements were identified prohibiting the placement of personal calls by OATCs. Station management stated that the individual's actions in this case were clearly unacceptable, but that there were instances where personal phone calls to or from the OATC would be acceptable (e.g., urgent or emergency situations). Management indicated that policies for phone use while on duty were being reviewed, and additional guidance would be issued. Management also informed the inspectors of additional actions which were being taken to evaluate on-shift OATC activities to ensure that the event did not reflect a widespread problem with inappropriate activities being performed by OATCs.

The inspectors reviewed the licensee's controls over OATC movements. To ensure that an operator remained present at the controls, the licensee had an established program for controlling OATC movements delineated in OPAP-0007. Procedure OPAP-0007, Section 6.3.6 and Attachment 1, defined three areas for operator movements: the work area, the limited time area, and the restricted work area. The OATC was allowed to leave the work area and enter the limited time area only to acknowledge annunciators, initiate corrective actions, to obtain reactor coolant pump vibration readings, or to obtain drawings. The OATC was not allowed to enter the restricted work area unless properly relieved.

The inspectors found that the OATC's movements during the event took him out of the designated work area through the limited time area and approximately two feet into the restricted work area. There were no valid reasons for the OATC to enter the limited time area, and a proper turnover was not conducted before the OATC entered the restricted work area. During the time the OATC was absent from the work area, no other licensed operators were present in the work area. The nearest other licensed operator was the Unit 2 SRO who was present at the SRO desk a few feet outside and overlooking the OATC work area.

10 CFR 50.54k requires that licensed operator be present at the controls at all times during facility operation. This requirement is implemented in part, by OPAP-0007, Section 6.3.6, which designates areas where a licensed operator must be present during facility operation. Contrary to this requirement, a licensed operator was not present in the designated Unit 2 controls area for a short time period. This licensee-identified and corrected violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-339/96009-02).

c. Conclusions

An NCV was identified involving a licensed operator not being present at the Unit 2 controls for a short time period contrary to 10 CFR 50.54k requirements.

O8.2 (Closed) LER 50-338/96005: Reactor Trip on High Negative Flux Rate

This LER discussed the August 27 Unit 1 trip from full power due to a rod control system failure. The licensee's response to the event and corrective actions for the associated equipment failures were reviewed and are discussed in Sections O1.2, O1.3, O2.1, O2.3 and M2.1.

II. Maintenance

M1 Conduct of Maintenance

M1.1 2H Emergency Diesel Generator (EDG) Post-Maintenance Inspection and Testing

a. Inspection Scope (62707)

On September 18 and 19, the inspectors observed inspection activities and testing performed on the 2H EDG.

b. Observations and Findings

During the routine refueling outage inspection of the 2H EDG, the licensee identified that the top and middle compression rings on number seven lower piston were worn. All three compression rings were replaced on this piston. The inspectors observed the 2H EDG operate during the break-in period and witnessed visual examination of new rings by an engineer and the vendor representative after the 2H EDG had operated at 65 percent load.

After the 65 percent load run, the 2H EDG exhaust manifold for the number seven cylinder was removed to allow inspection of the new rings. The inspectors observed that the number seven cylinder walls and lower piston skirt were less lubricated that the other cylinders. This was discussed with the engineer and vendor representative. They also examined the upper piston for proper lubrication. They concluded that there was no problem.

After the 100 percent load run, the inspectors again looked at the number seven cylinder. This time the oil coating appeared normal. The inspectors had no further concerns in this area. Based upon operating data, including crank case vacuum, taken during the break-in runs and the licensee's and inspectors' examinations, there were no indications of a problem with the new compression rings.

c. Conclusions

The 2H EDG post-maintenance inspections and testing were adequate to demonstrate that the new compression rings on the number seven lower piston were functioning correctly.

