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

January 31, 2008

EA-08-038

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. 05000280/2007005 AND 05000281/2007005 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Christian:

On December 31, 2007, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Surry Power Station, Units 1 and 2 and the Surry Independent Fuel Storage Installation. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 10, 2008, with Mr. Jernigan and other members of your staff.

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

This report documents two NRC identified and two self-revealing findings of very low safety significance (Green), which involved violations of NRC requirements. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they have been entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs), consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk Washington DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington D.C. 20555-0001; and the NRC Resident Inspector at the Surry Power Station.

In addition, the inspectors reviewed the events associated with heavy load lifts and industry uncertainty regarding the licensing bases for handling of reactor vessel heads, which led to the issuance of EGM 07-006, "Enforcement Discretion for Heavy Load Handling Activities," on September 28, 2007. The Nuclear Energy Institute (NEI) has informed NRC of industry approval of a formal initiative that specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices.

The inspection identified a violation of requirements to update the updated final safety analysis report pursuant to 10 CFR 50.71(e) to reflect aspects of heavy load lifts involving the reactor vessel head and include information from a reactor vessel head drop analysis. The NRC staff believes implementation of the initiative will resolve uncertainty in the licensing bases for heavy load handling, and enforcement discretion related to the uncertain aspects of the licensing basis is appropriate during the implementation of the initiative. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eugene F.Guthrie, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosure: Integrated Inspection Report 05000280, and 281/2007005
w/Attachment: Supplemental Information

(cc w/encl cont'd - See page 3)

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/RA/

Eugene F.Guthrie, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosure: Integrated Inspection Report 05000280/2007005 and 05000281/2007005
w/Attachment: Supplemental Information

(cc w/encl cont'd - See page 3)

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E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-280, 50-281, 72-055
License Nos.: DPR-32, DPR-37

Report Nos.: 05000280/2007005, 05000281/2007005

Licensee: Virginia Electric and Power Company (VEPCO)

Facilities: Surry Power Station, Units 1 & 2
Surry Independent Fuel Storage Installation

Location: 5850 Hog Island Road
Surry, VA 23883

Dates: October 1 - December 31, 2007

Inspectors: C. Welch, Senior Resident Inspector
D. Merzke Acting Resident Inspector
D. Arnett, Project Engineer
R. Carrion, Senior Reactor Inspector (1R08)
L. Garner, Senior Project Engineer (4OA2)
C. Fletcher, Reactor Inspector (1R08)
R. Hamilton, Senior Health Physicist (2OS2, 2PS1, 2PS3, 4OA1, 4OA5)
L. Lake, Senior Reactor Inspector (1R08)
W. Loo, Senior Health Physicist (2OS1, 4OA5)
R. Moore, Senior Reactor Inspector (4OA5)
W. Rogers, Senior Reactor Analysts
A. Nielsen, Health Physicist (2PS2)

Approved by: Eugene F.Guthrie, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000280/2007-05, IR 05000281/2007-05; 10/01/2007- 12/31/2007; Surry Power Station Units 1 & 2, and Independent Spent Fuel Storage Installation; Fire Protection, Inservice Inspection Activities, Post-Maintenance Testing, and Other Activities.

The report covered a three month period of inspection by resident and region-based inspectors. Four Green findings, all of which were non-cited violations (NCVs), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process", Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A Green NRC-identified non-cited violation of paragraph (a)(1) of 10 CFR 50.48, "Fire Protection," was identified for failure to maintain the fire suppression capability for the Unit 1 containment building as specified by the approved fire protection plan. On October 27, 2007, the licensee failed to provide equivalent fire suppression capacity when the Unit 1 containment fire hose stations were removed from service for repair. This finding was entered into the licensee's corrective action program as condition report CR025073. Planned corrective actions included developing equivalent fire suppression capacity determinations for other hose stations.

  This finding is more than minor because it was associated with a degradation of a fire protection feature. The finding is of very low safety significance because it involved low degradation of a fixed fire protection system. A significant cause of this finding involved the Decision-Making component of the cross-cutting area of Human Performance and the aspect of making safety-significant or risk-significant decisions using a systematic process, in that, a formal evaluation was not used to determine equivalent capacity (H.1.a). (Section 1R05)

- Green. The inspectors identified a Green non-cited violation of 10 CFR 50.55a(g)(4) for failure to meet requirements of ASME Section XI for the replacement of safety injection valve 2-SI-82 performed during the last Unit 2 refueling outage. The licensee failed to perform a visual examination for leakage of the upstream pipe to valve weld and failed to obtain the Authorized Nuclear Inservice Inspection (ANII) signature on the NIS-2 Form. The NRC relies on the ANII approval to ensure Code compliance. The licensee promptly entered the issue into their Corrective Action Program as condition report CR024453 for resolution during the next outage.

  The finding is more than minor because it affects the Mitigating Systems cornerstone objective and is associated with the cornerstone attribute of operability, availability, and reliability of a mitigating system, in that, the system was not properly tested and certified after repair/replacement activities. The inspectors assessed the finding using the Significance Determination Process and determined the finding to be of very low safety significance because there was no loss of operability of the safety injection system. A contributing cause of the violation was related to the Decision-Making component of the cross-cutting area of Human Performance, and the aspect of using conservative assumptions in decision making, in that, the licensee knew they did not meet the intent of the Code requirement but mistakenly believed that they met the Code as-written. (H.1.b). (Section 1R08)

- Green. A Green self-revealing non-cited violation of Technical Specification 6.4, "Unit Operating Procedures and Programs," was identified for failure to have an adequate maintenance procedure for the emergency service water (ESW) pump strainer. This resulted in the emergency service water pump 1-SW-P-1B being declared inoperable. This procedure failed to provide adequate instructions for the reassembly of ESW strainer 1-SW-STR-4B. The finding was documented in the licensee's corrective action program as condition report CR023818. Corrective action was taken to restore pump operability and to correct the procedure and post-maintenance test error.

  The finding is more than minor, because it is associated with the operability, availability, reliability, or function of a system or train in a mitigating system. This finding was evaluated using the Significance Determination Process and was determined to be of very low safety significance because it did not result in a loss of safety function or the loss of a single train of ESW for more than the allowed Technical Specification outage time. This finding has a cross-cutting aspect in the area of human performance work practices (H.4.a), because personnel proceeded in the face of uncertainty when they continued to re-assemble the strainer operating mechanism without the requisite work instructions. (Section 1R19)

- Green. The NRC identified a Green NCV of Unit 1 and Unit 2 Operating License Condition 3.I because the installed carbon dioxide (CO_2) fire suppression systems could not be shown to deliver the design basis gas concentration. This finding applied to the Unit 1 and Unit 2 normal switchgear rooms, the Unit 2 cable tunnel, and the Unit 1 and Unit 2 cable vaults. The licensee had implemented or initiated system modifications to address this violation.

  The finding is more than minor because it affects the Mitigating Systems cornerstone objective of ensuring reliability and capability of systems that respond to initiating events and the cornerstone attribute of protection against external factors, i.e. fire. The finding was determined to be of very low safety significance in a Significance Determination Process Phase 3 analysis. For the cable vault areas, the analysis showed that fires could initiate scenarios which could challenge the mitigating systems. However, the risk of these scenarios was calculated to be in the very low significance band. Analysis with respect to the normal switchgear rooms led to the conclusion that it was of very low safety significance primarily due to the frequency of fires potentially challenging mitigating systems being relatively low and the availability of unaffected safety-related shutdown systems. The finding for the Unit 2 cable tunnel was also of very low safety significance because it did not have any significant fixed ignition sources (cables were thermoset type) and the probability for transient combustible fires or hot work initiated fires damaging important cables was judged to be low. (Section 4OA5.1)

B. Licensee-Identified Violation

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking number is listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period operating at full Rated Thermal Power (RTP). The Unit separated from the electrical grid on October 20 at 12:18 a.m. and the reactor was shutdown at 12:51 a.m. to commence refueling outage (RFO) 21. Major activities accomplished during the RFO included: refueling, replacement of all three reactor coolant pumps seals, replacement of a reactor coolant and residual heat removal pump motor, safety injection check valve repairs, turbine rotor inspections, steam generator inspections, and containment sump modification. Criticality was achieved at 12:54 p.m. on November 30. The Unit connected to the electrical grid on December 1 at 6:19 p.m. and subsequently reached 100% RTP on December 3 at 5:21 p.m. The Unit continued to operate at or near full RTP for the remainder of the inspection period.

Unit 2 operated at RTP throughout the inspection period with the exception of brief power reduction to 90% RTP on October 4 for planned maintenance and testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors performed a seasonal review of the licensee's cold weather preparations. The inspectors reviewed licensee procedures 0-OSP-ZZ-001, "Cold Weather Preparations," and 0-EPM-1303-01, "Freeze Protection Inspection." The inspectors walked down portions of the turbine building, safeguards rooms, fire pump house, emergency diesel generators (EDGs), high level intake structure, low level intake structure, refueling water storage tanks (RWSTs), and condensate storage tanks (CSTs) to assess condition and operability of heat tracing, heaters, and insulation. The inspectors observed equipment condition to determine system readiness for cold weather. The inspectors reviewed the Update Final Safety Analysis Report (UFSAR) and Technical Specifications (TSs) requirements to verify that these systems would remain operable during cold weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

The inspectors performed four partial walkdowns of the risk-significant systems listed below to verify the systems were correctly aligned to perform their designated safety function. The walkdowns occurred during periods when the redundant train or system was out-of-service for maintenance and/or testing or following realignment after an extended system outage. The positions of critical valves, breakers, and control switches, required for system operability, were verified in the correct configuration by field walkdown and/or review of the main control board. To ascertain the required system configuration, the inspectors reviewed plant procedures, system drawings, the UFSAR, and the Technical Specifications. References used for this review are listed in the attachment to this report. The inspectors reviewed Dominion's corrective action program to verify that equipment alignment problems were being identified and properly resolved.

- Unit 2 auxiliary feedwater (AFW) system (while the AFW cross-tie from Unit 1 to Unit 2 was out-of-service for maintenance)
- Unit 1 AFW system prior to startup from RFO 21
- Unit 1 safety injection system prior to startup from RFO 21
- Unit 1 containment spray system prior to startup from RFO 21

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 Residual Heat Removal (RHR) System to review the system alignment and condition. The walkdown emphasized pump and piping overall condition, status of boric acid leaks and associated targets, plant issues associated with system deficiencies, valve and breaker position verifications, and component labeling. The documents reviewed by the inspectors are listed in the Attachment of this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Fire Protection - Tours

a. Inspection Scope

The inspectors toured seven areas of the plant and reviewed licensee documents to evaluate the fire protection program operational status and material condition, and the adequacy of conditions related to: (1) control of transient combustibles and ignition sources; (2) fire detection and suppression capability; (3) manual firefighting equipment and capability; (4) passive fire protection features; and (5) compensatory measures established for out-of-service, degraded or inoperable fire protection equipment, systems, or features. The inspectors reviewed the corrective action program to verify fire protection deficiencies were being identified and properly resolved. The references used for this review are listed in the attachment to this report. This inspection activity represents nine samples.

- Unit 1 Containment: 3 samples during RFO 21
- Unit 1 Turbine Building Operating Level El. 58' 6"
- Unit 2 Turbine Building Operating Level El. 58' 6"
- Unit 2 Cable Spreading Room
- Unit 2 Mechanical Equipment Room No. 4
- Emergency Service Water Pump House
- Fire Pump House

b. Findings

Introduction: An NRC-identified, Green, non-cited violation (NCV) of paragraph (a)(1) of 10 CFR 50.48, "Fire Protection," was identified for failure to maintain the fire suppression capability for the Unit 1 containment building as specified by the approved fire protection plan. On October 27, 2007, the licensee failed to provide equivalent fire suppression capacity when the Unit 1 containment fire hose stations were removed from service for repair.

Description: On October 27, 2007, the licensee removed the Unit 1 containment hose stations from service for repairs. To provide equivalent fire suppression capacity, the licensee placed a red gang box marked "Containment Fire Hose Only" outside the containment personnel hatch. This box contained 26 sections of 1 ½" fire hose and extra nozzles and spanner wrenches. The hoses would be attached to hose stations in the Auxiliary Building to fight a fire inside containment. Training on this equivalent fire suppression capacity was provided to fire brigade members.

In response to the inspectors' questions concerning the adequacy of these measures, the licensee performed calculation ME-0838 to determine if the 1 ½" hose would provide sufficient pressure/flow for all portions of containment. This calculation determined that equivalent fire suppression capacity was not available using 250' of 1 ½" hose needed to service all containment areas. However, 250' of 2 ½" fire hose which were not pre-staged, would have been sufficient. Since the Unit 1 containment fire hose stations were returned to service, on November 14, before the question was resolved, no immediate corrective actions were taken. The licensee entered this issue into their corrective action program as CR025073. Identified corrective actions include providing equivalent capacity determinations for other hose stations.

Analysis: This finding, a failure to provide equivalent fire suppression capacity, was a performance deficiency and is associated with the Mitigating Systems Cornerstone. In accordance with IMC 0612 Appendix B, "Issue Screening," the finding is more than minor because it was associated with degradation of a fire protection feature. Using IMC 0612 Appendix F, Attachment 1, "Part 1: Fire Protection SDP Phase 1 Worksheet," the finding was determined to be of very low safety significance, Green, because it involved low degradation of the fixed fire protection system. The low degradation rating was based upon the containment fire detection system being in service and the ability to obtain a larger fire hose if necessary.

The inspectors determined that the use of engineering judgement versus a formal evaluation was a significant cause of this performance deficiency. Failure to use a formal evaluation is directly related to the Decision-Making component of the cross-cutting area of Human Performance and the aspect of making safety-significant or risk-significant decisions using a systematic process (H.1.(a)).

Enforcement: Paragraph (a)(1) of 10 CFR 50.48 states, in part, "Each operating nuclear power plant must have a fire protection program that satisfies Criterion 3 of Appendix A to this part." Surry Unit 1 Operating License DPR-32, Condition 3.I, specifies, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR and as approved in the SER dated September 19, 1979, and subsequent supplements. UFSAR Section 9.10.2.2.4 states that: "The reactor containment of each unit is protected with normally dry interior hose stations." Contrary to the above, on October 27, 2007, the licensee failed to maintain in effect the provisions of the fire protection program as described in the UFSAR, in that, equivalent fire suppression capacity was not provided when the Unit 1 containment hose stations were removed from service. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program as CR025073, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. The finding is identified as NCV 05000280/2007005-01, Temporary Fire Suppression Capacity Not Equivalent To Unit 1 Containment Fire Hose Stations.

