UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

July 26, 2001

Virginia Electric and Power Company
ATTN: Mr. David A. Christian
Sr. Vice President and
Chief Nuclear Officer
Innsbrook Technical Center - 2SW
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION - NRC INTEGRATED INSPECTION REPORT NOS. 50-280/01-02, 50-281/01-02, 72-002/01-02

Dear Mr. Christian:

On June 30, 2001, the NRC completed an inspection at your Surry Power Station, Units 1 and 2, and the Surry Independent Spent Fuel Storage Installation. The enclosed report documents the inspection findings which were discussed on July 10, 2001, with Mr. R. Blount and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selective procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it had been entered into your corrective action program, the NRC is treating this issue as a non-cited violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Surry Power Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system

(ADAMS). ADAMS is accessible from the NRC Web site at
*http://www.nrc.gov/NRC/ADAMS/index.html* (the Public Electronic Reading Room).

Sincerely,

***/RA/***

Kerry D. Landis, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket Nos.: 50-280, 50-281, 72-002
License Nos.: DPR-32, DPR-37, SNM-2501

Enclosure: Integrated Inspection Report

Attachment: Supplemental Information

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REGION II

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|---|---|
| Docket Nos.: | 50-280, 50-281, 72-002 |
| License Nos.: | DPR-32, DPR-37, SNM-2501 |
| Report Nos.: | 50-280/01-02, 50-281/01-02, 72-002/01-02 |
| Licensee: | Virginia Electric and Power Company (VEPCO) |
| Facilities: | Surry Power Station, Units 1 & 2<br>Surry Independent Spent Fuel Storage Installation |
| Location: | 5850 Hog Island Road<br>Surry, VA 23883 |
| Dates: | April 1 - June 30, 2001 |
| Inspectors: | R. Musser, Senior Resident Inspector<br>K. Poertner, Resident Inspector<br>G. McCoy, Resident Inspector<br>R. Baldwin, Senior Operations Engineer (Section 1R11)<br>L. Miller, Senior Operations Engineer (Section 1R11)<br>S. Rose, Operations Engineer (Section 1R11)<br>J. Kreh, Emergency Preparedness Inspector (Section 1EP1)<br>G. Hopper, Senior Operations Engineer (Section 1EP1)<br>L. Moore, Reactor Inspector (Section 4OA3.1)<br>P. Fillion, Reactor Inspector (Section 4OA3.1)<br>L. Garner, Senior Project Engineer (Section 4OA3.2)<br>R. Schin, Senior Reactor Inspector (Section 4OA3.3) |
| Approved by: | K. Landis, Chief, Reactor Projects Branch 5<br>Division of Reactor Projects |

Enclosure

# SUMMARY OF FINDINGS

IR 05000280-01-02, IR 05000281-01-02, IR 07200002-01-02, on 04/01 - 06/30/2001; Virginia Electric and Power Co.; Surry Power Station, Units 1 & 2 and Independent Spent Fuel Storage Installation. Event Follow-up, Integrated Resident Inspector Report.

The inspection was conducted by resident inspectors, three senior operations engineers, an emergency preparedness inspector and a radiation specialist. In-office reviews were performed by a senior project engineer, a senior reactor inspector and two reactor inspectors. This inspection identified one green finding, which was a non-cited violation. The significance of issues is indicated by their color (green, white, yellow, red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at *http://www.nrc.gov/NRR/OVERSIGHT/index.html*.

Inspector Identified Finding

**Cornerstones: Mitigating Systems and Initiating Events**

- Green. A non-cited violation of Surry License Condition 3.I on fire protection was identified for failure to implement a fire brigade training program as required by the license condition. The licensee's fire brigade program inappropriately allowed the use of walk through drills, false alarms, and actual fires to satisfy the quarterly requirements for fire brigade drills. Consequently, the fire brigade received less than the minimum required training drills in 1998. Therefore, it could be less effective at fighting fires.

