UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

September 22, 2017

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION - NRC DESIGN BASES ASSURANCE INSPECTION (TEAM) REPORT NUMBER 05000280/2017007 AND 05000281/2017007

Dear Mr. Stoddard:

On September 1, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Surry Power Station, Units 1 and 2, and the NRC inspectors discussed the results of this inspection with Mr. Roy Simmons and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented one finding of very low safety significance (Green) in this report. This finding involved a violation of NRC requirements. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement; and the NRC resident inspector at the Surry Power Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000280/2017007
and 05000281/2017007, w/Attachment:
Supplementary Information

cc: Distribution via ListServ

SUBJECT: SURRY POWER STATION - NRC DESIGN BASES ASSURANCE INSPECTION (TEAM) REPORT NUMBER 05000280/2017007 AND 05000281/2017007

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	050000280, 05000281
License Nos.:	DPR-32, DPR-37
Report Nos.:	05000280/2017007, 05000281/2017007
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	Surry Power Station, Units 1 and 2
Location:	5850 Hog Island Road Surry, VA 23883
Dates:	August 14 – September 1, 2017
Inspectors:	E. Stamm, Senior Reactor Inspector (Lead) G. Ottenberg, Senior Reactor Inspector S. Downey, Senior Reactor Inspector (Trainee) R. Patterson, Reactor Inspector M. Schwieg, Resident Inspector S. Gardner, Contractor W. Sherbin, Contractor
Approved by:	Jonathan H. Bartley, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

IR 05000280/2017-007, 05000281/2017-007; 8/14/2017 – 9/1/2017; Surry Power Station, Units 1 and 2; Design Bases Assurance Inspection (Team).

The inspection activities described in this report were performed between August 14, 2017, and September 1, 2017, by five Nuclear Regulatory Commission (NRC) inspectors from Region II and two NRC contractors. The team identified one non-cited violation (NCV) of very low safety significance (Green). The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, or Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," (SDP) dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements were dispositioned in accordance with the NRC's Enforcement Policy dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green: The NRC identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for the licensee's failure to correctly evaluate the heat-up of the Main Steam Valve House (MSVH), which contains the auxilliary feedwater pumps as well as other safety-related mitigating systems. The violation was entered into the licensee's corrective action program as Condition Reports 1077007 and 1077684 and the licensee conducted a preliminary calculation and evaluation to determine the actual temperature increase and determined that the equipment located in MSVH remained operable.

  The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to evaluate worst-case design conditions resulted in a decreased margin for reliability and capability of mitigating systems contained in the MSVH. The inspectors determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the qualification of a mitigating structure, system, or component (SSC), and the SSC maintained its operability or functionality. This finding was not assigned a cross-cutting aspect because the underlying cause was a legacy issue and not indicative of current performance.. (Section 1R21.2.b.1)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Design Bases Assurance Inspection (Team) (71111.21M)

.1 Inspection Sample Selection Process

The team selected risk-significant samples and related operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included risk significant structures, systems, and components (SSCs) that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1E-6. The sample included six components selected based on risk significance, one component associated with containment large early release frequency (LERF), five modifications to mitigatiing SSCs, and one operating experience (OE) item.

The team performed a margin assessment and a detailed review of the selected risk-significant components and associated operator actions to verify that the design bases had been correctly implemented and maintained. Where possible, this margin was determined by the review of the design basis and Updated Final Safety Analysis Report (UFSAR). This margin assessment also considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Inspection Manual Chapter 0326 conditions, NRC Resident Inspector input regarding problem equipment, system health reports, industry OE, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, OE, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Component and Modification Reviews

a. Inspection Scope

Components Selected Based on Risk Significance

- emergency switchgear room air handling units – 1/2-VS-AC-6/7
- turbine-driven auxiliary feedwater pumps – 1/2-FW-P-2
- emergency diesel generator number 3 fuel oil transfer system
- pressurizer power-operated relief valves – 1/2-RC-PCV-1/2455C, 1/2-RC-PCV-1/2456
- 4160V / 480V transformer – 2-EP-TRAN-2J-1
- emergency diesel generator number 2 output breaker to 2H emergency bus – 2-EP-BKR-25H3