.2 Annual Fire Protection - Drill Observation

a. Inspection Scope

The inspectors observed a fire brigade drill on December 13 to evaluate the readiness of the licensee's personnel to fight fires. Specific aspects evaluated were: use of protective clothing and self-contained breathing apparatus; fire hose deployment and reach; approach into the fire area; effectiveness of communications among the fire brigade members and the control room; sufficiency of fire fighting equipment brought to the fire scene; and the drill objectives and acceptance criteria.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08P, Unit 1)

.1 Piping Systems ISI

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping system boundaries. The inspectors reviewed a sample from activities performed during the Unit 1-Fall 2007 / Refueling Outage (1SR21) including: 1) nondestructive examinations (NDE) required by the 1998 Edition, 2000 Addenda, of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2) examinations of reactor pressure vessel (RPV) head and head penetrations in accordance with NRC Order EA-03-009, 3) Boric Acid Program activities done in response to GL 88-05, 4) Steam Generator examination activities in accordance with Technical Specifications and industry guidelines, and 5) augmented examination commitments.

Specifically, the inspectors reviewed NDE procedures, reports, equipment calibration and certification records and personnel qualification records, for the following NDE activities:

- Ultrasonic (UT) examination of weld 12-RC-10/1-08
- Liquid Penetrant (PT) examination of weld 12-RC-10/1-08SW-51
- PT examination of weld 1-RHE-1B-07-weld 6
- UT examination of 4-inch weld RC-14/1-11
- Visual (VT) bare metal visual examinations of the Reactor Vessel Head and head penetrations
- Visual Examinations of Reactor Vessel Bottom Head and Bottom Mounted Instrument Connections

The following recordable indications, accepted by the licensee for continued service since the previous refueling outage, were reviewed:

- UT examination indications on 30-inch Main Steam line SHP-1-601, Weld #2-26

The inspectors reviewed the following Repair/Replacement Activities for compliance with ASME Code. Specifically, the inspectors reviewed weld process control sheets, welder operating instructions, welding procedure specifications, welding procedure qualification records, welder qualification records, Certified Material Test Reports for weld material, and NDE reports.

- Repair/Replacement plan 2006-424, replacement of Unit 2 Safety Injection Valve 2-SI-82

- Repair/Replacement plan 2007-209, replacement of Steam Generator Channel Head Drain for 01-RC-E-1B
- Repair/Replacement plan 2007-208, replacement of Steam Generator Channel Head Drain for 01-RC-E-1A
- Repair/Replacement plan 2007-210, replacement of Steam Generator Channel Head Drain for 01-RC-E-1C

b. Findings

Introduction: The inspectors identified an NCV of 10 CFR 50.55a(g)(4) for failure to meet the requirements of ASME Section XI on Unit 2. Specifically for safety injection valve 2-SI-82, replaced during the last Unit 2 refueling outage, the licensee failed to perform a VT-2 examination for leakage of the upstream pipe to valve weld, and failed to obtain the ANII signature on the NIS-2 Form.

Description: During review of the replacement plan for the ASME nuclear class 1 valve 2-SI-82, the inspectors identified that the licensee had not performed the required VT-2 examination during pressure tests conducted on the upstream pipe to valve weld, and did not obtain the Authorized Nuclear Inservice Inspectors (ANII) signature on the ASME Section XI form NIS-2, as required by article IWA-4000 of the 1998 Edition, 2000 Addenda, of ASME Section XI. The failure to perform the requirements of Section XI Code for replacement of a nuclear class 1 component was a violation of 10CFR50.55a(g)(4).

In October of 2006 an ASME Nuclear Class 1 safety Injection valve 2-SI-82 was replaced in accordance with the requirements of the 1998 Edition, 2000 Addenda, of ASME Section XI (Code). Article IWA-4000 of the Code requires NDE and a visual examination (VT-2) for leakage be conducted at RCS pressure, and form NIS-2 be completed and signed by the ANII, which certifies that all requirements have been met. In April 2007, while performing a post-work review of the Work Order # 600156-02 associated with the replacement of valve 2-SI-82, the licensee identified that the upstream pipe to valve weld had not received a VT-2 visual examination for leakage at RCS pressure and issued a Non-Conformance Report CR 009408. The licensee identified that they had performed the required UT and PT examinations, and performed VT-2 examinations on the upstream pipe to valve weld at less than RCS pressure. The licensee closed the CR with the conclusion that, in accordance with their interpretation of the Code, the VT-2 examinations conducted on 2-SI-82 valve replacement, at less than RCS pressure, met the Code requirement. The licensee identified that they realized that this position did not meet the intent of the Code requirement but met the Code as-written. The licensee also performed an operability determination and concluded that the appropriate construction code and preservice examination (liquid penetrant, radiography and ultrasonic examination) were sufficient to confirm pressure boundary integrity of the upstream weld and that this condition did not impact the safe operation of the unit.

The inspectors determined that the Code requirements for pressure testing repair/replacement of pressure retaining components are identified in Articles IWA-4000, IWA-5000, and for Class 1 components, in article IWB-5000. Article IWA-4000 of the Code requires repair/replacement activities performed by welding on a pressure retaining boundary shall include a system leakage test performed in accordance with IWA-5000. Article IWA-5000, in section IWA-5120(a), requires class 1 components, subject to repairs/replacement activities, be pressure tested at conditions (pressure and temperature) specified in IWB 5000. Article IWB-5000, section IWB-5221(a), states that a system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power (RCS pressure). Contrary to this requirement the inspectors found that the licensee conducted the VT-2 examination on the upstream pipe to valve weld at less than RCS pressure.

ASME Section XI, article IWA-4180, requires form NIS-2 be completed for all repair/replacement activities and, in article IWA-6000, requires form NIS-2 be maintained by the owner and be submitted to regulatory authorities with the Inservice Inspection Summary Report within 90 days of the completion of the Inservice Inspection conducted during each refueling outage. Upon completion of the valve 2-SI-82 replacement activities, the licensee completed and signed the NIS-2 form, however, they failed to obtain the ANII signature on the NIS-2 form. The inspectors found that this condition was not identified by the licensee. The NRC holds the signature of the ANII in high regard and relies on the ANII approval to ensure Code compliance. An incomplete NIS-2 form indicates that all requirements of the Code have not been met.

Analysis: The failure to perform a VT-2 examination at operating RCS pressure and failure to obtain the ANII signature on the NIS-2 form were performance deficiencies because the licensee was required by 10 CFR 50.55a(g)(4) to meet the requirements of the 1998 Edition, 2000 Addenda, of ASME Section XI Code.

The finding is more than minor because it affects the Mitigating Systems cornerstone objectives and was associated with the cornerstone attribute of operability, availability and reliability of a mitigating system, in that, the system was not properly tested and certified after repair/replacement activities. The inspectors assessed the finding using the SDP Manual Chapter 0609 Appendix A, and determined the finding to be of very low safety significance because there was no loss of operability of the safety injection system.

A contributing cause of the violation was related to the Decision-Making component of the cross-cutting area of human performance, and the aspect of using conservative assumptions in decision making, in that, the licensee knew they did not meet the intent of the Code requirement but mistakenly believed that they met the Code as-written. (H.1.b).

Enforcement: 10 CFR 50.55a(g)(4) requires, in part, that throughout the service life of a boiling or pressurized water reactor facility, components classified as ASME Code Class 1, 2, and 3 must meet the requirements set forth in Section XI of the ASME Code. The 1998 Edition of Section XI, requires a system leakage test be performed and a completed NIS-2 Form signed by an ANII that documents the owner has performed examinations in accordance all the requirements of Section XI. Contrary to the above, for the ASME Nuclear Class 1 valve 2-SI-82 valve replacement, the licensee did not perform the system leakage test at RCS pressure and did not have a completed NIS-2 Form signed by the ANII. Because the finding is very low safety significance and has been entered into the licensee’s corrective action program as CR 24453, this violation is being treated as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000281/2007005-02, Failure to meet ASME Section XI requirements for replacement of safety injection valve 2-SI-82.

.2 PWR Vessel Upper Head

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the RPV head and head penetrations in accordance with NRC Order EA-03-009. In addition the inspectors verified that activities performed were in accordance with the requirements of the order and that indications and/or defects detected were dispositioned in accordance with the ASME Code or an NRC approved alternative.

The inspectors reviewed NDE reports associated with VT-2 activities (examination for leakage) and Bare Metal Visual examinations performed to meet the examination requirements of the NRC Order EA-03-009. The inspectors reviewed photographic documentation and corrective actions taken for residue identified on the head surface.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Program

a. Inspection Scope

The inspectors reviewed the licensee's Boric Acid Corrosion Control Program to ensure compliance with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

The inspectors conducted an on-site record review, and an independent walkdown of the containment building, which is not normally accessible during at-power operations, to evaluate licensee compliance with their program procedures and applicable industry guidance. In particular, the inspectors verified that the licensee's visual examinations focused on locations where boric acid leaks could cause degradation of safety-significant components and that degraded or non-conforming conditions were properly identified in the licensee's corrective action program.

b. Findings

No findings of significance were identified.

.4 Steam Generators (SGs) - Unit 1

a. Inspection Scope

The inspectors reviewed licensee documentation and performed direct observation of licensee and vendor activities, conducted during the outage, and related to the eddy current examination (ECT) of tubes in Unit 1 B SG. The inspectors verified that inspection activities were being conducted in accordance with Technical Specifications and applicable industry standards. The inspectors' review of documentation included the SG Monitoring Program Pre-Outage assessment, the vendor's inspection plan, degradation assessment, pre-outage assessment, inspection procedures, site-specific Examination Technique Specification Sheets (ETSS), ECT bobbin and array probe certificates of compliance, equipment calibration certificates, calibration standard drawings, personnel qualifications, and personnel and Computer Data System site-specific performance demonstrations. The inspectors also confirmed that all areas of potential degradation (based on site-specific experience and industry experience) were being inspected.

The inspectors performed direct observation of data acquisition and analysis activities along with verification of equipment settings for ongoing data acquisition for tubes R40C50, R40C51, R41C51, R38C48, R39C48, R40C49, R30C27, R30C28, R38C48, R39C48R7C10, and R11C88. The tube at R11C88 was required to be plugged for permeability variations which could mask other relevant indications. The inspectors observed recordings of secondary side visual inspections in SG A, B, and C and reviewed documentation of secondary side inspections including vendor procedures and personnel qualifications.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related problems, including welding, and BACC program that were identified by the licensee and entered into the corrective action program as Condition Report (CR) documents. The inspectors reviewed the CRs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the report attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors observed a licensed operator simulator exam given on December 5, 2007. The exam was administered using scenario RQ-07.8-SP-1, Revision 0, and involved both operational transients and design basis events. The inspector verified that simulator conditions were consistent with the scenario and reflected the actual plant configuration (i.e., simulator fidelity). The inspector observed the crew's performance to determine whether the crew met the scenario objectives; accomplished the critical tasks; demonstrated the ability to take timely action in a safe direction and to prioritize, interpret, and verify alarms; demonstrated proper use of alarm response, abnormal, and emergency operating procedures; demonstrated proper command and control; communicated effectively; and appropriately classified events per the emergency plan. The inspector observed the evaluators' post scenario critique and confirmed items for improvement were identified and discussed with the operators to further enhance performance.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the two equipment issues described in the CRs listed below, the inspectors evaluated the licensee's effectiveness of the corresponding preventive and corrective maintenance. For each selected item below, the inspectors performed a detailed review of the problem history and associated circumstances, evaluated the extent of condition reviews, as required, and reviewed the generic implications of the equipment and/or work practice problem. Inspectors performed walkdowns of the accessible portions of the system, performed in-office reviews of procedures and evaluations, and held discussions with system engineers. Inspectors compared the licensee's actions with the requirements of the Maintenance Rule (10 CFR 50.65), VPAP 0815 "Maintenance Rule Program," and the Surry Maintenance Rule Scoping and Performance Criteria Matrix.

- CR 025180 Safety Injection System Cold Leg Check Valve Seat Leakage
- CR 025844 Wrong Direction Alarm on rod K-14

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated the following attributes for the six selected systems, structures, and components (SSCs) and activities listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of the assessed risk; (3) that, upon identification of an unforeseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that maintenance risk assessments and emergent work problems were adequately identified and resolved.