M1.2 Control Rod Surveillance Observations (61726)

a. Inspection Scope

On August 28 and September 8 and 16, the inspectors observed technicians and engineers performing 1/2-PT-17.2, Rod Drop Time Measurement, Revision 13/12-P1. The test was performed, on Unit 1 prior to restart following a reactor trip and on Unit 2 immediately following a shutdown for a refueling outage, to meet the licensee's commitments made in response to NRC Bulletin 96-01. In addition, the inspectors observed control rod drag testing of Unit 2 discharged fuel assemblies.

b. Observations and Findings

The inspectors observed that technicians properly followed procedures in obtaining the rod drop traces. Formal communications between the control room, the relay room, and the rod drive room were maintained throughout the repetitive rod drop sequences. The inspectors reviewed a sampling of rod drop traces and verified that measurements were being accurately recorded and that the traces were correctly shaped. The inspectors also observed that reactor engineers were reviewing the traces to verify that rod recoil was recorded. The inspectors reviewed the test procedure and verified that the licensee was meeting commitments made in response to NRC Bulletin 96-01. The test results were satisfactory for all rods.

On September 16, the inspectors observed drag testing of the final eight control rods being tested from the Unit 2 off-loaded core. In addition, the data for the other 40 control rods and fuel assembly pairs were reviewed. The maximum force required to move the control rod through the dash pot region and the area just above this region was recorded. All measured values were 40 lbs or less which was well below the 100 lb acceptance criteria.

c. Conclusions

The inspectors concluded that rod drop timing tests were properly performed for both units. The inspectors also verified that the Unit 2 control rod drag testing results were well within the acceptance criteria.

M1.3 Large Bore Snubber Testing (61726)

a. Inspection Scope

During the Unit 2 refueling outage, the first two large bore snubbers functionally tested failed to lockup in one or both directions. The inspectors observed subsequent test activities, reviewed test procedures, previous test records, procurement documents, specifications, engineering test reports, and TS requirements.

b. Observations and Findings

On September 15, 1,000 kip snubber 2-RC-HSS-11A failed to lockup during both the compression and tension activation tests conducted in the field. On September 16, 1,900 kip snubber 2-RC-HSS-01A failed to lockup during its field compression activation. The inspectors observed the bench test of snubber 2-RC-HSS-11A. Although the snubber locked up in both directions during the bench test, the acceptance criteria was not met. The speed at which lockup occurred was approximately 28 inches per minute as compared to a maximum allowed 20 inches per minute. A spare snubber replaced this snubber in Unit 2.

Testing was subsequently performed on other installed and spare snubbers with various results. However, the licensee noted that more recent tests were more likely to have acceptable results. Thus, except for snubber 2-RC-HSS-11A, it was hypothesized that earlier test results may have resulted from problems with the test apparatus or testing technique.

On September 19, the inspectors observed a spare 1,900 kip snubber, designated as spare 06, being field tested. The snubber met the acceptance criteria and performed satisfactorily.

Subsequent to the report period, snubber 2-RC-HSS-11A was disassembled and found to contain metal shavings in the hydraulic fluid. These metal shavings apparently caused this snubber's improper functioning. Based on this result, all 12 large bore snubbers installed in Unit 2 were tested. The inspectors were informed that the data from these twelve test, as well as, test results taken during previous outages were reviewed by engineering and the results presented to the SNSOC prior to restart of Unit 2.

The inspectors' review of data taken during previously outages indicated that snubber 2-RC-HSS-001B in 1992 and 2-RC-HSS-001C in 1993 did not clearly demonstrate that lockup had occurred during compression activation testing. An Unresolved Item (URI) is being identified to perform additional reviews of the anomalies observed in large bore snubber test data (50-339/960009-03).

c. Conclusions

The licensee functionally tested all 12 large bore snubbers installed in Unit 2 and considered that they were operable. An URI was identified to review anomalies in large bore snubber test data taken early in this refueling outage and during previous outages.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Equipment Problems Following Unit 1 Reactor Trip (62703, 62707)

a. Inspection Scope

On August 27 and 28, the inspectors reviewed conditions following a Unit 1 reactor trip to ensure the licensee was taking appropriate actions to identify and resolve equipment problems.