Components with LERF Implications

- reactor trip breakers – 1/2-RP-BKR-52-RTA/RTB

<u>Modifications to Mitigation SSCs</u>

- DCP 04-076 – Oversized Breakers and Undersized Cables Replacement
- DCP SU-10-01041 – High Head Safety Injection MOV Valve Pressure Locking Modification
- DCP SU-15-01014 – Thermal Overload Replacement for Safety Related Motor-Operated Valves
- DCP SU-15-01075 – Reconfiguring Reactor Protection Trip Breaker Ladder Logic Wiring
- DCP SU-16-00107 – Reconfigure Pressurizer Heaters

For the seven components listed above, the team reviewed the plant technical specifications (TS), UFSAR, design bases documents, and drawings to establish an overall understanding of the design bases of the components. Design calculations and procedures were reviewed to verify that the design and licensing bases had been appropriately translated into these documents and that the most limiting parameters and equipment line-ups were used. Logic and wiring diagrams were also reviewed to verify that operation of electrical components conformed to design requirements. Test procedures and recent test results were reviewed against design bases documents to verify the adequacy of test methods and that acceptance criteria for tested parameters were supported by calculations or other engineering documents, and that individual tests and analyses served to validate component operation under accident conditions. Maintenance procedures were reviewed to ensure components were appropriately included in the licensee's preventive maintenance program, that components or sub-components were being replaced before the end of their intended service life, and that the licensee has appropriate controls in place for components that are beyond vendor recommended life. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action program documents were reviewed (as applicable) in order to verify that the performance capability of the component was not negatively impacted, and that potential degradation was monitored or prevented. Maintenance Rule information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current Maintenance Rule status. Component walk downs (when accessible) and interviews were conducted to verify that the installed configurations would support their design and licensing bases functions under accident conditions, and had been maintained to be consistent with design assumptions.

For the five modifications listed above, the team reviewed design bases, licensing bases, and performance capability of components to ensure they had not been degraded through modifications. In addition, post-modification testing was reviewed to ensure operability was established by verifying unintended system interactions will not occur, SSC performance characteristics continue to meet the design bases, modification design assumptions are appropriate, and modification test acceptance criteria have been met. The team also verified design basis documentation was updated consistent with the design change, verified other design basis features were not adversely impacted, verified procedures and training plans affected by the modification were updated, and verified that affected test documentation was updated or initiated as required by applicable test programs. Walk downs (when accessible) and interviews were conducted as necessary to verify that the modifications were adequately implemented. Documents reviewed are listed in the Attachment.

Additionally, the team performed the following specific reviews:

- The team reviewed completed time critical operator actions associated with tripping the reactor coolant pumps and aligning the auxiliary feedwater unit cross-tie for simulator scenarios involving loss of normal feedwater events to evaluate whether the required action times were supported by event analyses.
- The team observed a simulator scenario involving a loss of secondary side cooling requiring the alignment for bleed-and-feed core cooling in accordance with emergency operating procedures (EOPs) to evaluate the capabilities of the pressurizer power operated relief valves. The steps in the station EOPs were compared against the Westinghouse Owners Group Emergency Response Guidelines to evaluate the appropriateness of the expected actions relative to the pressurizer power operated relief valve capacity.
- The team walked down the actions required to manually trip the reactor trip breakers to evaluate the feasibility of accomplishing the action.
- The team observed various scenarios in the simulator involving the throttling of auxiliary feedwater flow to evaluate the challenge to the motor operated valves' motor duty cycle rating.

b. Findings

.1 Failure to Evaluate Design Maximum Ambient Temperature Effect on Main Steam Valve House

Introduction: The NRC identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control, for the licensee's failure to correctly evaluate the heat-up of the Main Steam Valve House (MSVH), which contains the auxilliary feedwater (AFW) pumps as well as other safety-related mitigating systems.

Description: The MSVH is ventilated by natural circulation with air entering a motor-operated, normally-open, thermostatically-controlled damper at ground level and exiting through roof vents. There is also a non-safety related room exhaust fan, which is not credited for removing any heat generated from within. The team determined that since the outside air inlet damper was not safety-related or seismically qualified, nor were the thermostat or damper motor, it cannot be credited in a design basis accident, or seismic event. Failure of the damper to stay open would result in significantly less cooling air entering the MSVH.