- October 23-24, Unit 1 qualitative shutdown risk assessment for the elevated risk (Orange) due to outage surveillance testing that required the applicable safety bus be de-energized.
- October 24, Unit 2 on-line risk assessment for the elevated risk (Yellow) due to emergent switchyard breaker manipulations to de-energize off-site power transmission line 567.
- November 3, Unit 1 shutdown and Unit 2 on-line risk assessments for the elevated risk (Yellow) due to the auxiliary feedwater (AFW) cross-tie for both units being unavailable and diminished electrical power availability on Unit 1.
- November 11, Unit 1 shutdown risk and Unit 2 on-line risk assessments for the elevated Unit-1 risk (Yellow) and Unit-2 Green risk due to the AFW cross-tie capability from Unit 1 to Unit 2 being tagged out (T/O), “C” emergency service water (SW) pump T/O, 2A and 2B SW headers T/O and other various Unit 1 risk significant to Unit 2 components T/O.
- November 14, Unit 1 shutdown and Unit 2 on-line risk assessments for the elevated Unit-1 risk (Yellow) and Unit 2 Green risk due to the AFW cross-tie capability from Unit 1 to Unit 2 being tagged out (T/O), “C” emergency service water (SW) pump T/O, 2A and 2B SW headers T/O and various Unit 1 risk significant to Unit 2 components T/O during 2-PT-8.6, “Recirculation Mode Transfer Signal Automatic Switchover Logic Test.”
- November 22-25, Unit 1 shutdown risk assessments for the elevated risk (Yellow) while the reactor coolant system was in a mid-loop condition to repair safety injection check valves.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed six operability evaluations affecting risk significant systems, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered; (4) if compensatory measures were involved, whether the compensatory measures were in place, would work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified the impact on TS limiting condition for operations. Documents reviewed are listed in the Attachment to this report. The inspectors reviewed the following six operability evaluations:

- Unit 2, Condition Report 22082; GSI-191 Containment Sump strainer evaluation for potential steam flashing.
- Unit 1, Condition Reports 21502, 22346, 013873 and 009891; Residual Heat Removal Pump (1-RH-P-1B) motor oil leakage.
- Unit 2, Condition Report 24453; Improper pressure test performed on upstream weld of 2-SI-82.
- Unit 1, Condition Report 24437; rejectable indications found during PT exam of the RHR heat exchanger (1-RH-E-1B).
- Unit 1, Condition Report 24932; RHR heat exchanger (1-RH-E-1B) excavated area near min wall thickness.
- Unit 1, Condition Report 22460, charging pump (1-CH-P-1B) high vibration.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed seven post-maintenance test (PMT) procedures and/or test activities, as appropriate, for selected risk significant systems to assess whether: (1) plant testing had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function. The inspectors observed testing and/or reviewed the results of the following seven tests listed below:

- Maintenance Work Order (MWO) 0767571-02; 0-ECM-1509-05, “Rising Stem Motor Operated Valve Inspection and Quiklook Testing,” for 1-CH-MOV-1286B, Unit 1 “B” Charging Pump Discharge Valve
- MWO 00791742-01; 0-MCM-0114-02, “Emergency Service Water Pump Coupling, Clutch, Right Angle Gear Drive, and related Component Maintenance.”
- MWO 00762948-1-1 and 00763044-1-1, perform equipment hatch and emergency personnel escape airlock leakage test.
- MWO 00793407-01; 1-OPT-SI-014, “Cold Shutdown Test of SI Check Valves to RCS cold Legs section 6.3" for 1-SI-79 check valve
- MWO 00722256-06; 1-OPT-SI-014, “Cold Shutdown Test of SI Check Valves to RCS cold Legs section 6.4" for 1-SI-85 check valve
- MWO 00793412-01; 1-OPT-FW-020, “Turbine Driven AFW Pump Performance Less than 350° F/450 psig” for the return to service of the TDAFW pump

- MWO 00763251-01; 1-OPT-FW-003, “Turbine Driven Auxiliary Feedwater Pump 1-FW-P-2" to demonstrate operability

b. Findings

Introduction: A self-revealing NCV of Technical Specification 6.4, “Unit Operating Procedures and Programs” was identified for failure to have an adequate maintenance procedure for the emergency service water (ESW) pump strainer. This resulted in the emergency service water pump 1-SW-P-1B being declared inoperable. This procedure failed to provide adequate instructions for the reassembly of ESW strainer, 1-SW-STR-4B. This finding was documented in the licensee’s corrective action program as condition report CR023818. Corrective actions taken to restore pump operability and to correct the procedure and post-maintenance test error appear adequate.

Description: On October 30, 2007 promptly after starting emergency service water pump 1-SW-P-1B, the pump’s diesel engine overheated and tripped (CR023733). The pump was declared inoperable and tagged out-of-service for maintenance/investigation. A licensee investigation identified that corrective maintenance performed on the pump’s discharge strainer (1-SW-STR-4B) was incorrectly accomplished on October 29, 2007 (CR 023818). Specifically, the strainer’s internal ball valves were improperly re-assembled. As a result, all flow through the pump was blocked, including the cooling water flow to the diesel engine. The inspector reviewed maintenance procedure 0-MCM-0114-02, Rev. 5; “Emergency Service Water Pump Coupling, Clutch, Right Angle Gear Drive, and Related Component Maintenance,” used to perform the strainer maintenance, and found the procedure’s instructions for re-installation of the ball valves deficient. The document only provided instructions for re-installing the upper ball valve. Instructions for the lower ball valve, found installed 180 degrees out-of-position, were omitted. The maintenance error was not identified prior to declaring the emergency service water pump (1-SW-P-1B) operable and having returned it to service. Based on the specified PMT only requiring a mechanical joint leak check, performance of the PMT was deferred by the licensee until the next scheduled pump run.

Analysis: The inspectors determined that the improper assembly of the ESW strainer represented a performance deficiency because that caused emergency service water pump 1-SW-P-1B to be inoperable. The inspectors concluded that the licensee should have stopped work, generated a condition report, and obtain the necessary instructions for re-installing the lower ball valve in accordance with licensee guidance. The finding is more than minor because it is associated with the operability, availability, reliability, or function of a system or train in a mitigating system. This finding was evaluated using the SDP and was determined to be a finding of very low safety significance because it did not result in a loss of safety function or the loss of a single train of ESW for more than the allowed Technical Specification outage time. This finding has a cross-cutting aspect in the area of human performance work practices (H.4.a), because personnel proceeded in the face of uncertainty when they continued to re-assemble the strainer operating mechanism without the requisite work instructions.

Enforcement: Technical Specification 6.4, “Unit Operating Procedures and Programs,” states in part that detailed written procedures with appropriate check-off lists and instructions be provided for preventive and corrective maintenance operations which would have an effect on reactor safety. Contrary to the above on October 29, 2007, an inadequate procedure for corrective maintenance which omitted steps to perform the re-installation of the lower ball valve, on service water strainer 1-SW-STR-4B, caused emergency service water pump 1-SW-P-1B to be inoperable. Because this failure to comply with Technical Specification 6.4 is of very low safety significance and has been entered into the licensee's corrective action program (CRs 023733, 23818, and 23739), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000280/2007005-03, Emergency Service Water pump 1-SW-P-1B inoperable due to an inadequate maintenance instruction for re-assembly of the ESW strainer.

1R20 Refueling and Other Outage Activities (Unit 1)

.1 Unit-1 Refueling Outage

a. Inspection Scope

The inspectors performed the activities described below for the Unit 1 refueling outage that began on October 30 and ended December 1, 2007.

Review of Outage Plan

The inspectors reviewed the outage risk control plan to assess the outage impact on defense-in-depth for the five shutdown critical safety functions: electrical power availability, inventory control, decay heat removal, reactivity control, and containment and to verify risk, industry operating experience, and prior site specific problems had been appropriately considered.

The inspectors routinely reviewed the refueling outage work plan and daily shutdown risk assessments accounting for schedule changes and unplanned activities to verify adequate defense-in-depth was provided for each safety function, and/or the Licensee implemented planned contingencies to minimize the overall risk where redundancy was limited or not available. Detailed risk reviews for specific high risk periods/activities are documented in section 1R13 of this report.

Monitoring of Shutdown Activities

Portions of the shutdown and cooldown were observed and operator performance assessed with respect to communications, command and control, procedure adherence, and compliance with Technical Specification cooldown limits. Upon shutdown, a containment walkdown was performed to identify structures, piping, and supports in containment with stains or deposited material that could indicate previously unidentified leakage from components containing reactor coolant and/or signs of physical damage.

Licensee control of Outage Activities

Clearance Activities - The inspectors reviewed a sample of risk significant clearance activities and verified tags were properly hung and/or removed, equipment was appropriately configured per the clearance requirement, and that the clearance did not impact equipment credited to meet the shutdown critical safety functions.

Reactor Coolant System Instrumentation - The inspectors periodically observed and verified by diverse means that associated instruments for the reactor/refueling cavity/spent fuel pool (SFP) water level, the reactor coolant and spent fuel pool temperature, and the operating residual heat removal system were functioning properly and accurately.

Electrical Power - The inspectors verified that the status of electrical systems met TS requirements and the licensee's outage risk control plan. The inspectors verified that compensatory measures were implemented when electrical power supplies were impacted by outage work activities. The inspectors verified that credited backup power supplies were available.

Residual Heat Removal and SFP System Monitoring - The inspectors observed the SFP and RHR system status and operating parameters to verify that the cooling systems operated properly. Verification included periodic review of the SFP and reactor cavity level, temperature, and RHR system flow.

Inventory Control - The inspectors reviewed actions to establish, monitor, and maintain the proper water inventory in the reactor vessel/cavity and spent fuel pool. The inspectors reviewed the plant system flow paths and configurations established for reactor makeup and verified the configurations were consistent with the outage plan.

Reactivity Control - The inspectors reviewed the outage risk plan to verify that activities, systems, and/or components which could cause unexpected reactivity changes were identified and controlled accordingly.

Foreign Material Exclusion (FME) - The inspectors reviewed implementation of licensee procedures for FME control for the open reactor vessel, reactor cavity, and SFP.

Containment Closure - The inspectors reviewed activities during the outage to control containment penetrations and to maintain the capability to achieve containment closure in accordance with the refueling operations technical specifications. Periodic tours of containment were performed to review the control of work activities and containment conditions.

Problem Identification and Resolution - The inspectors verified the licensee was identifying outage related issues and had entered them into the corrective action program.

<u>Reduced Inventory and Mid-Loop Conditions</u>

The inspectors observed the evolution to establish mid-loop conditions and verified: containment closure capability was established, redundant means of reactor vessel level and temperature indication were functioning properly, redundant means to inject water into the reactor were available, an adequate vent path existed, and procedures were in-place and briefed to address a loss of decay heat removal or RCS inventory. The inspectors reviewed the licensee's risk assessment, the Significant Schedule Change Safety review form, the Shutdown safety Assessment Checklist, and the associated contingency actions for the mid-loop configuration. Additionally, the inspectors reviewed the licensee's response to Generic Letter (GL) 88-17 and compared the specified commitments against the implementing procedure requirements. Prior to entering the reduced inventory condition, the inspector verified action was taken to monitor and record reactor vessel temperature and level at 15 minute intervals, per operations checklist OC-18. The inspectors reviewed maintenance activities planned for mid-loop and the implementation of procedure 0-OP-ZZ-005, Rev.0; "Assessment of Maintenance Activities For Potential Loss of Reactor Coolant Inventory," used to document required actions for maintenance with the potential to affect RCS inventory during reduced RCS inventory conditions.

Refueling Activities

On a sampling basis, the inspectors verified the requirements of TS 3.10, Refueling, were met, and that refueling activities were conducted in accordance with station procedures. Activities were monitored from the control room and refueling bridge to observe the communications and coordination between personnel and to verify core reactivity was controlled and fuel movement was accomplished and tracked in accordance with the fuel movement schedule. The inspectors independently verified the as-loaded core configuration matched the designed core reload configuration for Unit 1 cycle 22.

Monitoring of Heatup and Startup Activities

Prior to startup, the inspectors examined the spaces inside the containment building to verify that debris had not been left which could affect performance of the containment sumps. On a sampling basis the inspectors verified that technical specifications, license conditions, and other requirements, commitments, and administrative procedure prerequisites were met for changes in plant configurations/modes. To monitor restart activities, the inspectors performed control room observations, plant walkdowns, and reviewed main control board indicators, operator logs, plant computer information, and station procedures. Control room observations included the approach to criticality, critical operations, low power physics testing, and the synchronization of the main turbine generator to the electrical grid.

b. Findings

On 11/28/07, during Unit 1 Containment Close-out walkdown, the inspectors identified that loose bat insulation had been placed in a 15' X 5' penetration in the 'C' Loop Room. The insulation had not been found by the licensee during their containment readiness verification walkdown. When the inspectors notified the licensee, they removed two 55 gallon bags which were approximately 45 lbs of fibrous insulation. This issue was documented in the corrective action program as Condition Report CR025641. The walkdown was performed by the inspectors to verify that the containment walkdown conducted by the licensee was in accordance with procedural requirements. Later investigation found that the insulation had been there for a number of years. The inspectors reviewed the affected start-up procedure, 1-GOP-1.7 Rev 2, "Unit Startup, RCS heatup from ambient to HSD" and determined that even though the procedure had several steps in attachment 3 to ensure containment was clear of debris and fibrous material, the associated licensee walkdown failed to reveal the presence of the loose insulation. The licensee performed a more thorough walkdown of containment and verified that no other loose material was present. The issues associated with the fibrous material left in containment and the effects on the containment sump are identified as an unresolved item (URI) pending additional inspection and review from the NRC. This URI is designated 05000280/2007005-04, Fibrous Material left in Unit 1 Containment.

.2 Control of Heavy Loads

a. Inspection Scope

In response to operational experience concerns regarding reactor vessel head lifts (NRC Operating Experience Smart Sample FY2007-03), the inspectors reviewed Dominion's programs and procedures to determine whether past and current practices were within the licensing basis, and consistent with guidance in NUREG-0612, "Control of Heavy loads at Nuclear Power Plants," and the Nuclear Energy Institute's (NEI) formal initiative to ensure that heavy load lifts were conducted safely. The inspectors reviewed Dominion's actions to manage the increased risk during these activities and observed the heavy load lifts for the Unit 1 reactor vessel head removal and reinstallation. The inspectors reviewed the procedures for the heavy load lifts involving the reactor coolant and residual heat removal pump motors. The inspectors reviewed the documents listed in the Attachment of this report related to heavy load lifts and conducted discussions with licensee personnel.

b. Findings

The inspectors identified that the licensee failed to incorporate a heavy load lift analysis into their UFSAR. Failure to update the UFSAR to reflect aspects of heavy load lifts involving the reactor vessel head and include information from a reactor vessel head drop analysis was a violation of 10 CFR 50.71(e).

The NRC has found industry uncertainty regarding the licensing bases for handling of reactor vessel heads, and as a result issued EGM 07-006, "Enforcement Discretion for Heavy Load Handling Activities," on September 28, 2007. NEI has informed NRC of industry approval of a formal initiative that specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices. The NRC staff believes implementation of the initiative will resolve uncertainty in the licensing bases for heavy load handling, and enforcement discretion related to the uncertain aspects of the licensing basis is appropriate during the implementation of the initiative.

During inspection of heavy load lifts, the inspectors determined that the licensee implemented interim actions prior to the specified lifts in accordance with the industry initiative, thereby meeting the following criteria to warrant enforcement discretion:

1) The licensee had neither a single-failure-proof crane nor a load drop analysis (generic or plant-specific) that bounded the planned lifts with respect to load

weight, load height, and medium present, so the licensee conducted the head lift at the minimum practicable height and flooded the refueling cavity with water during the head movement to limit the maximum potential impact velocity of the head. The licensee maintained the bottom of the head less than 15 feet above the refueling cavity water surface when the head was lifted above the guide studs. Once the cavity was fully flooded [greater than 23 feet above the reactor vessel flange], the reactor vessel head was allowed to be lifted more than 15 feet above the water surface as necessary to lift the head above immovable structures around the refueling cavity.