  This violation was of more than minor significance because a potentially less effective fire brigade has a credible impact on safety, in that, untimely or ineffective action by the fire brigade could credibly allow a fire to affect the operability or function of a system or train required for safe shutdown. This issue was determined to have very low safety significance because there was no identified degradation of the other parts of the fire protection defense in depth: fire barriers, fire alarms, and automatic fire suppression (Section 4OA3.3).

## Report Details

Unit 1 operated at power until May 5, 2001, when the unit was taken offline for main generator "B" phase bushing repairs. The unit was returned to service on May 6, 2001.

Unit 2 operated at power until May 12, 2001, when the unit was taken offline for pressurizer safety valve maintenance. The unit was returned to service on May 22, 2001.

Unit 1 and Unit 2 operated at power for the remainder of the reporting period.

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity**

1R04 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

For the systems identified below, the inspectors reviewed plant documents to determine correct system lineup, and observed equipment to verify that the system was correctly aligned:

- Number 2 Emergency Diesel Generator (EDG) (2-OP-EG-001A, “EDG 2 System Alignment,” Revision 4), while the Number 3 EDG was out of service for corrective maintenance,
- Unit 2 Auxiliary Feedwater System (2-OP-FW-001A, “Auxiliary Feedwater System Valve Alignment,” Revision 2-P1), and
- Unit 1 and 2 Control Room and Emergency Switchgear Ventilation (0-OP-VS-006A, “Control Room and Relay Room Ventilation System Alignment,” Revision 5-P1).

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a detailed walkdown on the accessible portions of the Unit 1 auxiliary feedwater system. The walkdown emphasized material condition and correct alignment of system components such as valves, breakers and hand switches. The inspectors used the following operating procedures (OP) and drawings:

- 1-OP-28A, “Auxiliary Feedwater System Alignment,” Revision 1,

- 1-OP-46.2C, “Instrument Air System Alignment,” Revision 22, and

- Drawings 11448-FM-064A, Sheet 1, Revision 45; 11448-FM-064A, Sheet 4, Revision 45; 11448-FM-068A, Sheet 1, Revision 55; 11448-FM-068A, Sheet 3, Revision 43; 11448-FM-075C, Sheet 3, Revision 30; and 11548-FM-068A, Sheet 3, Revision 49.

A review of outstanding work orders was performed to determine if any deficiencies existed which could affect the ability of the system to perform its function.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours to assess the adequacy of the fire protection program implementation. The inspectors checked the control of transient combustibles and the condition of the fire detection and fire suppression systems (using “SPS Appendix R Report,” Revision 17) in the following areas:

- Unit 2 Cable Spreading room,

- Main Control Room,

- Unit 2 Normal Switchgear Room,

- Number 1 EDG Room,

- Unit 1 Cable Vault, and

- Mechanical Equipment Room Number 3.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Biennial Requalification Program Review

a. Inspection Scope

The inspectors reviewed facility operating history since the last requalification program inspection for indications of operator weaknesses. The inspectors also reviewed two annual written examinations and evaluated their effectiveness in providing a basis for assessing operator knowledge of material covered in the requalification training program. Examination quality, licensee effectiveness in integrating industry, plant and student feedback into the requalification training program, and examination development methodology were evaluated for compliance with criteria contained in the licensee's "Functional Implementation Guidelines." The inspectors observed annual dynamic simulator examinations (five scenarios) for three operator teams to access the adequacy of the licensee's evaluation of operator knowledge and abilities. During these observations, the inspectors assessed licensee evaluator effectiveness in pinpointing operator performance deficiencies requiring supplemental training. The inspectors also evaluated and observed portions of the walkthrough examination administered during this requalification segment, to access evaluator performance.

The inspectors reviewed and evaluated the licensee's remedial training program for operator deficiencies identified during the previous year. The inspectors also reviewed a sample of on-shift licensed operator qualification records, watchstanding records and medical records to ensure compliance with 10 CFR 55.59, Requalification and 10CFR 55.53, Conditions of License.

b. Findings

No findings of significance were identified.

