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

March 17, 2016

Mr. Robert T. Simril
Site Vice President
Duke Energy Carolinas, LLC
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION - NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS REPORT 05000413/2017007 AND 05000414/2017007

Dear Mr. Simril:

On February 3, 2017, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Catawba Nuclear Station, Units 1 and 2, and on March 16, 2017, the NRC inspectors discussed the results of this inspection with you 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 this 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 Catawba Nuclear Station.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; and the NRC resident inspector at Catawba Nuclear Station.

This letter and its enclosure 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.: 05000413, 05000414
License Nos.: NPF-35, NPF-52

Enclosure:
Inspection Report 05000413/2017007
and 05000414/2017007 w/Attachment:
Supplemental Information

cc: Distribution via Listserv

SUBJECT: CATAWBA NUCLEAR STATION - NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS REPORT 05000413/2017007 AND 05000414/2017007

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NAME	M. RILEY	J. BOZGA	D. TERRY-WARD	G. CRESPO	F. EHRHARDT	J. BARTLEY
DATE	3/13/2017	3/13/2017	3/8/2017	3/9/2017	3/15/2017	3/17/2017
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-413, 50-414
License Nos.:	NPF-35, NPF-52
Report Nos.:	05000413/2017007, 05000414/2017007
Licensee:	Duke Energy Carolinas, LLC
Facility:	Catawba Nuclear Station
Location:	York, SC 29745
Dates:	January 30, 2017, through February 3, 2017
Inspectors:	M. Riley, Reactor Inspector (Team Leader) J. Bozga, Reactor Inspector (Region III) D. Terry-Ward, Construction Inspector G. Crespo, Senior Construction Inspector (Trainee)
Approved by:	Jonathan H. Bartley, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

Inspection Report (IR) 05000413/2017007 and 05000414/2017007; January 30 - February 3, 2017; Catawba Nuclear Station, Units 1 & 2; NRC Evaluations of Changes, Tests, and Experiments.

This report covers a one-week onsite inspection by a team of four NRC inspectors, three of which were from Region II and one from Region III. The team identified one Green non-cited violation (NCV). The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using IMC 0609, "Significance Determination Process" dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas" dated December 4, 2014. All violations of NRC requirements are 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.

<u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Barrier Integrity

- <u>Green</u>: The NRC identified a non-cited violation of Title 10 Code of Federal Regulations Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to translate the limiting high pressure design requirement of the containment valve injection water (NW) system surge chamber into station procedures. Specifically, the licensee failed to translate the NW surge chamber high pressure design limit of 85 psig from calculation CNC-1223.19-00-0004, "NW system setpoint calculation," Rev. 7, into procedure OP/1/A/6200/019, "Containment Valve Injection Water System," Rev. 36, to ensure the NW system could perform its intended safety function during a design basis accident. The licensee entered this issue into their corrective action program as action request 02096392, reviewed the issue for current and past operability, and issued an operations guide to limit the NW surge chamber pressures to 80 psig.

  The performance deficiency was determined to be more than minor because it adversely affected the design control attribute of the barrier integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to translate the 85 psig NW surge chamber pressure limit into procedures resulted in exceeding the NW surge chamber high pressure limit, which could result in an inability of the safety related nuclear service water system to provide makeup water to the NW surge chamber and result in entrainment of nitrogen gas in the surge chamber outlet. The team determined the finding to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, and heat removal components, and the finding did not involve an actual reduction in function of hydrogen igniters in the reactor containment. This finding was assigned a cross-cutting aspect of Evaluation in the Problem Identification and Resolution Area because the finding was indicative of present licensee performance, and the licensee did not thoroughly evaluate the issue identified in ARs 01912139 and 01912453 after the revision to the calculation was completed to ensure that the correct high pressure NW surge chamber design requirement would have been translated into procedures [P.2]. (Section 1R17)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 <u>Evaluations of Changes, Tests, and Experiments (71111.17T)</u>

a. <u>Inspection Scope</u>

<u>Evaluations of Changes, Tests, and Experiments</u>: The inspectors reviewed six safety evaluations performed pursuant to Title 10, *Code of Federal Regulations* (CFR) 50.59, "Changes, tests, and experiments," to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 19 screenings and one applicability determination where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issues requiring the changes, tests or experiments were resolved;
- the licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation used to support the change was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000.

This inspection constituted 26 evaluations, screenings, and/or applicability determination samples as defined in Inspection Procedure (IP) 71111.17-05. Documents reviewed are listed in the Attachment.

b. <u>Findings</u>

<u>Failure to Translate Design Requirements into Operating Procedures for NW System</u>

<u>Introduction</u>: The NRC identified a Green non-cited violation (NCV) of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to translate the limiting high pressure design requirement of the containment valve injection water (NW) system surge chamber into station procedures. Specifically, the licensee failed to translate the NW surge chamber high pressure design limit of 85 psig from calculation CNC-1223.19-00-0004, "NW System Setpoint Calculation," Rev. 7, into procedure OP/1/A/6200/019, "Containment Valve Injection Water System," Rev. 36, to ensure the NW system could perform its intended safety function during a design basis accident.