2) Included the movement of heavy loads as a configuration management activity in administrative controls established to implement 10 CFR50.65(a)(4).

Therefore, consistent with EGM 07-006, we are exercising enforcement discretion for the above violation in accordance with Section VII.B.6 of the NRC Enforcement Policy and are not issuing enforcement action for the violation.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the six risk-significant SSCs listed below to assess, as appropriate, whether the SSCs met TS, the UFSAR, and licensee procedural requirements. The inspectors also determined if the testing effectively demonstrated that the SSCs were ready and capable of performing their intended safety functions.

Surveillance Tests

- 0-OPT-EG-001, Revision 44; "Number 3 Emergency Diesel Generator Monthly Start Exercise Test."
- 1-OPT-ZZ-002, Revision 24 OTO1; "ESF Actuation with Undervoltage and Degraded Voltage 1J Bus."
- 1-NSP-RX-014, Revision 12; "Rod Exercise Test"

Inservice Tests

- 1-OPT-SI-002, Revision 17; "Refueling Test of the Low Head Safety Injection Check Valves to the Cold Legs,"

Containment Valve Test

- 1-OPT-CT-201, Revision 19; "Containment Isolation Valve Local Leak Rate Testing (Type C Containment Test)," Section 6.31, "Penetration 70 Recir Spray Pump 2B Discharge"
- 1-OPT-CT-305, Revision 8; "Containment Isolation Valve Local Leak Rate Testing (Type C Containment Test)," for Penetration 90, "Containment purge exhaust valve" and Penetration 91, "Containment purge air supply valve"

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the procedurally implemented Temporary Modifications to lower the alarm setpoints for the RHR system low flow and heat exchanger outlet high temperature alarms during mid-loop operations. The inspectors verified system operability/availability was not affected, that configuration control was maintained, and procedural controls were in-place to restore the alarm setpoints. The references used for this review are listed in the attachment to this report. This activity represents one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Controls To Radiologically Significant Areas

a. Inspection Scope

Access Controls

During the inspection the licensee's program activities for monitoring workers and controlling their access to radiologically significant areas and tasks were evaluated. The inspectors evaluated the adequacy of procedural guidance, directly observed implementation of administrative and physical controls, and assessed resultant worker exposures to radiation and radioactive material.

The inspectors evaluated the licensee's procedures for posting, surveying, and controlling access to radiation areas, high radiation areas (HRA), and Very HRAs, against the requirements of 10 CFR Part 20. During tours, the inspectors evaluated radiological postings against the current radiological surveys in select areas of the auxiliary building to determine the appropriateness of the established radiological controls. In addition, the inspectors independently verified the dose rates recorded on current survey maps at various locations in plant areas. General area dose rates were compared to licensee survey records. The inspectors observed Health Physics technician (HPT) proficiency in performing and documenting the radiation surveys for observed activities.

Access controls for Locked HRAs were reviewed and discussed with Radiation Protection (RP) management and supervision. The inspectors directly inspected the licensee's designated locked doors locations and reviewed documentation to verify the condition and status of the locked doors. The inspectors also evaluated implementation of key controls and postings for Very HRAs and Locked HRAs. During the inspection, radiological controls for activities associated with the Alloy 600 Replacement and Installation of Nozzle Dam Covers were observed and discussed with cognizant licensee representatives.

The inspectors observed radiologically significant work areas within radiation areas and HRAs as well as the spent fuel pool storage area. The licensee's physical and program controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel pool were also reviewed with licensee representatives. The inspectors conducted independent radiological surveys of selected plant areas and compared the results to the licensee's surveys. Radiological postings and barricade requirements were evaluated for the observed areas.

The inspectors reviewed the extent of airborne radiological hazards and associated controls. Airborne radiological areas and resulting internal exposures since the last NRC inspection were reviewed with the licensee's technical staff. During observation of selected tasks, the use of engineering controls to minimize airborne radioactivity was evaluated.

RP program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR Part 20; the UFSAR details in Section 12, RP; TS, Section 6.4; and approved licensee procedures. Licensee documents, records, and data reviewed within this inspection area are listed in Section 2OS1 of the report Attachment.

Problem Identification and Resolution

Issues identified through RP departmental self-assessments and Corrective Action Program (CAP) documents associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to resolve the issues identified in this RP program area. Specific assessments and Plant Issue documents reviewed and evaluated in detail for this inspection area are identified in Section 2OS1 of the report Attachment.

The inspectors completed 21 of the required 21 samples for Inspection Procedure (IP) 71121.01. All samples have now been completed for this IP.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA)

Implementation of the licensee's ALARA program during the 2007 U1 RFO was observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. Reviewed activities included containment scaffolding, head disassembly, alloy 600 work, sump maintenance, and routine health physics (HP) coverage. Incorporation of planning, established work controls, expected dose rates, and dose expenditure into the ALARA pre-job briefings and Radiation Work Permits (RWP) for those activities were also reviewed. Work in progress reviews were inspected for two RWPs. Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting implementation of the ALARA program goals. Shutdown chemistry program implementation and the resultant effect on Reactor Containment Building (RCB), and Reactor Auxiliary Building (RAB) dose rate trending data were reviewed and discussed with cognizant licensee representatives.

Trends in individual and collective personnel exposures at the facility were reviewed. The inspectors examined the dose records of participants in the declared pregnant worker program during 2007 to evaluate total or current gestation doses. Applicable procedures were reviewed to assess licensee controls for declared pregnant workers. Trends in the plant’s three-year rolling average collective exposure history, outage, non-outage, and total annual doses for selected years were reviewed and discussed with licensee representatives.

The licensee's ALARA program implementation and practices were evaluated for consistency with UFSAR Chapter 11, Radioactive Wastes and Radiation Protection; 10 CFR Part 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and licensee procedures. Documents reviewed during the inspection of this program area are listed in Section 2OS2 of the report Attachment.

Problem Identification and Resolution

The inspectors reviewed the CAP documents listed in Section 2OS2 of the report Attachment that were related to the licensee’s ALARA program. The inspectors assessed the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with PI-AA-200, Corrective Action, Rev. 0.

The inspectors completed 26 of the specified line-item samples detailed in IP 71121.02.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope

Area Radiation Monitoring Instrumentation and Post-Accident Sampling Systems

The operability, availability and reliability of selected direct area radiation monitor (ARM) and continuous air monitor (CAM) equipment used for routine monitoring activities were reviewed and evaluated. The inspectors observed material condition, installed configuration, where accessible, and the results of performance checks and calibrations for selected ARMs and CAMs. In addition, the inspectors reviewed the licensee's Post-Accident Sampling System (PASS) capabilities.

Licensee program activities in this area were reviewed against requirements specified in TS 6.4; applicable licensee procedures; and Section 11 of the UFSAR. Licensee guidance documents, records and data reviewed are listed in Section 2OS3 of the report Attachment.

Personnel Survey Instrumentation

Current program guidance and its implementation to maintain operability, accuracy, and availability of selected portable survey instruments were reviewed and evaluated. The inspectors observed licensee personnel selecting, inspecting, functional testing, and subsequently using portable survey instruments for routine surveillances and job coverage. Availability of portable instruments for licensee use was evaluated through observation of instruments staged for issue, and discussion with licensee personnel. Portable instrument calibration data was evaluated for selected instruments staged for use or recently used by HPTs during coverage of tasks within the Radiologically Control Area (RCA). The instrument calibration data reviewed is listed in Section 2OS3 of the report Attachment.

Operability and detection capabilities of personnel monitoring equipment used to survey individuals exiting the RCA for external and internal contamination were evaluated. The inspectors reviewed calibration records and discussed the functional testing and testing intervals for personnel contamination monitor (PCM) and portal monitor equipment located at the RCA and protected area exits. PCM equipment detection capabilities were demonstrated using a low level mixed source that was passed through the equipment at different locations on the inspector's clothing. The operability and analysis capabilities of the licensee's whole body counting (WBC) equipment was also evaluated. Recent WBC equipment quality control (QC) data was reviewed and discussed with responsible personnel. In addition, current dry active waste stream radionuclide results were discussed with HP staff to assess current calibration practices for personnel contamination and WBC equipment.

Licensee activities associated with personnel radiation monitoring instrumentation were reviewed against 10 CFR 20.1204 and 20.1501, and applicable licensee procedures listed in Section 2OS3 of the report Attachment.

Respiratory Protection - Self-Contained Breathing Apparatus (SCBA)

The inspectors reviewed the licensee's respiratory protection program guidance and its implementation for SCBA equipment. The SCBA units staged for emergency use in the Control Room and at the entry to the auxiliary building were inspected for material condition, air pressure status, and number of units available. The inspectors reviewed and evaluated selected records associated with supplied-air quality and SCBA equipment maintenance. Control room operators and RP personnel were interviewed and training material was reviewed to assess availability of spectacle inserts and training effectiveness for air cylinder change out. The inspectors verified that training, medical, and fit test qualifications were current for selected operations, HP, and maintenance personnel. The inspectors also assessed the licensee's logistics for supplying replacement air bottles to the Control Room on a sustained basis. In addition, licensee procedures were reviewed and personnel were interviewed regarding program guidance and training.

Licensee activities associated with maintenance and use of SCBA equipment were reviewed against 10 CFR Parts 20.1703 and 50.47(b); TS 6.4; Regulatory Guide (RG) 8.15, Acceptable Programs for Respiratory Protection, Rev. 1, October 1999; American Nuclear Standards Institute (ANSI)-Z88.2-1992, American National Standard Practices for Respiratory Protection; and applicable licensee procedures. Procedures and reviewed data are listed in Section 2OS3 of the report Attachment.

Problem Identification and Resolution

Selected licensee CAP documents, including audits, self-assessments, and Plant Issues associated with ARM and CAM equipment, portable radiation detection instrumentation, and respiratory protective program activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with licensee procedure PI-FA-200, Corrective Action, Rev. 0. Specific documents reviewed and evaluated are listed in Section 2OS3 of the report Attachment.

The inspectors completed 9 of 9 required samples for IP 71121.03.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

Effluent Release Processing and QC Activities

The inspectors reviewed the effect of routine effluent liquid releases made in accordance with Offsite Dose Calculation Manual (ODCM) requirements on tritium concentrations in ground water samples reported from onsite groundwater monitoring wells. In addition, reports associated with abnormal liquid releases and corrective actions were reviewed to evaluate the potential onsite/offsite environmental impact of significant leakage/spills from onsite systems, structures, and components. Also, the inspector verified that these areas had been properly documented in the licensee's site decommissioning files in accordance with 10 CFR 50.75(g), if required. Finally, licensee current capabilities and routine surveillances to minimize and rapidly identify any abnormal leaks from liquid radioactive waste tanks, processing lines, and the Spent Fuel Pool, were reviewed in detail. The actions resulting from industry initiative for groundwater protection were discussed at length. The sites hydrological assessment was discussed, as were sampling plans, communication plans and historical spills. The hydrology section of the UFSAR was reviewed and compared to the current hydrological assessment. The Ground Water Protection Action Plan and Voluntary Communication Plan were reviewed for completeness.

Offsite dose results were evaluated against details and guidance documented in the following: 10 CFR Part 20 and Appendix I to 10 CFR Part 50; ODCM; RG 1.21; RG 1.33, Quality Assurance Program Requirements (Operation); and Surry Plant TS. Procedures and records reviewed during the inspection are listed in Section 2PS1 of the report Attachment.

The inspectors completed 1 sample from IP 71122.01 as a Regional Initiative.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization

During inspector walk-downs, accessible sections of the liquid and solid radioactive waste (radwaste) processing systems were assessed for material condition and conformance with system design diagrams. Inspected equipment included monitor tanks, resin transfer piping, resin and filter packaging components, and abandoned solidification equipment. The inspectors discussed component function, processing system changes, and radwaste program implementation with licensee staff.

The 2006 Effluent Report and radionuclide characterizations from 2005 - 2007 for each major waste stream were reviewed and discussed with radwaste staff. For primary resin and Dry Active Waste (DAW), the inspectors evaluated analyses for hard-to-detect nuclides, reviewed the use of scaling factors, and examined comparison results between licensee waste stream characterizations and outside laboratory data. Waste stream mixing and concentration averaging methodology for resins and filters was evaluated and discussed with radwaste operators. The inspectors also reviewed the licensee's procedural guidance for monitoring changes in waste stream isotopic mixtures.

Radwaste processing activities and equipment configuration were reviewed for compliance with the licensee's Process Control Program and UFSAR, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 61, and guidance provided in the Branch Technical Position on Waste Classification and Waste Form. Reviewed documents are listed in Section 2PS2 of the report Attachment.

Transportation

The inspectors directly observed preparation activities for the shipment of a Residual Heat Removal pump motor. The inspectors noted package markings and placarding, performed independent dose rate measurements, and interviewed shipping technicians regarding Department of Transportation (DOT) regulations.

Five shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors reviewed emergency response information, DOT shipping package classification, radiation survey results, and evaluated whether receiving licensees were authorized to accept the packages. Licensee procedures for opening and closing Type A and Type B shipping casks were compared to recommended vendor protocols and Certificate of Compliance (CoC) requirements. In addition, training records and training curricula for selected individuals currently qualified to ship radioactive material were reviewed.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Part 20, 10 CFR Part 71, 49 CFR Parts 172-178; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution

Selected plant issue reports in the area of radwaste/shipping were reviewed in detail and discussed with licensee personnel. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure PI-AA-200, Corrective Action, Rev. 0. The inspectors also evaluated the scope of the licensee's internal audit program and reviewed recent assessment results. Documents reviewed for problem identification and resolution are listed in Section 2PS2 of the report Attachment.

The inspectors completed 6 of 6 samples as required by inspection procedure 71122.02.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

Unrestricted Release of Materials from the Radiologically Controlled Area (RCA)

The inspectors reviewed selected program procedures and observed surveys of potentially contaminated materials released from the RCA to assess the licensee's effectiveness in preventing the improper release of radioactive material for unrestricted use. The radionuclides identified within recent waste stream analyses were compared against current calibration source radionuclide types and results to evaluate the appropriateness and accuracy of release survey instrumentation. Licensee data to evaluate survey requirements for hard-to-detect radionuclides were reviewed and discussed with responsible personnel.