.2 Quarterly Requalification Activities Review

a. Inspection Scope

The inspectors observed licensed operator performance during simulator training session RQ-01.3-SE-1, "Loss of 4.16KV bus with main steam line break outside containment," to determine whether the operators:

- were familiar with and could successfully implement the procedures associated with recognizing and recovering from the scenario,
- recognized the high-risk actions in those procedures, and
- were familiar with related industry operating experiences.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described in the plant issues listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) using VPAP 0815, "Maintenance Rule Program," Revision 11, and the Surry Maintenance Rule Scoping and Performance Criteria Matrix, Revision 12, with respect to the characterization of failures, the appropriateness of the associated a(1) or a(2) classification, and the appropriateness of either the associated a(2) performance criteria or the associated a(1) goals and corrective actions:

- S-2001-1672, 2-IPT-FT-FW-2485 comparator exhibits erratic behavior,
- S-2001-1537, Steam generator level comparator tripping erratically,
- S-2001-1402, 2-SS-TV-202A indicates intermediate position,
- S-2001-0973, Unit 1 instrument air dryer failure,
- S-2001-0966, 1-CC-RI-105 failed, and
- S-2001-0784, 2-CV-TV-250D failed stroke time test.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated the adequacy, accuracy, and completeness of plant risk assessments performed prior to any changes in plant configuration for maintenance activities or in response to emergent conditions. When applicable, the inspectors determined if the licensee entered the appropriate risk category in accordance with plant procedures. Specifically, the inspectors reviewed the:

- Simultaneous removal from service of 1-CC-P-1A (“A” component cooling water pump), 1-SA-C-2 (number 2 service air compressor) and 2-IA-C-4A (Unit 2 “A” containment instrument air compressor) while performing maintenance on 2-RD-CAB-2BD (BD full length control rod cabinet) and switchyard activity (maintenance on number 1 transformer),
- Simultaneous removal from service of 3-EE-EG-1 (Number 3 EDG), 1-SW-P-1B (“B” emergency service water pump), and 1-CC-P-1D (“D” component cooling water pump),
- Simultaneous removal from service of 3-EE-EG-1 (Number 3 EDG), 2-FW-P-2 (Unit 2 turbine driven auxiliary feedwater pump (TDAFWP)) and 1-VS-E-4E (“E” control room chiller),
- Simultaneous removal from service of 1-FW-P-2 (Unit 1 TDAFWP), 0-OSP-AAC-001 (alternate AC diesel generator), 1-VS-E-4A (A control room chiller), and 1-SW-P-1B (“B” emergency service water pump),
- Simultaneous removal from service of 1-SA-C-1 (Unit 1 service air compressor), 1-SW-P-1C (“C” emergency service water pump), 1-CC-E-1D (“D” component

cooling water heat exchanger) and 1-SA-C-2C (diesel powered air compressor), and

- Simultaneous removal from service of 2-VS-AC-6 (Unit 2 emergency switchgear room air handling unit), 2-BC-E-1B (“B” bearing cooling heat exchanger), 1-SW-P-1A (“A” emergency service water pump), 1-FP-P-2 (diesel driven fire pump), and 1-VS-E-3 (central air conditioning water chiller) during the performance of 2-IPT-FT-RP-SI-001 (reactor safety actuation logic functional test).

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

Inspectors reviewed the performance of on-watch personnel during a plant transient caused by a momentary voltage fluctuation on the 2-III vital bus. This voltage transient was caused by a loose wire contacting a bus bar during work in the “2A2" uninterruptible power supply regulating line conditioner. The voltage spike caused the Unit 2 controllers for charging flow, volume control tank level, and “B” feed regulating valve to shift to manual control. Abnormalities or spikes were noted on some of the recorders on the main control board. Operators verified proper operation of the controllers and returned them to automatic control.