b. Observations and Findings

The inspectors found that after the trip, the following significant equipment problems occurred:

- Main Feedwater Regulating Valves (MFRVs) leaked by the seats. Operator action was required to isolate two of the three MFRVs.
- Steam dump 1-MS-TCV-1408B indication limit switch failed. Operators were momentarily concerned that the dump was not opening, and operator action was required to locally verify that the dump was properly responding.
- Extraction steam flow control valve 1-MS-FCV-104C failed to close. Operator action was required to manually isolate the valve to limit RCS cooldown.
- Condenser vacuum was lost due to inadequate turbine gland sealing steam. Operators were required to use the steam generator atmospheric relief valves to control RCS temperature.
- Several secondary reliefs lifted and several failed to properly reseat. Operators were required to locally isolate various components to reduce the flow of water into the turbine building sumps.

The inspectors verified that all the above problems were appropriately resolved prior to unit restart. The inspectors noted that concerning the loss of condenser vacuum, operators had previously opened the gland steam header dump bypass valve on both units in order to reduce gland steam pressure at full power. The pressure was postulated to be high due to excessive leakoff from the turbine valve glands. This configuration required operator action on a trip to close the bypass valve to maintain gland steam header pressure. This had been identified as an Operator Work Around (OWA) and added to the licensee's OWA list on June 11, 1996, as a low priority (category C) OWA. However, trip response or other procedures had not been modified to alert operators concerning the need to shut the dump bypass valve following a trip.

c. Conclusions

The inspectors concluded that following the Unit 1 reactor trip, five secondary plant equipment problems occurred requiring operator compensatory actions.

M4 Maintenance Staff Knowledge and Performance

M4.1 Foreign Material Exclusion (FME) Controls (71707)

On September 18, the inspectors observed that the limit switch compartment cover for 2-SI-MOV-2865A, the Unit 2 A accumulator discharge isolation valve, was not installed. Approximately ten feet away (over to one side and above the valve), an individual was grinding on a support for the instrument tubing for this accumulator. This was called to the attention of a nearby Health Physics (HP) technician who stopped the work until the valve switch compartment could be covered. The inspectors subsequently observed personnel performing "VOTES" testing on another accumulator discharge valve and discussed the missing cover with them. The individuals indicated that they had removed the cover earlier in the day to do testing and were not aware of the grinding work in the area. The individuals returned to 2-SI-MOV-2865A and replaced the cover. This item was discussed with Maintenance supervision as representing a lack of sensitivity to FME issues and as a potential for equipment degradation. The licensee indicated that the valve would be checked out prior to its return to service.

III. Engineering

E2 Engineering Support of Facilities and Equipment (37551)

E2.1 Seismic Concerns Regarding Containment Particulate and Gaseous Radiation Monitors (RMs)

a. Inspection Scope

The inspectors reviewed concerns raised by the licensee involving the seismic qualifications for the containment particulate and gaseous RMs to ascertain if the licensee complied with equipment operability requirements.

b. Observations and Findings

On September 10, the inspectors found that DR N-96-1743 was originated by engineers to document identification of a concern with the qualification of containment particulate and gaseous RMs to remain operable following a seismic event. The DR originated from reviews of Potential Problem Report (PPR) 96-019 which questioned how the RMs could remain operable following a seismic event given that a loss of the non-seismically qualified instrument air supplies to the system's containment supply and return trip valves could result in isolation of the RMs. The PPR originated from questions raised during training on lessons learned from an event earlier in 1996 where the RMs were identified to be inoperable to meet TS 3.4.6.1 requirements for seismically-qualified leakage detection systems due primarily to non-safety related power supplies (LER 50-338, 339/96004; NRC Inspection Report 50-338, 339/96-07). On September 11, a second DR, DR N-96-1783, was originated by Nuclear Oversight personnel who identified that a similar issue existed for the containment air recirculation fan dampers which were a supporting system to the RMs, but were not classified as safety related or seismically qualified.