The licensee practices and implementation of their monitoring activities were evaluated against 10 CFR Part 20, TS, UFSAR, and applicable procedures documented in the Section 2PS3 of the report Attachment.

Problem Identification and Resolution

Audits, self-assessments and selected licensee corrective actions associated with REMP, meteorological monitoring activities and unrestricted release of materials from the RCA were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues. Corrective action program documents were reviewed and evaluated for effective corrective actions. These documents are identified in Section 2PS3 of the report Attachment.

The inspectors completed ten of the specified line-item samples detailed in IP 71122.03.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Barrier Integrity Cornerstone

Reactor Coolant System Leakage Performance Indicator

a. Inspection Scope

The inspectors reviewed the “Reactor Coolant System Leakage” performance indicator for Units 1 and 2 from the third quarter 2006 through the third quarter 2007 to evaluate the completeness and accuracy of the data and whether the performance indicator was calculated in accordance with the guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline.” Documents reviewed included applicable monthly operating reports, and operator logs.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Cornerstone

Mitigating Systems Performance Index (MSPI)

a. Inspection Scope

The inspectors reviewed the Mitigating Systems Performance Index performance indicators for Units 1 and 2 for the fourth quarter 2006 through the third quarter of 2007 to assess the accuracy and completeness of the submitted data and whether the performance indicators were calculated in accordance with the guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline.” This evaluation included verification of compliance with the licensee’s “NRC Mitigating System Performance Index Basis Document,” and a review of selected consolidated entry forms for accuracy of information entered into the MSPI calculation computer program. Data verified for monitored components included unavailability, reliability and run times; and, number of starts, and failures to start and run. Information from logs and other plant documentation was used to verify that data was accurate. The performance indicators data gathering and entry were discussed with cognizant personnel. This inspection activity represents the following 10 samples.

- Unit-1 and 2 Emergency AC Power System
- Unit-1 and 2 High Pressure Injection System
- Unit-1 and 2 Heat Removal System
- Unit-1 and 2 Residual Heat Removal System
- Unit-1 and 2 Cooling Water Systems

b. Findings

No findings of significance were identified.

.3 Occupational Radiation Safety Cornerstone

Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed the Occupational Exposure Control Effectiveness PI results from February, 2006 through September, 2007. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and assessed corrective action program documents to determine whether HRA, VHRA, or unintended radiation exposures had occurred. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. In addition, the inspectors reviewed selected personnel contamination event data and internal dose assessment results. Report section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas. Documents reviewed are listed in sections 2OS1 and 4OA1 of the report Attachment.

b. Findings

No findings of significance were identified.

.4 Public Radiation Safety (PS) Cornerstone

Radiological Effluent Technical Specification/ODCM Radiological Effluent Occurrences

a. Inspection Scope

To evaluate the Radiological Effluent Technical Specification/ODCM Radiological Effluent Occurrences Performance Indicator, the inspectors reviewed data for the period of January 2006 through August 2007. This included records, such as monthly effluent dose calculations, that are used by the licensee to identify occurrences of quarterly doses from liquid and gaseous effluents in excess of the values specified in NEI 99-02 guidance. The inspectors reviewed a cross section of effluent release permits for the month of October 2007 including continuous and batch liquid and gas releases. The inspectors also interviewed licensee personnel that were responsible for collecting and reporting the Performance Indicator data. In addition, licensee procedural guidance for classifying and reporting Performance Indicator events was evaluated. Reviewed documents are listed in Section 4OA1 of the report Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Daily Review of Items Entered into the Corrective Action Program:

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems”, and in order to help identify repetitive, long-term, or latent equipment failures, or specific human performance issues for follow-up; the inspectors performed a daily screening of items entered into Dominion’s corrective action program. This review was accomplished by either reviewing hard copies of each condition report, attending daily screening meetings, and/or accessing and reviewing the licensee’s computerized database.

b. Findings

No findings of significance were identified.

.2 Annual Samples:

(1) Review of Unit 1 Seal Return Line Temporary Blockage

a. Inspection Scope

The inspectors performed a review of the root cause evaluation and corrective actions associated with the lifting of two relief valves on the reactor coolant pump seal return line during the Unit 1 shutdown on October 21, 2007. Approximately 1600 gallons of RCS inventory was released to the pressurizer relief tank. This issue was documented in the corrective action program as Condition Report 022876. The review was performed to ensure the full extent of the issue was identified, an adequate cause evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the plant issue against the requirements of the licensee’s corrective action program as delineated in Station Administrative Procedure VPAP-1601, “Corrective Action,” and 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action.”

b. Findings and Observations

No findings of significance were identified.

The inspectors determined that the licensee thoroughly investigated the event. The investigation included (1) development of a time line, (2) analyses of seal return line pressure and temperature data, (3) boroscope / visual inspections and component disassembly to evaluate possible causes, (4) verification of relief valve setpoints, and (5) mock-up testing to simulate plant conditions to verify that the suspected phenomenon would cause temporary blockage of the seal return line. The inspectors concluded that the licensee’s determination that boric acid had precipitated out and then re-dissolved back into solution in the seal return line heat exchanger due to an abnormal system alignment which had resulted in a higher than normal boric acid concentration was reasonable. The inspectors determined that corrective actions taken and planned adequately addressed the causes of the event; the licensee's engineering evaluations, testing, and equipment inspections properly assessed the material condition of the components which had experienced higher than normal system pressures; and that the licensee repaired equipment as necessary.

(2) Review of Operator Workaround

a. Inspection Scope

The inspectors performed an in-depth operator workaround review to verify the licensee was identifying operator workaround problems at an appropriate threshold and entering them into the corrective action program, and had proposed or implemented appropriate corrective actions. The inspectors evaluated whether the workarounds could affect multiple mitigating systems and whether the cumulative effects of operator workarounds on the reliability, availability, and potential for misoperation of a system, adversely impacted on the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspection was accomplished by document reviews, plant tours, and interviews with licensed and non-licensed operators.
The inspection focused on identification of risk significant operator workarounds involving mitigating systems to determine if the mitigating system function and or operator's ability to implement abnormal and emergency operating procedures was affected. Workarounds, formalized as long-term corrective action for a degraded or non-conforming condition were also sought out and particular attention given to identifying workarounds that increased the potential for personnel error, or:

- require operations contrary to past training or require more detailed knowledge of system than routinely provided,
- require a change from longstanding operational practices,
- require operation of a system or component in a manner dissimilar from similar systems or components,
- create the potential for the compensatory action to be performed on equipment under conditions for which it is not appropriate,
- impair access to required indications, increase dependance on oral communications, or require actions under adverse environmental conditions, and
- require the use of equipment and interfaces that had not been designed with consideration of the task being performed.

b. Findings and Observations

No findings of significance were identified that adversely impacted on the ability of operators to respond in a correct and timely manner to plant transients and accidents. Procedural direction has been provided to address long-standing design issues associated with auxiliary feedwater system and its susceptibility to high piping vibrations and excess pump flows. Incompleteness of the interim corrective actions was identified in September 2007 during the PI&R inspection and addressed in NRC IR 2007-008. On each unit, one-of-two pressurizer power operated relief valves (PORVs) was isolated due to valve seat leakage. Adequate procedural guidance exists for the main control room operator to open the block valve and restore the PORV to service if necessary.

.3 Semi-Annual Review to Identify Trends

a. Inspection Scope

A semi-annual review of plant issues was performed for the period June - December, 2007, to identify trends that might indicate the existence of a more significant safety issue. Included within the scope of this review are repetitive or closely related issues which may have been captured outside of the normal corrective action program, such as in trend reports, plant performance indicators, major equipment problem lists or equipment reliability program reports, repetitive and rework maintenance lists, challenge lists, system health reports, workaround lists, maintenance rule assessments, and self-assessments.

b. Assessment and Observations

No findings of significance were identified to indicate the existence of a more significant safety issue. However, declining valve performance, predominantly in the form of valve seat leakage, challenges operations due to abnormal system configurations, inter-system leakage, maintenance isolations, and equipment performance issues. The inspectors discussed with the licensee the indications of a decline in performance of the licensee's valve maintenance program.

4OA5 Other Activities

.1 (Closed) URI 05000280, 281/2006009-01, Carbon Dioxide Suppression System Degraded in Two Fire Areas at Unit 1 and Three Fire Areas at Unit 2

a. Inspection Scope

A Senior Reactor Analyst with support from the Probabilistic Risk Assessment Branch in the Office of Nuclear Reactor Regulation and a consultant from Sandia National Laboratory performed a Phase 3 analysis under the Significance Determination Process (SDP) for this URI, which included on-site inspection.

b. Findings

Introduction: The NRC identified a Green NCV of Unit 1 and Unit 2 Operating License Condition 3.I because the installed carbon dioxide (CO_2) fire suppression systems could not be shown to deliver the design basis gas concentration. This finding applied to in the Unit 1 and Unit 2 normal switchgear rooms, the Unit 2 cable tunnel, and the Unit 1 and Unit 2 cable vaults.

Description: The Surry CO_2 gas suppression systems were designed in accordance with National Fire Protection Association standard-12 (NFPA 12), Standard on Carbon Dioxide Extinguishing Systems. NFPA 12 requires a minimum 50 percent concentration to extinguish fires in dry electrical, wiring insulation hazards. The normal switchgear rooms, cable vaults and cable tunnels primarily contain dry electrical wiring insulation hazards in the form of cables routed in cable trays. Therefore, a minimum 50 percent extinguishing concentration would be required to protect the hazards in these areas. In two Unit 1 fire zones and three Unit 2 fire zones the licensee could not show that the 50 percent concentration could be met. The problem was identified by NRC inspectors when they performed calculations of the initial CO_2 concentration that would be achieved upon system activation. The apparent cause of this problem was that the room volumes for the subject areas were incorrectly calculated at the time of system design and allowance for leakage was not provided for in the design. Since system discharge testing had not been performed, the problem went undetected until identified by the inspectors.

Subsequent to the inspection, the Surry plant and its consultant performed complete system design calculations on all the CO_2 protected areas. They also performed a fan door test on all of the areas. The purpose of the fan door test was to determine the leakage area in the envelope of the protected spaces. The new analysis showed that the initial design was less than adequate in each of the areas. The licensee stated that the new analysis was done according to the current NFPA-12 standard, which was more stringent than the original standard in that it specified a definite hold time of twenty minutes.

With regard to the normal switchgear rooms, the fan door test showed that leakage past ventilation dampers was excessive. Corrective actions taken or planned to bring the CO_2 systems in the normal switchgear rooms into compliance included increasing the CO_2 injection time and installing seal rated, low-leakage dampers. The suppression systems in these rooms did not fall under Appendix R requirements because the rooms do not contain safe shutdown equipment. However, the suppression systems were described in the Fire Protection Report and as such, must be maintained functional pursuant to Operating License Condition 3.I.

At the time of the triennial fire protection inspection, the CO_2 system in the Unit 1 cable tunnel was considered acceptable, and this was confirmed by the licensee's new analysis. The new analysis also confirmed that the CO_2 system in the Unit 2 cable tunnel did not meet the design criteria. The system for this area was brought into full compliance by increasing the CO_2 discharge time. The licensee plans to change one ventilation damper to a seal rated damper to optimize the system.

With regard to the cable vault/penetration area, the licensee's new analysis confirmed that initial design did not achieve the desired concentration. The systems will be brought into full compliance by increasing the injection time, increasing the diameter of pipes at the system extremities and installing new nozzles.

Analysis: The finding was a performance deficiency because it was within the licensee's control to identify that the CO_2 systems in various fire areas did not meet the criterion for gas concentration contained in industry standards to which they are committed. The finding was more than minor because it was associated with the reactor safety Mitigating Systems cornerstone attribute of protection against external factors, i.e. fire, and it affected the objective of ensuring reliability and capability of systems that respond to initiating events. NRC Inspection Report No. 05000280, 281/2006009 stated the significance of the finding in the normal switchgear rooms and Unit 2 cable tunnel, taken individually, were of very low safety significance, and this was confirmed by the NRC's Phase 3 analysis.

With regard to the cable vault/penetration area, the site visit ascertained the location of cables which had been determined to be important through evaluation of the safe shutdown procedure and SDP work sheets. It was established that the Unit 2 cable vault/penetration area was more risk significant than the corresponding Unit 1 area. Fires were then postulated to start at motor control centers 2J1-2 and 2H1-2. The scenarios which could be initiated by these fires involved the need to utilize the inter-unit cross tie provided for the charging system. Also, only one flow path for auxiliary feedwater to provide secondary side cooling was available in the scenarios. However, the analysis showed that transient fires, self initiated cable fires, welding/cutting initiated fires, and hot gas layer damage were not significant risk contributors. Specifically, for the cable vault areas, the analysis showed that fires could initiate scenarios which could challenge the mitigating systems. However, the risk of these scenarios was calculated to be in the very low significance band. Analysis with respect to the normal switchgear rooms led to the conclusion that it was of very low safety significance primarily due to the frequency of fires potentially challenging mitigating systems being relatively low and the availability of unaffected safety-related shutdown systems. The finding for the Unit 2 cable tunnel was also of very low safety significance because it did not have any significant fixed ignition sources (cables were thermoset type) and the probability for transient combustible fires or hot work initiated fires damaging important cables was judged to be low. In summary, the Phase 3 analysis determined that the sum of the risk associated with each of the areas described above was of very low safety significance.

<u>Enforcement</u>: Surry Units 1 and 2 Operating License Condition 3.I specifies that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR. UFSAR Section 9.10, Fire Protection, states that low pressure fixed carbon dioxide suppression systems are provided at the normal switchgear rooms, the service building cable tunnel and the containment cable vaults, and other areas. Also, 10 CFR 50, Appendix R, Section III.G.3 requires a fixed fire suppression for the cable tunnel and vault areas since they contain safe shutdown equipment and the licensee has chosen to utilize alternative safe shutdown methods for these areas. The Surry CO_2 gas suppression systems were designed in accordance with NFPA 12, 1968 Edition. NFPA 12, 1968, specified that an acceptable CO_2 system deliver and hold a minimum gas concentration of 50 percent in the protected area.