Inspectors reviewed the operator logs and interviewed personnel to determine what occurred and how the operators responded. Inspectors reviewed procedure 2-AP-10.03, “Loss of Vital Bus III,” Revision 4 to evaluate the operator response.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors evaluated the technical adequacy of the following operability evaluations to ensure that operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The operability evaluations were described in the engineering transmittals (ETs) and plant issues listed below:

- S-2001-1227, Piping repair on the outlet of 1-CC-E-1C (“C” component cooling water heat exchanger),

- S-2001-1312, Leakage of 1-MS-NRV-101C (Unit 1 “C” non-return valve),

- ET S-01-0088, Inservice test reference values for 1-VS-P-1C following replacement of rotating assembly,
- ET S-01-0085, Inservice test reference values for 1-SW-P-1A following weight change, and
- S-2001-1760, Elevated vibration on 1-CH-P-1A pump outboard bearing.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance test procedures and activities associated with the repair or replacement of the following components to determine that the procedures and test activities were adequate to verify operability and functional capability following maintenance of the following equipment:

- Work Order (WO) 446196 - Return to service after overhaul of 1-CC-P-1A (“A” component cooling water pump),
- Replacement of Control Board Handswitch for 2-FW-P-3A (“A” Motor Driven Auxiliary Feedwater Pump) in accordance with 0-ECM-1807-03, “Protective Relay Control Circuit Operability Check,” Revision 0,
- Replacement of Auto/Exercise Switch for 3-EE-EG-1 (Number 3 EDG),
- Post maintenance test in accordance with 0-OPT-SW-001 after overhaul of “A” emergency service water pump motor,
- WO 445351 - Oil change and preventive maintenance on Unit 2 TDAFW pump, and
- WO 448320 - Replace component cooling water pump 1A coupling.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the surveillance tests listed below, the inspectors examined the test procedure and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately demonstrated that the affected equipment was functional and operable:

- 2-IPT-FT-RP-SI-001A, “Train A Safeguards Actuation Logic Functional Test,” Revision 8,
- 1-PT-8.1, “Reactor Protection System Logic (For Normal Operations),” Revision 20,
- 1-OPT-RX-005, “Control Rod Assembly Partial Movement,” Revision 8,
- 0-OPT-EG-001, “Number 3 Emergency Diesel Generator Monthly Start Exercise Test,” Revision 17,
- 0-OPT-VS-002, “Auxiliary Ventilation Filter Train Test,” Revision 16, and
- 2-OPT-CH-001, “Charging Pump Operability and Performance Test for 2-CH-P-1A,” Revision 25.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification S1-01-005, “Bypassing the Unit 1 Proportional Pressurizer Heater Controller (Robicon),” to determine whether system operability / availability was affected, that configuration control was maintained, and that the associated safety evaluation (01-030) adequately justified implementation.

b. Findings

No findings of significance were identified.

**Cornerstone: Emergency Preparedness**

1EP1 Exercise Evaluation

a. Inspection Scope

The inspectors reviewed the objectives and scenario to determine whether they were designed to test major elements of the licensee’s emergency plan. The inspectors observed and evaluated the licensee’s performance in the exercise, conducted on June 26, 2001, from 10:00 a.m. to 2:33 p.m., as well as, selected proceedings related to the licensee’s conduct of the exercise. Licensee activities inspected during the exercise included those occurring in the Control Room Simulator, Technical Support Center, Operational Support Center, and Local Emergency Operations Facility. The NRC’s evaluation focused on the risk-significant activities of event classification, notification of governmental authorities, onsite protective actions, offsite protective action recommendations, and accident mitigation. The inspectors also evaluated command and control, the transfer of emergency responsibilities between facilities, communications, and adherence to emergency plan implementing procedures. Licensee performance was evaluated against applicable licensee procedures and regulatory requirements. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process, as well as, the presentation of critique results to plant management.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

On April 18, 2001, the inspectors observed a simulator based training evolution to verify that proper emergency plan classifications, notifications, and protective action recommendations were made.

The inspectors observed an emergency response training drill conducted on May 9, 2001, to assess the licensee's performance in emergency classification, notification, and protective action recommendation development.

b. Findings

No findings of significance were identified.