MSVH ventilation calculation ME-0800, "MSVH Loss of Ventilation," Revision 0, Addendum A, had previously been performed to predict the room temperature with the damper closed, but was done with an outside ambient temperature of 70 degrees Fahrenheit (°F). It was noted that the room temperature near the AFW pumps approached 120°F in this analysis, which is the maximum allowed temperature described in the plant's Environmental Zone Description for this area. The team asked the licensee if the MSVH room temperature was evaluated with the summer outdoor design temperature of 95°F, with the damper closed, and was told there was no evaluation of this design basis condition. The licensee agreed that an evaluation was necessary at the higher temperature outside with the inlet damper failed closed.

Additionally, the team determined that calucation ME-0800 contained a non-conservative loss coefficient for the modeling of the roof vent. The roof vent contains missile barriers which cause a torturous path for air flow through the vent. The pressure loss coefficient in components depends primarily on the construction of the component and the impact the construction has on the fluid flow due to change in velocity and direction of flow. The higher the loss coefficient, the higher the pressure drop through the component, thus a lower flow rate. The inspectors determined that the loss coefficient value of 2.0 used in the calculation was non-conservative.

To evaluate these conditions, a preliminary calculation was performed to include a constant outside air temperature of 95°F, inlet damper closure, and loss of the forced ventilation at the start of the AFW pump 8-hour mission time. A conservative value of 5.0 was used for the loss coefficient of the roof vents. The preliminary results indicated a maximum temperature of 140°F on the 27’ elevation, near the AFW pumps, and 160°F in the upper levels of the MSVH.

Dominion Design Engineering performed a preliminary review of the qualification documents for the components located in the MSVH, and concluded that the components would be expected to perform their function at 140°F on the 27’ elevation, and 160°F in upper levels, for the duration of the calculated MSVH room temperature evaluation. The team reviewed the results of the evaluation and qualification documents and agreed with the licensee that the components would be expected to perform their functions.

Analysis: The team determined that the licensee’s failure to evaluate design maximum outside ambient temperature when predicting room temperatures in the MSVH was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to evaluate worst-case design conditions resulted in a decreased margin which affected the reliability and capability of mitigating systems contained in the MSVH. The inspectors used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued December 7, 2016, for Mitigating Systems, and IMC 0609, App. A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the qualification of a mitigating SSC, and the SSC maintained its operability or functionality. This finding was not assigned a cross-cutting aspect because the underlying cause was a legacy issue and not indicative of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, Design Control, states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Calculation ME-0800, MSVH Loss of Ventilation, determines the maximum design basis temperatures experienced by mitigating systems contained in the MSVH. Contrary to the above, since original construction, the licensee failed to correctly translate the maximum design basis temperature into calculation ME-0800. Specifically, the licensee failed to evaluate the maximum MSVH room temperature and its potential effect on equipment operability at the higher temperature. The licensee conducted a preliminary calculation and evaluation to determine the actual temperature increase and determined that the equipment located in MSVH remained operable. The violation was entered into the licensee's corrective action program as Condition Reports 1077007 and 1077684. This violation is being treated as an NCV consistent with section 2.3.2.a of the Enforcement Policy. (NCV 05000280, 281/2017007-01, Failure to Evaluate Design Maximum Ambient Temperature Effect on Main Steam Valve House)

.3 Operating Experience

a. Inspection Scope

The team reviewed two operating experience items for applicability at the Surry Power Station. The team performed an independent review for these issues and, where applicable, assessed the licensee's evaluation and disposition of each item. The issues that received a detailed review by the team included:

- 10 CFR Part 21 Report Log Number 2013-09-01 – Wedge Pin Failure in Anchor Darling Motor Operated Double Disc Gate Valve with Threaded Stem to Upper Wedge Connections, dated July 11, 2017 (ADAMS Accession No. ML17194A825)
- OE20140 – Non-conservative Unverified Assumptions Used as Design Inputs for Calculating Thermal Overload (TOL) Sizes