Contrary to the above, the CO_2 systems in the Unit 1 and Unit 2 normal switchgear rooms, the Unit 2 cable tunnel, and the Unit 1 and Unit 2 cable vaults could not deliver the 50 percent minimum gas concentration. This violation has existed since February 17, 1981, when 10 CFR 50.48 became effective. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as PI S-2006-2627 and PI S-2006-2701, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC's Enforcement Policy: NCV 05000280, 281/2007005-05, Carbon Dioxide Suppression System Degraded in Two Fire Areas in Unit 1 and Three Fire Areas in Unit 2.

.2 Independent Spent Fuel Storage Installations (07200002 and 07200055)

a. Inspection Scope

Access controls and surveillance results for the licensee's Independent Spent Fuel Storage Installation (ISFSI) activities were evaluated. The evaluation included review of ISFSI radiation control surveillance procedures and assessment of ISFSI radiological surveillance data. The inspectors toured the ISFSI facilities and observed access controls, thermoluminsescent dosimeter locations and condition, and radiological postings on the perimeter security fence. The inspectors conducted independent radiation surveys of Pads 1 and 2 general areas and compared the data with licensee survey results.

Program guidance, access controls, postings, equipment material condition and surveillance data results were reviewed against details documented in applicable sections of the UFSAR, TS; 10 CFR Parts 20 and 72, and applicable licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Section 4OA5 of the report Attachment.

One sample is documented as required to indicate that activity was performed; the majority of the effort was completed under IP 60855.1.

b. Findings

No findings of significance were identified.

.3 (Unit 1 - Closed: Unit 2 - Open) Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)

a. Inspection Scope

The inspectors verified the implementation of the licensee's commitments documented in their September 1, 2005, response (serial No. 05-212) to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors. The commitments, referenced as corrective actions in the response, included plant modifications and procedure changes. Modifications included Incore sump room drain modifications, changing recirculation spray pumps' start initiation from a time delay function to a RWST level function, piping Insulation Modifications (installation of design basis accident (DBA) qualified jacketing on calcium-silicate insulation and repair of damaged insulation), and modified containment sump strainer installation. Corrective action commitments also included program controls to assure the assumptions regarding post loss of coolant accident (LOCA) debris generation and transport remain valid.

This inspection included review of the sump screen assembly installation procedure, 10 CFR 50.59 evaluations for GSI-191 related modifications, structural debris loading calculation, and validation testing of the modified sump screen design. The inspector also reviewed the foreign materials exclusion controls and the completed Quality Assurance / Quality Control records for the screen assembly installation. The inspector reviewed the resolution of problems identified in condition reports for the Unit 1 modifications. A visual walkdown was performed to verify the screen assembly configuration being installed was consistent with drawings and the tested configuration. The installation work documentation was reviewed to verify that quality control inspection were accomplished as required. The sump screen modification was not completed during the inspection. The resident inspectors verified the completion of the containment sump screen installation at the end of the outage.

b. Findings and Observations

No findings of significance were identified.

The licensee's actions stated in their September 1, 2005 response to GL 2004-02 were complete at the end of the refueling outage. The following is a listing of the corrective action commitments listed in the licensee's GL 2004-02 response and the status:

1. Completion of activities described in the response related to modifications to the containment sump. These modifications included the following:

   - Incore sump room drain modifications. [COMPLETE]
     RWST level engineered safety feature actuation system (ESFAS) function to support GSI-191 sump modifications (recirculation spray (RS) pumps to start on RWST level instead of time delay) [COMPLETE]
   - Piping Insulation Modifications (installation of DBA qualified jacketing on type calcium-silicate insulation and repair of damaged insulation) [COMPLETE]
   - Containment sump strainer installation [COMPLETE]

2. Evaluate the adequacy of strainer design to include margin for head loss due to chemical effects once the test results to quantify the chemical precipitant impact on head loss are known. [OPEN - extension request in progress; completion scheduled for Nov. 30, 2008]

3. Completion of any corrective actions that are shown to be necessary (for components affected by downstream effects) as a result of these evaluations (long term wear). [OPEN - extension request in progress; scheduled to complete long term wear analysis and identify necessary corrective actions by Nov. 30, 2008]

4. Programmatic controls for containment debris sources will be put into existing procedures as necessary to ensure the potential containment debris load is adequately controlled to maintain the emergency core cooling system (ECCS) pump net positive suction head (NPSH) margin. [COMPLETE - Alternate action to establish new fleet procedures to control debris load]

5. The licensee will report the minimum NPSH margin in the SPS plant specific license amendment request (LAR) described in item 2(e). Note that 2(e) discusses two LARs: one to change the method of starting RS pumps from

timer relays to RWST level; and one to change the containment air partial pressure operation limits in TS Figure 3.9-1. [COMPLETE]

6. The licensee will submit the GOTHIC containment analysis methodology with plant-specific analyses that support the proposed changes to TS Figure 3.8-1 and the RS pump start method via LAR. [COMPLETE]

7. The licensee will submit a revised alternate source term (AST) LOCA analysis for Surry in December 2005 via LAR. [COMPLETE]

8. Evaluate the potential jet impingement and missile load on sump screens. [COMPLETE]

9. Evaluate down stream flow paths to determine potential blockage due to debris passing through stainer. [COMPLETE]

<u>TI 2515/166 is closed for Unit 1</u>. The outstanding commitment items related to component long term wear (item 3 above) and chemical effects (item 2 above) were not scheduled to be completed until May 2008, via an NRC approved extension request. Since these items are the same as for Unit 2, the Unit 2 inspection will encompass these items for Unit 1.

<u>TI 2515/166 remains open for Unit 2</u>. An extension was previously approved for the Unit 2 implementation of GL 2004-02 commitments. The following commitment items remain open for Unit 2 and will be inspected during the spring 2008 outage in May 2008.

1. Complete installation of screen modules. Partial installation was performed in the spring 2006 outage.
2. Resolve cal-sil insulation issue. Piping Insulation Modifications (installation of DBA qualified jacketing on type calcium-silicate insulation and repair of damaged insulation)
3. Analysis issues extended for chemical effects and long term wear of downstream components.

4OA6 <u>Meetings, Including Exit</u>

<u>Exit Meeting Summary</u>

On January 10, 2008, the inspection results were presented to Mr. Don Jernigan, and members of his staff who acknowledged the findings. The inspector asked the licensee whether any proprietary material examined during the inspection was not returned. No proprietary information was identified.

4OA7 <u>Licensee-Identified Violations</u>

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 6.4-1.B requires that procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure. Dominion Department Administrative Procedure No. HPAP-1081, Radiation Work Permit Program, Rev. 5, Section 6.2.4.b, RWP Preparation Process, states that requirements for RWP briefings with workers shall be included in the RWP preparation procedure. Radiation Work Permit No. 07-1-2112, Rev. 3, U1 RFO: RCP Maintenance and Seal Work, states that for entry into all posted High Radiation Areas a Digital Alarming Dosimeter, Telemetric DAD, or DAD with LED interface is required. Contrary to this, on October 27, 2007, a worker entered the “B” RCP Cubicle, a posted Locked High Radiation Area, in Unit 1 Containment and did not have his DAD. This was identified in the licensee’s CAP as CR 023492. This finding is of very low safety significance because an HP technician accompanied the worker in the cubicle, dose rates in the cubicle did not exceed 1,000 mR/hr, and the worker’s dose was 1 mrem.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

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NRC

E. Guthrie, Chief, Branch 5, Division of Reactor Projects, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000280/2007005-04	URI	Fibrous Material left in Unit 1 Containment (Section 1R20)

Opened and Closed

05000280/2007005-01	NCV	Temporary Fire Suppression Capacity Not Equivalent To Unit 1 Containment Fire Hose Stations (Section 1R05)

05000281/2007005-02	NCV	Failure to meet ASME Section XI requirements for replacement of safety injection valve 2-SI-82 (Section 1R08)
05000280/2007005-03	NCV	Emergency Service Water pump 1-SW-P-1B inoperable due to an inadequate maintenance instruction for re-assembly of the ESW strainer (Section 1R19)
05000280, 281/2007005-05	NCV	Carbon Dioxide Suppression System Degraded in Two Fire Areas in Unit 1 and Three Fire Areas in Unit 2 (Section 4OA5.1)

Closed

05000280, 281/2006009-01	URI	Carbon Dioxide Suppression System Degraded in Two Fire Areas at Unit 1 and Three Fire Areas at Unit 2 (Section 4OA5.1)
2515/166 (Unit 1)	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) (Section 4OA5.3)

Discussed

2515/166 (Unit 2)	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Adverse Weather Protection
DWG 11448-FMC-087A, Rev. 15, Residual Heat Removal System
DWG 11448-FMC-089A, Rev. 42, Safety Injection System
DWG 11448-FMC-089B, Rev. 12, Safety Injection System
DWG 11448-FMC-084A, Rev. 24, Containment Spray System
DWG 11448-FMC-068A, Rev. 50, Feedwater System

Section 1R05: Fire Protection
1-FS-FP-170, Rev. 1; Unit 1 Turbine Building - Operating Level Elevation 58 Feet - 6 Inches
2-FS-FP-170, Rev. 1; Unit 2 Turbine Building - Operating Level Elevation 58 Feet - 6 Inches

Section 1R08: Inservice Inspection Activities
Procedures
Inservice Inspection Manual - ASME Section XI Visual Training/Certification Program and Visual Procedure, Revision 1
0-NSP-RC-003, Visual Examination of Reactor Pressure Vessel Bottom Mounted Instrumentation (BMI), Revision 1
CM-AP-BAC-10, Boric Acid Corrosion Control Program, Revision 0
DNAP-1004, Boric Acid Corrosion Control Program, Revision 7
VPAP-1103, ASME Section XI Visual Examination Program (VT-1,2,3 & General), Revision 10
1-NPT-CS-004, Inservice System Pressure Test of Unit 2 Refueling Water Chemical Addition Tank and associated Piping, Revision 2/09/2006
1-NPT-CN-001, Inservice System Pressure Test of 1-CN-TK-I and associated Piping, Rev. 0
1-NPT-CS-001, Inservice System Pressure Test of 1-CS-TK-I and associated Piping, Rev. 0
ER-AA-NDE-VT-604, Visual Examination for Leakage of PWR Reactor Head Penetrations, Rev. 0
VPAP-0307, Repair and Replacement of ASME Section XI or High Safety Significant Components Program/Plan, Rev. 16
0-NSP-RC-003, Visual Examination of Reactor Pressure Vessel Bottom Mounted Instrumentation (BMI), Rev. 1
ER-AA-NDE-121, Dominion Written Practice for Certification of Nondestructive Examination Personnel in Accordance with ASME Section XI, Appendix VII requirements, Rev. 0
ER-AA-NDE-120, Dominion Written Practice for Certification of Nondestructive Examination Personnel, Rev. 0
ER-AA-NDE-UT-802, Ultrasonic Examination of Austenitic Piping Welds in accordance with ASME Section XI, VII, Rev. 0
NDE-PT-701, Visible Solvent Removable Liquid Penetrant Examination Procedure, Rev. 6
1-OPT-RC-10.1, Reactor Coolant Leakage Walkdown at Cold Shutdown, Rev. 9

Corrective Action and Evaluation Documents
CR 019354, Boric Acid Body-to-Bonnet Leak Found during Plant Walkdown
CR 019331, Boric Acid Packing Leaks Found during Plant Walkdown
CR 022906, Boric Acid Was Identified during Performance of 1-OPT-RC-10.1 Walkdown on the Packing for Component 1-CH-LCV-1460A

CR 022911, Boric Acid Was Identified during Performance of 1-OPT-RC-10.1 Walkdown on the Threaded Connections for Component 1-RC-ICV-3024, 3026, 3050, and 3331
CR 022926, Boric Acid Was Identified during Performance of 1-OPT-RC-10.1 Walkdown on the Packing for Component 1-SI-244, 337, and 341
CR 022928, Boric Acid Was Identified during Performance of 1-OPT-RC-10.1 Walkdown on the Packing for Component 1-SI-HCV-1850B
CR S-2006-2942-R1, Observed Dry Boric Acid Buildup at Bottom Flange of 1-SI-TK-10
CR S-2006-1283-R1, Several Boric Acid Leaks Identified while Performing Annual/Monthly Walkdowns behind Infrequently Accessed Areas
CR 024098, Row 11 Tube 88 Plugged due to Permeability Variations
CR 024135, Loose Parts found in C Steam Generator
CR 024131, Foreign Object found in A Steam Generator
CR 021305, ASME Buried Piping Testing
CR 024210, Incorrect Construction Code referenced on Repair/Replacement Plan
CR 024453, Replacement of ASME Class 1 Valve 2-SI-82, for ASME Section XI Requirements
CR 009408, Question wether ASME Section XI requirements were met.
Letter dated August 10, 2007, Virginia Electric and Power Company Response to Request for Additional Information 2006 Steam Generator Inservice Inspection Reports
Letter dated August 18, 2006, Virginia Electric and Power Company Surry Power Station Owners Activity Report
Steam Generator Monitoring Program Pre-Outage Assessment Surry Unit 1-Fall 2007
ITC-SA-06-05, Boric Acid Corrosion Program Self Assessment, dated 9/13/07
Eddy Current Examination Plan for Surry Power Station Unit 1, dated October 2007
Surry Power Station Units 1 and 2 Steam Generator Monitoring Examination Plan

Section 1R12: Maintenance Effectiveness
0-MCM-0417-01, Rev. 16; Velan Swing Check Valves Inspection and Overhaul
WO 754900-01, Open/Inspect Check Valve 38-01-SI-79
WO 754898-01, Open/Inspect Check Valve 38-01-SI-241
WO 722260-01, Open/Inspect Check Valve 38-01-SI-243
WO 722259-01, Open/Inspect Check Valve 38-01-SI-85

Plant Issues and Condition Reports
S-2004-0294, S-2005-0655, S-2005-0804, S-2005-1793, S-2005-3110, S-2005-3236, S-2005-3401, S-2005-3570, S-2005-3730, S-2005-3911, S-2005-3975, S-2005-4054, S-2005-4165, S-2005-4622, S-2005-4672, S-2005-4814, S-2005-4841, S-2005-4849, S-2005-4990, S-2005-5343, S-2006-0075, CR 5401, CR 9530, CR025844

Procedures
0-OPT-SW-001, Emergency Service Water Pump 1-SW-P-1A
0-ICM-RD-RPI-002, CERPI rod Position Indication Adjustments
1-OPT-RX-005, Control Rod Assembly Partial Movement
ET-NAF-06-0004, Revised Recovery Withdrawal Rates for Control Rod K14 in S2C20
0-AP-1.00, Rod Control system Malfunction
1-NSP-RX-014, Rod Exercise Test