## 4 OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Unplanned Scrams per 7000 Critical Hours PI

a. Inspection Scope

The inspectors performed a periodic review of the Unplanned Scrams per 7000 Critical Hours PI for Units 1 and 2. Specifically the inspectors reviewed this PI from the third quarter 2000 through the first quarter 2001. Documents reviewed included applicable monthly operating reports and licensee event reports.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant System Leakage PI

a. Inspection Scope

The inspectors performed a periodic review of the reactor coolant system leakage PI for Units 1 and 2. Specifically, the inspectors reviewed this PI from the second quarter of 2000 through the first quarter of 2001. Documents reviewed included operator logs and leakage surveillance records.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Activity PI

a. Inspection Scope

The inspectors performed a periodic review of the reactor coolant system activity PI for Units 1 and 2. Specifically, the inspectors reviewed this PI from the second quarter of 2000 through the first quarter of 2001. Documents reviewed included applicable monthly operating reports and chemistry department logs.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Unresolved Item (URI) 50-280, 50-281/00007-01: Ability of the low head safety injection (LHSI) pump to meet the design basis flow requirement with a sustained degraded voltage above trip setpoint coincident with a single failure of the redundant train. During the Safety System Design Inspection, the inspectors reviewed an evaluation by the licensee which dealt with the topic of offsite power and the setpoint for the degraded grid protection relay. The conclusions section of this evaluation contained the following statement: "Although the functional performance of safety-related equipment operated at a reduced voltage (i.e., slightly above the degraded voltage protection system setpoint) may be diminished, sustained operation at this condition is beyond the design basis of the electrical distribution system." The inspectors were concerned with this statement and pursued this issue to determine the following facts. The present setpoint of the degraded grid protection relay corresponds to a voltage of 85 percent of motor rated voltage at the terminals of many 480 volt motors. A licensee review of the flow requirements for the safety-related pumps powered by 480 volt motors and of the overcurrent protection shows that, with one exception, sufficient margin exists such that the design basis is met even at 85 percent voltage. That exception was the LHSI pump 1B which the licensee stated needed further analysis. The licensee took the position that it really was not a problem in any case, because it was beyond the design basis to postulate sustained degraded voltage together with single failures, and LHSI pump 1A was not affected. The inspectors informed the licensee that this condition was in fact within the design basis. At the end of the onsite inspection period, the licensee presented another engineering study, Engineering Transmittal CEE-98-013, Revision 0, “Voltage Profile Analysis Surry Power Station, Units 1&2,” which had been completed in June 1998. The study calculated actual motor speeds at reduced voltages. For the LHSI pump 1B, this study calculated that the speed at 85 percent motor rated voltage would be reduced from the full load rated voltage speed of 1763 rpm to 1749 rpm. After the onsite portion of the inspection, the NRC reviewed CEE-98-013 and found that the calculation used assumptions and techniques given in Institute of Electrical and Electronics Engineers (IEEE) Standard 666-1991 (reaffirmed 1996), “IEEE Design Guide for Electric Power Service Systems for Generating Stations,” to calculate the motor / pump speed at reduced voltages. Based on their review, the inspectors agreed with the licensee as to what the actual pump speed would be at 85 percent voltage.

The licensee then made additional calculations for the relevant systems. Calculation ME-0616, “LHSI Minimum Flow, LHSI Pump Operability,” Revision 0, determined the worst case minimum flow values for anticipated reactor coolant system pressures and containment pressures using the new speed for the LHSI pump of 1749 rpm. These flow values were used as design input in the Large Break Loss of Coolant Accident (LBLOCA) model, Dominion Calculation SM-1294, “Surry Power Station LBLOCA Re-analysis,” Revision 0. The result of the analysis was that no core parameters exceeded the limits specified in 10 CFR 50.46. The inspectors concluded that the licensee had adequately determined that the LHSI pumps would meet their design flow requirements under conditions of sustained degraded voltage above the voltage relay setpoint. No violations of NRC requirements were identified.