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On September 1, 2017, the team presented the inspection results to Mr. Roy Simmons and other members of the licensee's staff. Proprietary information that was reviewed during the inspection was returned to the licensee or destroyed in accordance with prescribed controls.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
M. Antol, System Engineer
C. Bruce, Supervisor, Engineering Coordination
N. Dodenhoff, Supervisor, Engineering Design
B. Garber, Manager, Station Licensing
P. Hargrave, MOV Engineer
L. Helstosky, Licensing Engineer
J. Henderson, Director, Engineering
R. Herbert, Manager, Design Engineering
G. Hill, Unit Supervisor
J. Holloway, Manager, Site Engineering
R. Johnson, Manager, Operations
J. LaFlam, System Engineer
J. Lansing, System Engineer
F. Mladen, Site Vice President
J. Pelletier, System Engineer
J. Rosenberger, Director, Nuclear Safety and Licensing
J. Sears, Electrical Design Engineer
R. Simmons, Plant Manager
J. Stauffer, System Engineer
C. Watson, I&C Design Engineer

NRC personnel:
P. McKenna, Senior Resident Inspector, Surry
C. Jones, Resident Inspector, Surry
G. MacDonald, Senior Risk Analyst, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened & Closed

05000280, 281/2017007-01	NCV	Failure to Evaluate Design Maximum Ambient Temperature Effect on Main Steam Valve House (Section 1R21.b.1)

LIST OF DOCUMENTS REVIEWED

Corrective Action Documents Written as a Result of the Inspection
CR 1075852, Breaker Lifter Not Seismically Secured
CR 1075974, Insulation Resistance Acceptance Criteria is Non-Conservative
CR 1076271, Evaluate 0-AP-50.00 Wording for MCR ESGR Ventilation Equipment
CR 1076279, Basis for Assumption in Ventilation Calculation is Not Stated
CR 1076716, U2 MSVH Roof Block Ventilation Covers Eye Bolts Nuts Not Tight
CR 1076771, Possible Historical Issue With Tape Used in Mounting of RP S1 and S2 Test Switches
CR 1076945, Clarify Documentation for 2014 NRC NCV Regarding PT of 125VDC MCCBs
CR 1077007, Additional Calculation Results to Determine MSVH Temperatures
CR 1077020, NRC DBAI Team Identified DC SU-15-01014 Did Not Document Reduced TOL Margin
CR 1077040, Evaluate Need for Continuing Training on Throttle MOVs
CR 1077684, The Flow Coefficients for the MSVH Roof Vents in Two Calculations Are Different

Procedures
0-AP-13.02, Loss of ESGR Cooling, Rev. 9
0-AP-50.00, Opposite Unit Emergency (With 8 Attachments), Rev. 47
0-DRP-010, Air Operated Valve and Instrument Air Regulator Setpoints, Rev. 45
0-ECM-1201-02, Pressurizer Heater Maintenance, Rev. 18
0-EPM-2406-01, 4160/480 Volt Transformer Preventive Maintenance, Rev. 10
0-FCA-11.00, Remote Monitoring (with 7 attachments), Rev. 6
0-MCM-0701-11, Recirculation and Filtration of Emergency Diesel Underground Fuel Oil Tanks, Rev. 1
0-MCM-1935-02, Cleaning and Inspection of Fuel Oil Tanks, Rev. 2
0-MPM-0700-01, Emergency Diesel Generator Engine Eighteen Month Service and Inspection, Rev. 45
0-MPM-1900-02, Backflow Preventer, Rev. 17
0-NSP-CW-006, Canal Level Probe Removal, Inspection and Cleaning, performed 5/5/17
0-NSP-CW-006, Canal Level Probe Removal, Inspection and Cleaning, performed 6/6/17
0-NSP-CW-006, Canal Level Probe Removal, Inspection and Cleaning, performed 7/13/17
0-NSP-CW-006, Canal Level Probe Removal, Inspection and Cleaning, performed 7/28/17
0-NSP-VS-006, CR Envelope Air Conditioning Flow Measurement, performed 6/17/15
0-NSP-VS-006, CR Envelope Air Conditioning Flow Measurement, performed 8/18/15
0-OP-HS-001, Fuel Oil Storage Tanks, Rev. 25
0-OP-VS-006, Control Room and Relay Room Ventilation System, Rev. 72
0-OP-VS-006A, Control Room and Relay Room Ventilation System Alignment, Rev. 13
0-OPT-EG-005, 28 Day Freq. PT: No. 3 EDG Monthly FO Sys. Test – OC-22A, Rev. 20
0-OSP-HS-002, Static Test of Underground Fuel Oil Fill Piping, Rev. 8
0-OSP-TCA-001, Time Critical Action Validation and Verification, Revs. 12 and 13
0-VSP-M4, Flood Control Panel Trouble Alarm, Rev. 5
1-E-0, Reactor Trip or Safety Injection (with 10 attachments), Rev. 72
1-ECA_0.0, Loss of All AC Power (with 12 attachments), Rev. 42
1-ES-0.1, Reactor Trip Response (with 8 attachments), Rev. 53
1-FR-H.1, Response to Loss of Secondary Heat Sink (With 8 Attachments), Rev. 38
1-FR-S.1, Response to Nuclear Power Generation/ATWS (with 3 attachments), Rev. 27
1-IMP-C-IA-97, PCV-1455C b/u Bottled Air Low Pressure Check, performed 10/25/161-PT-8.1, Reactor Protection Logic (for normal operations), Rev. 42
1-OP-FW-001A, Auxiliary Feedwater System Valve Alignment, Rev. 7