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Surry Unit 1 2007 RFO Shutdown Risk Review Report (Rev. 0 - Rev. 7)
VPAP-2805, Rev. 9; Shutdown Risk Program
STA-OI-22A, Rev. 0; Conducting of Shutdown Safety Assessment
SNS-GL-03, Rev. 3; Surry Power Station, Station Nuclear Safety Guideline Guidelines for Outage Readiness Review
DNAP-2000, Rev. 6; Dominion Work Management Process
PLAP-2000, Rev. 7; Supplemental Work management Process
SEAP-0002, Rev. 10; Shift technical Advisor
Operations Check List OC-97, Protected Equipment Program (effective date 8/21/07)

Section 1R15: Operability Evaluations

OP-AA-102-1001, Rev. 0; Development of Technical Basis to Support Operability Determinations
OP-AA-102, Rev. 1; Operability determination
NRC Safety Evaluation Related to License Amendment 256

Section 1R19: Post-Maintenance Testing

1-OPT-CT-302, Rev. 7; Equipment Hatch and Emergency Personnel Escape Airlock Leakage Test
0-NPT-CT-210, Rev. 3; Type B and C Leakage Tracking

Section 1R20: Refueling and Other Outage Activities

Surry Unit 1 2007 RFO Shutdown Risk Review Report (Rev. 0 - Rev. 7)
VPAP-2805, Rev. 9; Shutdown Risk Program
STA-OI-22A, Rev. 0; Conducting of Shutdown Safety Assessment
SNS-GL-03, Rev. 3; Surry Power Station Station Nuclear Safety Guideline Guidelines for Outage Readiness Review
DNAP-2000, Rev. 6; Dominion Work Management Process
PLAP-2000, Rev. 7; Supplemental Work management Process
SEAP-0002, Rev. 10; Shift technical Advisor
1-OSP-ZZ-004, Rev. 33; Unit 1 Safety Systems Status List For Cold Shutdown/Refueling Conditions
1-OSP-ZZ-003, Rev. 34; Unit 1 Safety Systems Status List for Reactor > 200° F
0-OP-4.16, Rev. 10; Pre-Core Loading And Core Mapping Verification
1-OP-FH-001, Rev. 19; Controlling Procedure for Refueling
GMP-001, Rev 22; Heavy Load Rigging & Movement
MA-AA-101, Rev. 1; Fleet Lifting and Material Handling
1-MCM-1150-01, Rev. 8; Unit One Reactor Disassembly and Reassembly
DWG 11448-FM-150D, Safe Load Path Reactor Containment Plan EL 47'4" (Retired)
DWG 11448-FM-150K, Allowable Floor Loads During Outage Reactor Containment Operating Floor Surry Power Station Unit 1 (Retired)
UFSAR Appendix 9B, Rev. 39; Movement of Heavy Loads
Virginia Electric and Power Company North Anna and Surry Power Stations Units 1 and 2 Response to NRC Bulletin NRC 96-02, dated May 13, 1996
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads, dated July 1, 1981

Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads Surry Power Station Unit 2, dated November 16, 1981
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads North Anna Surry Power Stations, dated September 30, 1981
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2, dated December 22, 1981
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2, dated March 22, 1982
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads Surry Power Station Units 1 and 2 North Anna Power Station Units 1 and 2, dated October 18, 1982
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads Phase I Surry Power Station Units 1 and 2, dated July 26, 1983
Virginia Electric and Power Company NUREG 0612 Control of Heavy Loads Phase I Surry Power Station Units 1 and 2, dated March 30, 1984
NRC Letter dated May 16, 1984; Control of Heavy Loads (Phase I)
NRC Letter dated June 18, 1984; Control of Heavy Loads NUREG-0612 (Phase II)
WCAP-9198, Reactor Vessel Head Drop Analyses (Westinghouse Proprietary Class 3)
Unit 1 Cycle 21 Reactivity Plan for Ramp Offline for Refueling Outage
1-GOP-2.8, Rev 6; Unit Cooldown, HSD to CSD for Refueling
1-GOP-2.7, Rev. 6; Unit Shutdown, Power Decrease From Allowable Power to Unit Offline for Refueling Outage
1-OPT-RC-10.1, Rev 9; Reactor Coolant Leakage Walkdown at Cold Shutdown
Surry Unit 1 Cycle 22 BOC Reactivity Plan
1-GOP-1.7, Unit Startup, RCS Heatup From Ambient to HSD
1-OP-RX-009, Rev. 9; Dilution To Critical Conditions Following Refueling
1-OP-RX-008, Rev. 2; The Calculation of Estimated Critical Conditions Following Refueling
1-NPT-RX-008, Rev 19; Startup Physics Testing
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated January 6, 1989
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated February 3, 1989
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated May 4, 1989
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated September 29, 1989
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated October 31, 1989
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated November 6, 1990
Virginia Electric and Power Company Surry Power Station Units 1 and 2 response to Generic Letter 88-17 Loss of Decay Heat Removal, dated November 14, 1991
1-OP-RC-005, Rev. 9; Draining the RCS from Flange Level to Mid-Nozzle (Reduced Inventory)
1-PT-36-CSD, Rev. 85; Unit 1 CSD Operations Special Log
1-OP-RC-013, Rev. 7; Reactor Head Vent and Standpipe Operation
0-OP-ZZ-005, Rev. 0; Assessment of maintenance Activities for Potential Loss of Reactor Coolant Inventory

Section 1R23: Temporary Plant Modifications

1-IPM-RH-F-605, Rev. 3; RHR Heat Exchangers Bypass Flow Control Loop Calibration
1-CAL-587, Rev. 3; T-1-606 Residual HX Outlet Temperature
1-OP-RC-005, Rev. 8; draining the RCS From Flange Level TO Mid Nozzle (Reduced Inventory)
1-OP-RC-002, Rev 20; Reactor Coolant System Fill

Section 2OS1: Access Controls To Radiologically Significant Areas (71121.01)

Procedures and Guidance Documents

Dominion, Department Administrative Procedure, Procedure No. HPAP-1081, Radiation Work Permit Program, Rev. 5
Dominion, Nuclear Fleet Administrative Procedure, Procedure No. PI-AA-200, Corrective Action, Rev. 0
Dominion, Nuclear Fleet Technical, Radiation Protection, Procedure No. RP-AA-201, Access Controls for High and Very High Radiation Areas, Rev. 0
Dominion, Nuclear Health Physics (NHP), Health Physics (HP), Procedure No. C-HP-1031.023, RWP Dosimetry: Exposure Control Support, Rev. 3
Dominion, NHP, HP, Procedure No. C-HP-1032.020, Radiological Survey Criteria and Scheduling, Rev. 5
Dominion, NHP, HP, Procedure No. C-HP-1032.030, Radiation Surveys, Rev. 4
Dominion, NHP, HP, Procedure No. C-HP-1032.040, Contamination Surveys, Rev. 5
Dominion, NHP, HP, Procedure No. C-HP-1032.050, Airborne Radioactivity Surveys, Rev. 6
Dominion, NHP, HP, Procedure No. C-HP-1032.061, High Radiation Area Key Control, Rev. 4
Dominion, NHP, HP, Procedure No. C-HP-1041.022, Internal Dose Calculation Based on DAC-Hour Exposure, Rev. 3
Dominion, NHP, HP, Procedure No. C-HP-1041.023, Internal Dose Calculation Based on Radionuclide Intake, Rev. 4
Dominion, NHP, HP, Procedure No. C-HP-1061.120, Hot Particle Control, Rev. 3
Dominion, NHP, HP, Procedure No. C-HP-1081.010, Radiological Work Permits: Preparing and Approving, Rev. 9
Dominion, NHP, HP, Procedure No. C-HP-1081.020, Radiological Work Permits: RWP Briefing and Controlling Work, Rev. 6
Dominion, NHP, HP, Procedure No. C-HP-1081.040, Radiological Work Permits: Providing HP Coverage During Work, Rev. 3
Dominion, Surry Power Station (SPS), Health Physics (HP), Procedure No. HP-1071.020, Controlling Contaminated Material, Rev. 6

Radiation Work Permit (RWPs)

RWP Number (No.) 07-1-2111, Rev. 2, U1 RFO: Steam Generator Primary Side Maintenance
RWP No. 07-1-2111-3, Rev. 2, Diaphragm Removal/Replacement, HP Survey, Cover Installation and Removal, and Bowl Closeout
RWP. No. 07-1-2112, Rev. 3, U1 RFO: RCP Maintenance and Seal Work
RWP No. 07-1-2116, Rev. 3, U1 RFO: SI Check Valve Inspection and Repair
RWP No. 07-1-2120, Rev. 2, U1 RFO: 1-RC-R-1 Rx Head & Upper Internals Removal and Replacement

RWP No. 07-1-2124, Rev. 1, U1 RFO: CTMT Sump Modification
RWP No. 07-1-2124-3, Rev. 0, U1 RFO: Coatings for GSI Project
RWP No. 07-1-2124-4, Rev. 1, U1 RFO: GSI-191 Scaffolding Activities
RWP No. 07-1-2126, Rev. 2, U1 RFO: Alloy 600 Replacement

Records and Data Reviewed
2007 Unit 1 Containment 47' Annulus Area Aux Feedwater Check Valve Radiography
Contaminated Skin Dose Assessment Record for a Worker, Dated 10/30/07
Dose Determination and Evaluation Report for a Worker, Dated 10/27/07

Radiological Survey Map and Records, Map No. 5-3, Steam Generator Primary Diaphragms, Dated 10/25/07
Radiological Survey Map and Records, Map No. 100, Unit 1 Containment 47' Elevation, Dated 11/02/07
Radiological Survey Map and Records, Map No. 125, Unit 1 Containment 18'4" Elevation, Dated 11/02/07
Radiological Survey Map and Records, Map No. 127, Unit 1 Containment "B" RCP Cube, Dated 10/26/07
Radiological Survey Map and Records, Map No. 155, Unit 1 Containment Steam Generator Channel Head, Dated 10/28/07
SPS Areas with Potential to Become High Radiation Areas List
SPS High Radiation Areas List
SPS Locked High Radiation Areas List

Corrective Action Program (CAP) Documents
Audit 07-06: Radiological Protection, Process Control Program, and Chemistry, Dated 08/24/07
CR 002571, U-2 C Loop Room door has been chained and locked as a result of the base plate being loose
CR 004813, Locked High Rad Gate -7 self-closing mechanism will not close gate automatically during ingress/egress
CR 004819, Locking mechanism/latch for LHRA Gate-22 randomly sticks open causing door not to self close and latch
CR 004825, Locked High Radiation Area gates 17 and 18 (Unit 1 and Unit 2 VCT cubicles) locked gates are configured such that someone with a ladder or climbing skills may be able to circumvent the locked boundary by maneuvering over the gate
CR 011347, A worker's DAD went into "dose rate" alarm while on the boric acid flats
CR 023492, Worker entered containment without his DAD
CR 023674, An employee got a small sliver of contaminated metal in his right middle finger
Dominion, Informal Self-Assessment Report, No. SPS-SA-06-06, High Radiation Area Access, Dated 08/31/06

Section 2OS2: ALARA Planning and Controls (71121.02)
Procedures and Guidance Documents
VPAP-2101, Radiation Protection Program, Rev. 30
VPAP-2102, Station ALARA Program, Rev. 12
VPAP-2103S, Offsite Dose Calculation Manual (Surry), Rev. 13
VPAP-2105, Temporary Shielding Program, Rev. 9

C-HP-1032.020, Radiological Survey Criteria and Scheduling, Rev. 5
C-HP-1032.030, Radiation Surveys, Rev. 4
C-HP-1032.040, Contamination Surveys, Rev. 5
C-HP-1032.050, Airborne Radioactivity Surveys, Rev. 6
C-HP-1032.060, Radiological Posting and Access Control, Rev. 1
C-HP-1032.061, High Radiation Area Key Control, Rev. 4
C-HP-1061.120, Hot Particle Control, Rev. 3
C-HP-1081.010, Radiation Work Permits: Preparing and Approving, Rev. 9
C-HP-1081.040, Radiation Work Permits: Providing HP Coverage During Work, Rev. 3
C-HP-1081.020, Radiation Work Permits: Rwp Briefing and Controlling Work, Rev. 6
HP-1032.100, Elemental Cobalt Sampling, Rev. 2
HP-1032.110, Standard Radiation Monitoring & Dose Rate Trending, Rev. 0
HP-1032.120, Radiation Hot Spot Program Survey Criteria and Scheduling, Rev. 0
HPAP-1071, Radioactive Material Control, Rev. 8
RP-AP-1001, Source Term Reduction and Control, Rev. 0

Records and Data

Surry Station ALARA Committee Meeting Minutes for 9/19/06, 1/21/06,10/2/06, 10/3/06, 10/4/06, 10/10/06, 10/12/06, 10/19/06, 10/27/06, 8/23/07, 9/11/07, 9/13/07, and 9/20/07
2006 Annual ALARA Report
2006 U1 Outage Report
2006 U2 Outage Report
2007 Station Goals, Rev. 1
Memo: Surry Unit 1 EOC 20 (04-2006) Primary Chemistry Shutdown Report, 8/10/2006
Spreadsheet: Outage Day 1-16(11/5/07) Dose by RWP
Spreadsheet: Daily Dose Projections by RWP
Spreadsheet: RWP Dose Projections Revised
Spreadsheet: Top 10 RWPs for Dose By Day
Spreadsheet::Department and Station Goals for 2005, 2006 and 2007
Excellence Plan- Alternate Resin
Excellence Plan- Cobalt Reduction
Excellence Plan - Hot Spot ID and Elimination
Excellence Plan - Zinc Injection
Source Term Reduction Control Strategic Plan
Source Term White Paper
RWP 06-2-3113, U2 RFO: Gate 15 Maintenance Package
RWP 06-2-3116, U2 RFO: SI Check Valve Inspection and Repair
RWP 06-0-1105, Perform Radiography
06-2-3105, U2 RFO: Valves, Tanks, Pumps and Pipe Maintenance Excluding AOVs and MOVs
C-HP-1091.251, ALARA Program: Surveillance and Evaluation, Rev. 2