After identifying the problem, the inspectors verified that the licensee took actions required by TS 3.4.6.1 for inoperable leakage detection systems on Unit 1. The licensee planned to modify the system to ensure the system met seismic requirements prior to exceeding the 30 day TS allowed outage time. The TS 3.4.6.1 requirement was applicable only in MODES 1 - 4 and did not apply at the time to Unit 2 which was in MODE 5. The licensee was required to submit an LER for the problem, and the inspectors will further review the problem's significance when closing the LER.

The licensee also reviewed the issue prior to Unit 2's entry into MODE 6 and concluded that seismic qualification was not required for the monitors to meet TS 3.3.3.1 and 3.9.9 requirements for operability as a part of the automatic containment isolation system. The inspectors reviewed the basis for this conclusion with licensee engineers. The engineers provided the inspectors with the licensee's proposed TS change and the NRC's Safety Evaluation Report (SER) related to a February 1996 TS change which allowed refueling to be conducted with the containment personnel air lock doors open. In those documents, the design bases for a Fuel Handling Accident (FHA) were clarified. The clarification also stated that the containment isolation system was non-safety related. Additionally, the documents referenced a facility original licensing SER, NUREG-0053, Supplement 7, which, in the context of discussing a FHA in containment, referred to the monitors as non-safety grade. The inspectors concluded that the licensee was correct in stating that the RMs did not have to be seismically qualified to support TS operability requirements in MODE 6.

During the review of the documents related to the February 1996 TS change, the inspectors found a discrepancy with the licensee's implementation of the change. In the change request (Virginia Power letter to the NRC from J. O'Hanlon dated October 17, 1995), the licensee stated that upon verbal notification of a FHA or upon receipt of a high radiation signal, the control room would be manually isolated and the bottled air supply initiated. The inspectors reviewed procedure 0-AP-30, Fuel Failure During Handling, Revision 4, and found that it did not direct operators to initiate the control room bottled air supply during a FHA. The inspectors informed the licensee concerning this finding. After review, the licensee concluded that this was a discrepancy and changed 0-AP-30 to correct the discrepancy prior to commencing core on load for Unit 2.

The inspectors also reviewed the licensee's Safety Evaluation (SE) for the TS change, 95-SE-OT-34, and found that it made several assumptions in concluding that the change did not represent an unreviewed safety question. Included in these assumptions were that actions would be taken to develop an abnormal procedure to require operator action to manually isolate the control room and initiate bottled air. Also, the SE indicated that action would be taken to train operators on the importance of isolating the control room within two minutes. Neither of these actions were completed by the licensee from the time of implementing the change on February 27, 1996, until identified by the inspectors. During that period, the licensee had performed core alterations on at least two occasions: Unit 1 core on load from February 29 to March 3, 1996, and Unit 2 core off load from September 14 to 16, 1996. On both occasions, core alterations were performed with the personnel hatch doors open.

The inspectors then reviewed the significance of not pressurizing the control room bottled air system during a FHA. The inspectors found that the TS change request used this assumption during basis calculations to demonstrate that the dose to control room operators during a FHA would remain less that the limit required by 10 CFR 50 General Design Criteria (GDC) 19. The licensee's amendment request analysis found that with control room pressurization, inleakage was assumed to be 10 cfm and the resultant estimated dose to the thyroid of control room operators was 19 rem. The inspectors concluded that for the assumptions made in the amendment analysis, without control room pressurization, the inleakage would be significantly higher, and the dose to the thyroid of control room operators would likely go from 19 rem to a value exceeding the GDC 19 design criteria of 30 rem (5 rem whole body equivalent).

On September 24, the inspectors discussed the safety significance with licensee fuel and analysis engineers. The engineers conceded that without pressurization, the control room inleakage would likely be significantly higher. As a result, if the basis calculation was performed with all other factors the same, the GDC 19 design criteria for dose to the thyroid would probably be exceeded. However, the engineers pointed out numerous conservative assumptions in the analyses.

These included:

- The analysis assumed containment was not isolated. Procedures included direction to immediately isolate containment manually.