.2 (Closed) URI 50-280, 281/00009-01: Determine the risk significance of the inside recirculation spray pumps not having a check valve installed in the discharge piping. During a loss of coolant accident with offsite power available and after the inside recirculation spray pumps have started spraying containment, a subsequent loss of offsite electrical power could result in the pumps rotating backwards while the spray headers drain through the pumps. An automatic start of the EDGs then would re-power these pumps and potentially cause a common mode failure of both inside recirculation trains. Probabilistic risk analysis information reveals that the probability of occurrence of this sequence of events is very small. Consequently, the overall risk significance of this design is also very small.

.3 (Closed) URI 50-280, 281/99004-02: Review and Evaluate the Licensee’s Practice of Walk Through Fire Brigade Drills Being Used to Fulfill Annual Fire Drill Requirements.

a. Inspection Scope

This URI had been opened for NRC review of the Surry licensing basis for fire protection to determine if the licensee’s use of walk through fire brigade drills to fulfill annual drill requirements had been accepted by the NRC. The licensee had stated that walk through fire brigade drills were fire brigade group discussions of a possible fire situation in a plant area and were run like a desktop exercise. The NRC inspector had noted that the walk through fire brigade drills were not consistent with NRC guidelines in that they were not performed in response to fire alarms, did not exercise the fire brigade simulated use of fire equipment, and did not exercise the fire brigade in full turnout gear or the self contained breathing apparatus (SCBA). The licensee had credited five walk through drills in the 1998 total of 22 drills for five operational shifts. Licensee personnel had stated that the NRC had approved the use of walk through fire brigade drills in a Safety Evaluation Report (SER) dated September 19, 1979.

To resolve this issue, the inspector obtained the following documents and reviewed them in the office: Surry facility operating License Condition 3.I on fire protection; the NRC SER on the Surry Fire Protection Program dated September 19, 1979; Supplements to the SER dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and July 23, 1992; the NRC SER dated December 16, 1998, for TS Amendment No. 217 regarding the Surry Fire Protection Program; NRC letter to Surry dated June 14, 1978; Surry letters to the NRC dated March 6, 1978, and July 21, 1978; 10 CFR 50.48, "Fire Protection;" 10 CFR 50, Appendix A, Criterion 3, "Fire Protection;" 10 CFR 50, "Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979;" NRC Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants;" NRC supplemental guidance "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance;" National Fire Protection Association (NFPA) 27 - 1975, "Private Fire Brigades;" and Virginia Power Station Administrative Procedure (VPAP)-2401, "Fire Protection Program," Revision 11.

b. <u>Findings</u>

Based on an in-office review of the above documents, the inspector found that License Condition 3.I requires that the licensee implement and maintain the fire protection program as approved in the NRC SER dated September 19, 1979, and supplements to that SER as listed above. The NRC SER dated September 19, 1979, approved the licensee's fire protection program (including fire brigade drills) for compliance with requirements of 10 CFR 50, Appendix A, Criterion 3, based on the licensee's stated conformance with the NRC guidance in NRC Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" and NRC supplemental guidance "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance." The supplements to the SER did not alter the conditions of that approval with respect to fire brigade drills or training.

The 1979 SER discussed one deviation from the NRC guidance on fire brigade training that had been requested by the licensee. In a letter dated March 6, 1978, the licensee had stated that the NRC guidelines for fire brigade training, as stated in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," were being met with one exception. That exception was for Section 2.0, Practice, and was stated as follows: "Practice sessions are being conducted which simulates possible fire conditions that could occur within the plant. However, to provide fires of similar magnitude, complexity and difficulty as those which could occur within the plant would require duplication of the plant facilities." In a letter dated July 21, 1978, the licensee restated this exception. The inspector noted that "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," Section 2.0, Practice, stated the following: "Practice sessions should be held for fire brigade members on the proper method of fighting various types of fires of similar magnitude, complexity, and difficulty as those which could occur in a nuclear power plant. These sessions should provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions. These practice sessions should be provided at regular intervals but not to exceed one year for each fire brigade member."