CAP Documents

CR 001341, Dose Projection for Packaging RO Filters Was 350 mRem, Only 40 mRem Was Expended
CR 001775, Scheduled Radiography (RT) Exposure Exceeded Daily Projection
CR 010517, RWP 06-2-3122, GSI-191, Exposure Was Under Projection by 26.444 Rem
CR 002671, Steam Generator Manways Removed Prior to AREVA Setup

CR 002465, U2 Lithium Addition Created An Increase In Charging and Letdown Piping [Doserates]
CR 010509, Several RWPs Exceeded 5 Rem and Were >50% Over <u>Initial</u> Projection

Section 2PS1: Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

<u>Procedures and Guidance Documents</u>
VPAP-2101, Radiation Protection Program, Rev.30
VPAP-2103S, Offsite Dose Calculation Manual (Surry), Rev. 13

<u>Records and Data</u>
Annual Radioactive Effluent Release Report- Surry Power Station, 1/1/04-12/31/05
Annual Radioactive Effluent Release Report- Surry Power Station, 1/1/05-12/31/06

<u>CAP Documents</u>
None

Section 2PS2: Radioactive Material Processing and Transportation (71121.02)

<u>Procedures, Manuals, and Guides</u>
C-HP-1020.010, Radiological Protection Failed Fuel Action Plan, Rev. 4
C-HP-1071.040, Packaging and Shipment of Radioactive Material, Rev. 3
HP-1072.020, Sampling, Analyzing, and Classifying Radioactive Waste, Rev. 3
ROP-23.09, Handling Radwaste Transport Cask CNS 14-215H, Rev. 8
VPAP-2104, Radioactive Waste Process Control Program (PCP), Rev. 5
General Order 20, Emergency Response Telephone Notification for Radioactive Material Shipments, Rev. 3
CoC No. 9168, Model No. CNS 8-120B Shipping Package, Rev. 15
PI-AA-200, Corrective Action, Rev. 0

<u>Shipping Records and Radwaste Data</u>
B2005-07, DAW, 10/20/05
B2006-1, Primary resin, 2/16/06
B2006-4, DAW, 7/19/06
SH-2006-062, Cutting tools, 11/16/06
B2007-2, Filters, 7/25/07
Radioactive Materials License SNM-1168, AREVA NP, Inc., Amendment 12
10 CFR Part 61 Radioactive Waste Stream Analysis Reports, Primary Resin, DAW, and selected Filters, 2005 - 2007
Selected HAZMAT training records, 2007

<u>CAP Documents</u>
Solid Radioactive Waste Program Evaluation, November 2005 - May 2007
S-2005-3319-E1, Notification from Barnwell that shipment B2005-5 did not have required signature in block 10 of Form 540, 6/24/05
S-2005-3676-R1, Check sources discovered in non-rad shipment, 7/26/05
S-2006-1369-R1, Filter characterized as greater than Class C, 4/18/06

S-2006-1880-R1, Limited quantity shipment received at Surry with dose rates of 2 mrem/hr, 5/6/06
CR 002020, Initial torque pass exceeded limit for primary lid on Type A cask, 10/3/06
CR 011214, Shipper's certification statement is worded wrong in shipping procedure, 4/26/07

Section 2PS3: Radiological Environmental Monitoring Program and Radioactive Material Control Program (71122.03)

Procedures, Instructions, Lesson Plans, and Manuals
Dominion, NHP, HP, Procedure No.C-HP-1033.440, Rev. 4
Dominion, NHP, HP, Procedure No.C-HP-1033.710, Rev. 2
Dominion, SPS, HP, Procedure No. HP-1033.711, Rev. 1
Dominion, SPS, HP, Procedure No. HP-1033.721, Rev. 3

Instrument Calibration and Performance Data Records
Calibration Certificates – Eberline PCM-1C, S/N 125, Dated 05/30/06 and 05/29/07
Calibration Certificates – Eberline PCM-1C, S/N 128, Dated 05/31/06 and 05/30/07
Calibration Certificates – Eberline PCM-1C, S/N 134, Dated 03/14/05, 03/23/06, and 11/15/06
Calibration Certificates – Eberline PCM-1C, S/N 1516, Dated 07/27/06 and 07/12/07
Calibration Certificates – Eberline PM-7, S/N 335, Dated 02/27/06, 08/23/06, 02/23/07, and 08/16/07
Calibration Certificates – Eberline PM-7, S/N 336, Dated 02/27/06, 08/23/06, 02/23/07, and 08/16/07
Calibration Certificates – Eberline PM-7, S/N 390, Dated 01/19/06, 07/25/06, 01/29/07, and 07/16/07
Calibration Certificates – Eberline PM-7, S/N 467, Dated 11/01/05, 05/30/06, 11/07/06, and 05/30/07
Calibration Certificates- Portable Air Sampler, SN 8079, 6827,2185-1, 8078, 7133, 7726, 12331, 7725, Dated 7/17/07
Calibration Certificates- Portable Air Sampler, SN 7724, 7124, 7131, Dated 7/2/07
Calibration Certificates- Portable Air Sampler, SN 2185-1, Dated 1/18/07
Calibration Certificates- Portable Air Sampler, SN 12331, 7726, Dated 1/17/07
Calibration Certificates- Portable Air Sampler, SN 6827, Dated 1/16/07
Calibration Certificates- Portable Air Sampler, SN 5022, 8078, Dated 1/11/07
Calibration Certificates- Portable Air Sampler, SN 7133, 8079,7725, Dated 1/4/07
Calibration Certificates- Portable Air Sampler, SN 7724,7131,Dated 1/24/07
Calibration Certificates- Portable Air Sampler, SN 8078, 7130, 6828, 5022,, Dated 7/18/06
Calibration Certificates- Portable Air Sampler, SN 12331, 7729, 7129, Dated 7/17/06
Calibration Certificates- Portable Air Sampler, SN 8079, 6827, 5022, 7130, 7131,6828, 8078,7724, Dated 1/17/06
SPS Calibration Certificates – NE Technology SAM-9/SAM-11, S/N 147, Dated 02/13/06, 08/30/06, 02/21/07, and 08/24/07
SPS Calibration Certificates – NE Technology SAM-9/SAM-11, S/N 149, Dated 02/13/06, 08/30/06, 02/21/07, and 08/21/07
SPS Calibration Certificates – NE Technology SAM-9/SAM-11, S/N 392, Dated 03/21/06, 09/12/06, 03/20/07, and 08/18/07
Calibration Certificate- Canberra Genie/CAS, Various Geometries, 8/28-9/10/06

0-HSP-REMP-0001, Surveillance: Land Use Census, Rev. 5, Dated 12/6/06
0-HSP-REMP-0002, Surveillance: Environmental Radiation Monitors, Rev. 1, Dated 10/9/07
0-1PM-MM-PRO-001, Primary Meteorological Tower Calibration, Rev. 2, Dated 9/4/07
0-1PM-MM-PRO-002, Back-up Meteorological Tower Calibration, , Dated 9/5/07

CAP Documents
CR 002714, Loss of power to an environmental air sampling station
CR 002352, Environmental air sampler found inoperable
CR 019577, Alternate clam sampling location for REMP
CR 006518, Bacons Castle environmental air sampler not operational

Section 4OA1: Performance Indicator Verification
ER-SU-SPI-1001, Rev 0; Implementation of the Consolidated Data Entry (CDE) Reporting System

Technical Report No. SE-0006, Rev. 0; NRC Mitigating System Performance Index (MSPI) Basis Document, Surry Power Station
NEI 99-02, Rev. 5; Regulatory Assessment Performance Indicator Guideline
Dominion, Department Administrative Procedure, Procedure No. HPAP-2802, NRC Performance Indicator Program, Rev. 3
Dominion, Regulatory Assessment Performance Indicators, Radiological Protection, 02/02/06 - 09/04/07

Section 4OA2: Identification and Resolution of Problems
OP-AA-1700, Operations Aggregate Impact
Unit-1 Operator Workaround List (dated September 28, 2007)
Unit-2 Operator Workaround List (dated September 28, 2007)
Unit-1 and 2 control room deficiency and disabled annunciator logs as of November 2007
Unit-1 and 2 active operational decision making items (ODMI) as of November 2007
Unit-1 and 2 open operability determinations as of November 2007
Plant Health - Top 10 list
Surry System Health Report 3rd Quarter 2007
Surry Equipment Reliability Health Report 3rd Quarter 2007
Dominion Nuclear Trend Report Surry Power Station 3rd Quarter 2007
Dominion Nuclear Trend Report Surry Power Station 2rd Quarter 2007
Memo Equipment out of service greater than 30 days, November 24 2007
Dominion, Department Administrative Procedure, Procedure No. HPAP-2802, NRC Performance Indicator Program, Rev. 3

Records and Data Reviewed
Dominion, Regulatory Assessment Performance Indicators, Radiological Protection, 02/02/06 - 09/04/07

Section 4OA5: Other Activities

Independent Spent Fuel Storage Installation (60855.1)

Procedures and Guidance Documents

0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 12
0-HSP-ISFSI-002, Nuhoms Dry Spent Fuel Storage System Surveillance, Rev.1
0-HSP-ISFSI-003, Castor V/21, Castor X/33, NAC-128, MC-10 & TN-32 Dry Storage Cask Surveillance Requirements, Rev. 0
HP-1061.500, Nuhoms Spent Fuel Cask Preparation / Loading and Transport to ISFSI, Rev

Surveillances and Records

0-HSP-ISFSI-002, Nuhoms Dry Spent Fuel Storage System Surveillance, Rev. 0, Completed 8/14-17/07
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 1/11/06
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 3/30/06
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 6/16/06
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 7/6/06

0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 10/16/06
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 1/9/07
Record Correction Notice dated 4/17/07 for 1/9/07 surveillance
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 4/10/07
0-HPT-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI) Quarterly Radiological Surveillance, Rev. 11, Completed 7/9/07

List of Documents Reviewed for TI

Design Change Notice (DCN) 05-049, NRC GSI-191 Containment Sump Strainer Design/Surry Unit 1, dated 10/16/07
DCN-06-022, NRC GSI-191 Incore Sump Room Drain Modifications, dated 5/10/07
DCN-06-041, RWST Level ESFAS Function to Support GSI-191 Containment Sump Modifications/Surry unit 1, dated 5/10/07
DCN 07-015, NRC GSI-191 Piping Insulation Modifications, Surry Unit 1, dated 10/16/07
DCP-049 Test Plan Surry Power Station, dated 10/19/07
SUR2-34325-ASD-001, AECL Assessment Document, Strainer Sizing for Surry Power Station, Units 1&2. Based on NUREG/CR-6224 correlation, Rev. 0
CM-AA-CRS-100, Program Standards, requirements and Guidance for the Containment Recirculation Sump, Rev. 0

SUR1-34325-AR-001, AECL Analysis Report, Hydraulic Performance of Replacement Containment Sump Strainers, rev. 2
CM-AA-CRS-10, Containment Recirculation Sump GSI-191 Program, Rev. 0
Condition report (CR) 011911, Material receipt discrepancies for containment sump screen metal sheets, dated 5/16/07
CR 003318, Work activities with potential to impact on the modified GSI-191 design basis, Unit 2, dated 10/27/06
CR 016660, Error in GOTHIC NPSH model over predicts NPSH available for LHSI and RS pumps, dated 7/27/07
CR 022178, Vendor supplied metal jacket insulation material does not meet purchase specification, dated 10/12/07
CR 023137, Containment coating walk down, acceptability of tape use, dated 10/25/07
Calculation SM-1476, Surry GOTHIC Analysis of NPSH Available for the LHSI and RS Pumps for Service Water TS Limit of 100 Degrees F, dated 3/12/07
Engineering Transmittal (ET) S-07-0038, High Energy Line Break for the New Containment Sump Strainer, Surry Unit 1, dated 11/13/07
Supplemental Work Instruction 05-049-001, Install GSI-191 Sump Strainer, dated 10/18/07
Calculation ME-0790, Attachment 11, GSI-191 Downstream Effects - Flow Characteristics, dated 9/20/05

LIST OF ACRONYMS

AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
ANII	Authorized Nuclear Inservice Inspector
ANSI	American Nuclear Standards Institute
ARM	Area Radiation Monitor
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
BACC	Boric Acid Corrosion Control
CAM	Continuous Air Monitor
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CO_2	Carbon Dioxide
CoC	Certificate of Compliance
CR	Condition Report
CST	Condensate Storage Tank
CY	Calendar Year
DAD	Digital Alarming Dosimeter
DAW	Dry Active Waste
DBA	Design Basis Accident
DOT	Department of Transportation
ECCS	Emergency Core Cooling System
ECT	Eddy Current Examination
EDG	Emergency Diesel Generator
ESFAS	Engineered Safety Feature Actuation System
ETSS	Examination Technique Specification Sheet
ESW	Emergency Service Water
FME	Foreign Material Exclusion
GL	Generic Letter
HP	Health Physics
HPGe	High Purity Germanium
HPT	Health Physics Technician
HRA	High Radiation Areas
I&C	Instrument and Control
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
ISI	Inservice Inspection
LAR	License Amendment Request
LOCA	Loss of Coolant Accident
MSPI	Mitigating Systems Performance Index
MWO	Maintenance Work Order
NCV	Non-cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association

NPSH	Net Positive Suction Head
ODCM	Offsite Dose Calculation Manual
OS	Occupational Radiation Safety
PASS	Post Accident Sampling System
PCMs	Personnel Contamination Monitor
PI	Performance Indicator
PMT	Post Maintenance Test
PORV	Power Operated Relief Valve
PS	Public Radiation Safety
PT	Liquid Pentrant
QC	Quality Control
RAB	Reactor Auxiliary Building
RCA	Radiologically Controlled Area
RCB	Reactor Containment Building
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RFO	Refueling Outage
RG	Regulatory Guide
RHR	Residual Heat Removal
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RS	Recirculation Spray
RTP	Rated Thermal Power
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SCBA	Self-Contained Breathing Apparatus
SDP	Significant Determination Process
SFP	Spent Fuel Pool
SG	Steam Generator
SRF	Surry Radwaste Facility
SSC	System, Structure and Component
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
U1	Unit 1
U2	Unit 2
URI	Unresolved Item
UT	Ultrasonic
VHRA	Very High Radiation Area
VT	Visual
WBC	Whole Body Counting