- The analysis took no credit for control room ventilation charcoal filters during the first hour. Procedures included direction to immediately start the fans/filters on recirculation during the first hour.

- The analysis assumed containment release concentrations were present at the control room intake. Weather and plant layout would actually reduce the concentrations likely to leak into the control room.

- The analysis contained numerous additional conservative source term assumptions with regard to fuel composition and amount of radioactivity released.

The inspectors found that the conservative assumptions meant that had a FHA actually occurred, the dose to control room operators would likely not exceed the GDC 19 limits. However, if all assumptions used in the basis calculation actually occurred, the GDC 19 limit would likely have been exceeded.

TS 6.8.1 requires that written procedures be established, implemented and maintained, including by reference to Appendix A of Regulatory Guide 1.33, Revision 2, procedures for irradiated fuel damage while refueling. The licensee's TS change request and the NRC's TS Amendment Nos. 198 and 179, stated that upon notification of a FHA, the control room bottled air supply would be initiated by operators within two minutes. Contrary to these requirements, from February 27, 1996, until September 19, 1996, procedure 0-AP-30 was inadequate in that it did not direct operators to initiate the control room bottled air supply within two minutes of notification of a FHA. This is identified as Violation (VIO) 50-338, 339/96009-04.

c. <u>Conclusions</u>

A violation was identified concerning an inadequate abnormal procedure. The procedure did not contain direction to operators to initiate the control room bottled air system during a FHA as assumed in the bases for a TS amendment allowing refueling with the containment personnel hatches open.

E7 Quality Assurance in Engineering Activities

E7.1 <u>Review of Updated Final Safety Analysis Report (UFSAR) Commitments</u>

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compared plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls (71750)

a. Inspection Scope

On September 14, the inspectors observed technicians performing portions of a high radiation area survey on the refueling deck in containment.

b. Observations and Findings

The inspectors observed that during off load of the first fuel assembly, the assembly was stopped in the transfer tube and radiation surveys were performed to verify high radiation area boundaries in containment. The inspectors observed portions of these surveys on the refueling deck in containment. No discrepancies were noted in survey techniques. However, the inspectors observed several inappropriate work practices by a contract HP technician. In an area near the fuel transfer system operating panel, the technician attempted to remove a radiation area sign from the plastic mesh FME area barrier in order to place it on a high radiation barrier rope. When the technician found that the sign was attached to the mesh with tie-wraps and could not be pulled loose by hand, the technician unzipped both life jacket and PC coverall and reached inside the PCs to obtain a pocketknife. The technician then used the small knife to cut the tie-wraps and remove the sign. When cut, one of the tie-wraps flew approximately twenty feet in the air and fell through a grating near the refueling cavity transfer canal and into the lower containment levels. The other tie-wrap fell near the sign and was retrieved. This event was observed by an inspector and a reactor operator who was part of the refueling crew. The inspectors notified the refueling SRO and HP supervision concerning the observation, and DR N-96-1902 was submitted documenting the event.

The inspectors reviewed the requirements for FME controls. Unit 2 TS 6.8.1 requires that written procedures be established implemented and maintained, including by reference to Appendix A of NRC Regulatory Guide 1.33, Revision 2, procedures for refueling and core alterations. This requirement was implemented in part by VPAP-1302, Foreign Material Exclusion Program, Revision 8, and 2-OP-4.1 which delineated the licensee's FME controls in the reactor cavity area during refueling. Procedures VPAP-1302 and 2-OP 4.1 required that all items which fit through a 2-3/4 inch hole must be logged by the cavity watch prior to an individual's entry into the FME control area. Contrary to these requirements, the HP technician had failed to inform the cavity watch that the small pocketknife was in his possession prior to entry to the FME controls area and prior to using the knife inside the area. This failure constitutes a violation of minor significance and is being treated as an NCV consistent with Section IV of the NRC Enforcement Policy (50-339/96009-05).

c. Conclusions

A non-cited violation was identified for an HP Technician's failure to follow procedures for FME control during refueling.