The NRC SER stated that during discussions about the requested deviations from the NRC guidance, the licensee had agreed to the following: "to perform practice sessions which include simulations in plant areas (walk throughs, dry runs) of the proper fire fighting methods for various fires of similar magnitudes and complexity that could occur in a nuclear power plant. The duplication of actual room configurations in various plant areas will not be required." The SER further stated: "The licensee's original objections to these practice sessions were based on their assumption that to conduct these practice sessions would require duplication of actual room configurations in various plant areas. The licensee has agreed to perform these practice sessions, after being informed that the duplication of actual room configurations of various plant areas is not required by the NRC guidance document, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance." The inspector noted that licensee procedure VPAP-2401 required that each member of the fire brigade attend annual retraining sessions that include actual fire extinguishment and donning protective equipment including emergency breathing apparatus. The inspector concluded that procedure VPAP-2401 was reasonably consistent with the NRC SER regarding annual practice sessions for individual fire brigade members.

The inspector noted that procedure VPAP-2401 also included requirements for fire brigade drills. VPAP-2401 stated that fire brigade drills were to be conducted quarterly for each shift. It also stated that actual fires, false alarms, and walk through drills when properly critiqued can be used to fulfill annual drill requirements. The NRC guidance on fire brigade drills was stated in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," Section 3.0, Drills. The NRC guidance also recommended following NFPA 27 - 1975, "Private Fire Brigades," for fire brigade training. The inspector found that the licensee had requested and the NRC SER had approved no exceptions to this guidance. Section 3.0, Drills, included the following guidance:

- fire brigade drills should be performed in the plant so that the fire brigade can practice as a team and should be performed at regular intervals not to exceed three months,

- drills should assess fire alarm effectiveness, time required to notify and assemble fire brigades, and selection, placement, and use of equipment,

- drills should include the simulated use of fire fighting equipment in all safety related areas containing significant fire hazards, and

- drills should be pre-planned to establish the training objectives of the drill and should be critiqued to determine how well the training objectives have been met.

NFPA-27 - 1975 included the following additional guidance:

- In drills, equipment should be operated whenever possible. For example, portable extinguishers should actually be discharged, respiratory protective equipment should be operated and water should be turned into hose lines.

The inspectors concluded that procedure VPAP-2401 was not consistent with the NRC SER with regard to fire brigade drills. Specifically, the provision for crediting walk through drills, false alarms, and actual fires to fulfill quarterly drill requirements did not follow the NRC guidance as required by the SER. Walk through drills, conducted as group discussions like table top exercises, do not satisfy the criteria of: 1) being performed in safety related areas of the plant containing significant fire hazards, 2) assessing fire alarm effectiveness and time required to notify and assemble fire brigades, 3) assessing selection, placement, and use of equipment, 4) simulated use of fire fighting equipment, 5) actual operation of equipment whenever possible, including discharging portable fire extinguishers, operating respiratory protective equipment, and turning water into hose lines. In addition, false alarms and actual fires do not satisfy the criteria of being pre-planned. The procedural provisions for crediting walk through drills, false alarms, and actual fires to fulfill annual drill requirements, and the resulting crediting of five walk through (desktop group discussion) drills of the 22 total drills for five shifts in 1998, constitute nonconformances with the NRC SER dated September 19, 1979. These nonconformances represent a violation of NRC requirements as stated in Surry facility operating License Condition 3.I. The nonconformances also represent a performance deficiency wherein the fire brigade was receiving less than the minimum required training drills and therefore could be less effective at fighting fires.

The condition of the fire brigade receiving less than the minimum required training drills had more that minor safety significance because it had a credible impact on safety, in that it could result in the brigade being less effective in fighting fires. For example, fires could burn longer, increasing the risk of fire barrier failure and challenging safe shutdown equipment. Therefore, this lack of training drills could adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. This issue was determined to have very low safety significance (Green) because the fire brigade was only potentially less effective and there were no identified degraded conditions in the other parts of the fire protection defense in depth: fire barriers, fire alarms, and automatic fire suppression systems. Also, the potentially less effective fire brigade did not result in any inoperable equipment relied upon to mitigate accidents and did not initiate any events.