1-OP-FW-002, Turbine Driven AFW Pump Startup and Shutdown, Rev. 23
1-OPT-EG-001, Emergency Diesel Generator Monthly Start Exercise Test, Rev. 69
1-OPT-FW-003, Operations Periodic Test, Turbine Driven AFW Pump 1-FW-P-2, Rev. 52
1-OPT-FW-003, 84 Day Frequency PT: Turbine Driven Aux. Feedwater Pump, performed 4/5/17
1-OPT-FW-021, 84 Day Frequency PT: Stroke Exercise Test of the AFW Crossover MOVs, performed 6/16/17
1-OPT-RC-001, Pressurizer PORV RFO Test, performed 11/2/16
1-PT-8.5, Consequence Limiting Safeguards Logic (Hi-Hi Train), Rev. 28
2-0PT-FW-006, Auxiliary Feedwater MOV Test, Rev. 11
2-DRP-007, Motor Operated Valve Operating Bands, Rev. 41
2-PT-8.2, Reactor Protection Logic, Rev. 26
AD-AA-102, Procedure Use and Adherence, Rev. 11
CM-AA-DDC-201, Design Changes, Rev. 20
CM-AA-TCA-101, Operator Time Critical Actions, Rev. 1
CY-AA-AUX-310, Diesel Fuel Oil Sampling and Testing, Rev. 9
DNES-VA-EEN-0011, Standard for Protective Device Settings, Rev. 1
ECM-0306-Electrical Corrective Maintenance, Rev. 24
ECM-0309-01, Control Panel Maintenance, Rev. 8
ER-AA-BPM-101, Underground Piping and Tank Integrity Program, Rev. 10
OP-AP-104, Emergency and Abnormal Operating Procedures, Rev. 4
NUS-2030, Specification for Electrical Installation, Revision 21
VPAP-0505, Writers Guide for Dual-Column Procedures, Rev. 9

<u>Drawings</u>
113E244, Reactor Protection System Surry Power Station-Unit 1, Rev. 21
11448-FB-4A, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 19
11448-FB-4C, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 13
11448-FB-4D, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 2
11448-FB-4E, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 2
11448-FB-4F, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 1
11448-FB-4G, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 2
11448-FB-4H, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 1
11448-FB-4J, Yard Fuel oil Lines, Surry Power Station – Unit 1, Rev. 2
11448-FB-25D, Ventilation and Air Conditioning, Rev. 16
11448-FB-25E, Ventilation and Air Conditioning, Rev. 22
11448-FB-038A, Flow/Valve Operating Numbers Diagram, Fuel Oil Lines, Surry Power Station Unit 1, Sheet 1, Rev. 27
11448-FB-038A, Flow/Valve Operating Numbers Diagram, Fuel Oil Lines, Surry Power Station Unit 1, Sheet 2, Rev. 49
11448-FB-038A, Flow/Valve Operating Numbers Diagram, Fuel Oil Lines, Surry Power Station Unit 1, Sheet 3, Rev. 24
11448-FB-038A, Flow/Valve Operating Numbers Diagram, Fuel Oil Lines, Surry Power Station Unit 1, Sheet 4, Rev. 3
11448-FB-041A, Chill Water System, Rev. 72
11448-FB-041B, Chill Water System, Rev. 38
11448-FM-068A, Flow and Valve Operating Numbers Diagram, Feedwater System, Sheet 3, Rev. 69
11448-FM-075C, Flow and Valve Operating Numbers Diagram, Compressed Air System, Sheet 2, Rev. 40