P1 Conduct of Emergency Preparedness Activities (71750)

On September 6 and 7, the inspectors reviewed the licensee's response to the discovery that 28 of 55 emergency sirens were inoperable. The sirens were found to be inoperable when they failed to respond during a test during Tropical Storm Fran (Section O2.3). The sirens were degraded due primarily to losses of local power to the sirens caused by damage from the storm. The licensee contacted the Virginia State Department of Emergency Services and verified that alternate notification means (route alerting) remained available in accordance with the site Emergency Plan. The licensee informed the inspectors when a majority of the sirens were returned to service the following day. The inspectors concluded that the licensee properly resolved the problem.

S1 Conduct of Security and Safeguards Activities (71750)

S1.1 Unescorted Visitor (71750)

On September 19 at approximately 12:45 p.m., the inspectors entered the 2H EDG room and observed that the vendor representative, visitor badge V-0018, was not with his escort. He was on the control side of the EDG and was not observable by the three badged employees on the other side of the EDG. The unescorted visitor accompanied the inspectors to the other side of the EDG where a member of the plant staff assumed escort responsibilities for the individual. The person who had been watching the visitor re-entered the 2H EDG room a few moments later. The person had left the area to obtain work materials.

The inspectors reported the event to security supervision. Security personnel subsequently counseled the involved personnel on escort/visitor responsibilities. When the licensee issued a DR concerning this observation, another example was included which had been identified by the licensee on September 15. This second example also involved personnel in the 2H EDG room. Pending additional review of previous unescorted visitor occurrences, this item is identified as an URI (50-339/96009-06).

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 27 and October 18, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Funderburk, Superintendent, Outage and Planning
E. Grecheck, Assistant Station Manager, Operations and Maintenance
J. Hayes, Superintendent, Operations
D. Heacock, Assistant Station Manager, Nuclear Safety and Licensing
P. Kemp, Supervisor, Licensing
T. Maddy, Superintendent, Security
W. Matthews, Station Manager
M. McCarthy, Director, Nuclear Oversight
D. Roberts, Supervisor, Station Nuclear Safety
H. Royal, Superintendent, Nuclear Training
R. Saunders, Vice President, Nuclear Operations
D. Schappell, Superintendent, Site Services
R. Shears, Superintendent, Maintenance
J. Smith, Superintendent, Station Engineering
A. Stafford, Superintendent, Radiological Protection

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 40500: Effectiveness of Licensing Controls in Identifying, Resolving, and Preventing Problems
IP 61726: Surveillance Observations
IP 62703: Maintenance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 92700: Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901: Followup - Plant Operations
IP 93702: Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-339/96009-01	NCV	Failure to Follow Procedure for Racking In Charging Pump Breaker (Section O4.1).
50-339/96009-02	NCV	Failure to Meet 10 CFR 50.54k Requirements for Operator Presence at Unit Controls (Section O8.1) (EA 96-292).
50-339/96009-03	URI	Review Anomalies in Large Bore Snubber Test Data (Section M1.3).

50-338, 339/96009-04	VIO	Inadequate Procedure for Fuel Handling Accident (Section E2.1).
50-339/96009-05	NCV	Failure to Follow Procedures For FME Control By HP Technician Near Reactor Cavity (Section R1).
50-339/96009-06	URI	Review occurrences of unescorted visitors (Section S1.1).

Closed

50-339/96007-01	URI	Review Compliance With 10 CFR 50.54k Requirements For Operator Presence at Unit Controls (Section 08.1) (EA 96-292).
50-339/96009-01	NCV	Failure to Follow Procedure for Racking In Charging Pump Breaker (Section 04.1).
50-339/96009-02	NCV	Failure to Meet 10 CFR 50.54k Requirements for Operator Presence at Unit Controls (Section 08.1) (EA 96-292).
50-339/96009-05	NCV	Failure to Follow Procedures For FME Control By HP Technician Near Reactor Cavity (Section R1).
50-338/96005	LER	Reactor Trip on High Negative Flux Rate (Section 08.2).