Surry operating License Condition 3.I requires that the licensee implement and maintain the fire protection program as approved in the NRC SER dated September 19, 1979. The failure to implement the fire protection program as approved in the SER, with respect to fire brigade drills, is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 50-280, 281/01002-01. The licensee entered this issue into their corrective action program as Surry Plant Issue S-2001-1829.

4OA5 Other

Transnuclear (TN)-32 Spent Fuel Cask Loose Lid Bolts

As discussed in Section 4OA5 of NRC Integrated Inspection Report No. 50-280, 281/72-002/01-01, the licensee was in the process of completing a root cause evaluation for loose cask lid bolts discovered on February 2, 2001, on cask 2-9.

During this inspection period, the licensee completed the root cause evaluation and implemented corrective actions to preclude further incidents of bolt relaxation. Specific corrective actions implemented were: (1) increasing the final nominal bolt torque to 1140 ft-lbs (previous value was 930 ft-lbs), (2) changing the bolt lubricant from neo-lube to High Purity N-5000, and (3) instituting a thermal soak time of 12 hours following cask drying and helium backfill to allow the cask to reach thermal equilibrium. In addition, the lid bolts on all in service TN-32 casks will be tightened to the new nominal torque value.

The licensee reloaded cask 2-9, implementing the above specified corrective actions, and returned the cask to the Independent Spent Fuel Storage Installation.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results with Mr. R. Blount, the Site Vice President, and other members of licensee management on July 10, 2001.

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

.2 Other Meetings

The NRC Senior Resident Inspector and the Division of Reactor Projects Branch Chief assigned to the Surry Power Station met on June 28 with Virginia Electric and Power Company (VEPCO) to discuss the NRC's Reactor Oversight Process (ROP) annual assessment of safety performance for Surry Power Station, Units 1 and 2, for the period of April 2, 2000 - March 31, 2001. The major topics addressed were: the NRC's assessment program, the results of the Surry Power Station assessment, and the NRC's Agency Action Matrix. Attendees included VEPCO management, plant staff members, local officials, and news media personnel.

Following the annual assessment meeting, a meeting was held with local officials to discuss the ROP and NRC activities involving Surry Power Station.

Both meetings were open to the public. Information used for the discussions of the ROP is available from the NRC's document system (ADAMS) as accession number ML011980088. ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/NRC/ADAMS/index.html* (the Public Electronic Reading Room).

## ATTACHMENT

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

M. Adams, Manager, Engineering
R. Allen, Manager, Maintenance
R. Blount, Site Vice President
M. Crist, Manager, Nuclear Oversight
B. Foster, Director, Nuclear Station Safety and Licensing
D. Llewellyn, Manager, Training
M. Small, Supervisor, Licensing
T. Sowers, Director, Nuclear Station Operations and Maintenance
T. Steed, Manager, Radiological Protection
J. Swientoniewski, Manager, Operations

## ITEMS OPENED AND CLOSED

Opened and Closed

| | | |
|---|---|---|
| 50-280, 281/01002-01 | NCV | Failure to implement a fire brigade drill program as required by Surry License Condition 3.I (Section 4OA3.3) |

Closed

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| 50-280, 281/00007-01 | URI | Ability of the LHSI pump to meet the design basis flow requirement with a sustained degraded voltage above trip setpoint coincident with a single failure of the redundant train (Section 4OA3.1) |
| 50-280, 281/00009-01 | URI | Determine the risk significance of the inside recirculation spray pumps not having a check valve installed in the discharge piping (Section 4OA3.2) |
| 50-280, 281/99004-02 | URI | Review and evaluate the licensee’s practice of walk through fire brigade drills being used to fulfill annual fire drill requirements (Section 4OA3.3) |