11448-FM-086B, Flow and Valve Operating Numbers Diagram, Reactor Coolant System, Sheet 1, Rev. 33
11448-FV-36A, Underground Fuel Oil Storage Tanks, Rev. 4
11448-RE-25H, Vertical Control Boards, Rev. 22
11448-RE-25J, Vertical Control Boards, Rev. 9
11548-ESK-11AC, Bus H Degraded and Undervoltage Protection, Rev. 5
11548-FE-1A, Main One Line Diagram Surry Power Station, Rev. 25
11548-FE-1D, 4160V One Line Surry Power Station Unit 2, Rev. 19
11548-FE-1Q, 480V One Line Diagram Emergency Switchgear 2H1 & 2J1, Rev. 15
11548-FE-19AG, Elementary Diagram EDG Engine Auxiliaries, Rev. 21
11548-FE-21Q, DC Elementary Diagram 4160 Bus Bkr 25H3 & 25H8, Rev. 13
1501075-113E244A-B, Reactor Protection System Surry Power Station-Unit 2, Rev. 0
1501075-E-801-A, Reactor Protection Trip Breaker Ladder Logic, Rev. 0
1600107-11448-FE-1E, 480 V One Line Diagram, Surry Power Station Unit 1, Rev 0
1600107-11448-FE-1F, 480 V One Line Diagram, Surry Power Station Unit 1, Rev 0
D-338840, Pressurizer PORV Drawing, Series D Valve Assy. with Model 1000 Actuator, Rev.3
S-95003-3-C-001, Site Plan & Details, Fuel Oil Line Replacement, Surry Power Station, Rev. 9
S-95003-3-C-002, Base Line Yard Plan & Profile, Fuel Oil Lines Replacement, Surry Power Station, Rev. 5
S-95003-3-C-003, Base Line Yard Plan & Profile, Fuel Oil Lines Replacement, Surry Power Station, Rev. 6
S-95003-3-S-101, Plan & Sections – Fuel Oil Line Missile Shields, Fuel Oil Line Replacement, Surry Power Station Units 1 & 2, Rev. 3

<u>Calculations</u>
85-046B, Evaluation of Installing Noise Dampening Capacitor in FT-1433 Loop, dated 2/20/85
7797.04-E-001, Effect of Maximum Fault Current on Cable from Safety Related 4KV, Rev. 0
010139.1010-US(B)-95, SBO Loss of Ventilation Transient, Rev. 1
010139.3410-M-006, Fuel Oil Pumping Requirements from the Underground Storage Tanks to the Day Tanks, Rev. 0
14257.88-S-001, Design of MSVH Roof Plugs, Rev. 2
52308.04-C-013, Reevaluation of the Over-turning Capacity of the Surry ECSTs, Rev. 0
CE-1210, Pipe Stress Analysis Of 2.5 Inch Fuel Oil Lines Buried in the Yard Between Fuel Oil Pump House and Service Building, Rev. 0
CE-1228, Evaluation of Buried EDG Fuel Oil Lines Structural Response Under the Cask Transporter Loading, Rev. 0, Addendum 00B
DEO-0096, PORV Redundant Bottle Air System Capacity, Rev. 0
DEO-0598, Underground Fuel Oil Storage Tank Pressure Requirements, Rev. 1
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CA 3052804	CR 561995	CR 1045354	CR 1075747	CR 1075764
CR 37581	CR 562327	CR 1049951	CR 1075748	CR 1075766
CR 119541	CR 564698	CR 1052133	CR 1075749	CR 1075767
CR 247650	CR 565104	CR 1052211	CR 1075750	CR 1075768
CR 398628	CR 565188	CR 1052268	CR 1075751	CR 1075769
CR 541177	CR 569764	CR 1058209	CR 1075752	CR 1075771
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38013766808	38102948149	38103601786	38103740349
38047961501	38103030140	38103601789	38103755804
38047968201	38103262896	38103629816	38103778492
38073631012	38103294407	38103634886	38103784535
38077654101	38103368197	38103639754	38103789566
38102326076	38103368198	38103639759	38103789865
38102624398	38103368201	38103672278	38103790200
38102755794	38103390153	38103673668	38103790989
38102764999	38103403942	38103685915	38103797590
38102765015	38103454875	38103690856	38103797591
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