Description: Updated Final Safety Analyses Report (UFSAR) Section 6.2.4.2.2 states that the purpose of the NW system is to prevent leakage of the containment atmosphere past certain gate valves used for containment isolation following a loss of coolant accident (LOCA) by injecting seal water between the two seating surfaces of the flex wedge valves. The system consists of two independent and redundant trains, each train consisting of a surge chamber that is filled with water and pressurized with nitrogen. If the water level in the surge chamber drops below the low-low level or if the surge chamber nitrogen pressure drops below the low-low pressure after a containment isolation phase A signal, a solenoid valve in the supply line from the nuclear service water (RN) system will automatically open, assuring a safety-related makeup source to the NW system at a pressure greater than 110% of peak containment accident pressure for 30 days following the postulated LOCA.

On December 12, 2012, the licensee identified in ARs 01912139 and 01912453 that the NW surge chamber pressure should be limited to 90 psig in their station procedures to avoid nitrogen entrainment in the system and to ensure that the RN system could provide safety-related makeup water to the NW system. The pressure limit of 90 psig specified in the ARs was based on analysis provided in QA-1 calculation CNC-1223.19-00-0004. The calculation also specified that the established procedural setpoint for the NW surge chamber pressure should be limited to 85 psig to account for an instrument uncertainty of +/- 6.1 psig. As a result of inspector questioning, it was identified that the high pressure design limit of 85 psig needed to be translated into the procedure to account for instrument uncertainty, instead of the 90 psig design limit proposed in the corrective actions. Upon investigation of previous NW surge tank pressure and levels, the licensee identified several instances where the NW surge tanks had operated beyond the 85 psig pressure limit specified in the calculation. Exceeding the high pressure design limit for the NW surge chamber could result in an inability of the RN system to provide safety-related makeup water to the NW system and adversely affect the ability of the NW system to maintain a water seal to prevent leakage past certain containment isolation valves following a LOCA.

On January 31, 2017, the licensee entered this issue into their corrective action program (CAP) as action request (AR) 02096392 to revise station procedure OP/1/A/6200/019. The licensee also reviewed the issue for current and past operability of the NW system. The licensee concluded that, while individual trains of the NW system were inoperable during certain periods in the past three years, there was not a complete loss of system safety function since the other train was always available to perform its safety function if needed during a design basis accident. The licensee issued an operations guide to ensure the NW surge chamber pressures are limited to 80 psig in order to maintain margin to the 85 psig limit.

Analysis: The licensee's failure to translate the limiting high pressure design requirement of the NW surge chamber into station procedures, in accordance with Title 10 CFR Part 50, Appendix B, Criterion III," Design Control," was a performance deficiency (PD). The PD was determined to be more than minor because it adversely affected the design control attribute of the barrier integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to translate the 85 psig NW surge chamber pressure limit into procedures resulted in exceeding the NW surge chamber high pressure limit, which could result in an inability of the safety related RN system to provide makeup water to the chamber and result in entrainment of nitrogen gas in the surge chamber outlet. This would adversely affect the function of the NW system to provide a water seal to the containment isolation valves.

The team used Inspection Manual Chapter (IMC) 0609, Att. 4, “Initial Characterization of Findings,” issued October 7, 2016, for barrier integrity, 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 did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, and heat removal components, and the finding did not involve an actual reduction in function of hydrogen igniters in the reactor containment.

Since Revision 7 of calculation CNC-1223.19-00-0004 was approved on May 9, 2014, the team determined the finding was indicative of present licensee performance and was associated with the cross-cutting aspect of Evaluation, in the area of Problem Identification and Resolution, per IMC 0310, “Aspects Within Cross Cutting Areas,” issued December 4, 2014. Specifically, the licensee did not thoroughly evaluate the proposed corrective actions in ARs 01912139 and 01912453 after the revision to the calculation was completed to ensure that the correct high pressure NW surge chamber design requirement would have been translated into procedures [P.2].

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” required 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. Contrary to the above, since at least May 9, 2014, the licensee failed to translate the NW surge chamber high-pressure design limit of 85 psig from calculation CNC-1223.19-00-0004, “NW system setpoint calculation,” Rev. 7, into procedure OP/1/A/6200/019, “Containment Valve Injection Water System,” Rev. 36. The failure to translate the NW surge chamber pressure limit into procedures resulted in exceeding the NW surge chamber high pressure limit and could adversely affect the function of the NW system to provide a water seal to the containment isolation valves. The licensee entered this issue into their CAP as AR 02096392, reviewed for current and past operability, and issued an operations guide to limit the NW surge chamber pressures to 80 psig. This violation is being treated as an NCV consistent with Section 2.3.2.a of the Enforcement Policy (NCV 05000413, 414/2017007-01, Failure to translate design requirements into operating procedures for NW system)

4OA6 Meetings, Including Exit

On February 3, 2017, the inspectors presented inspection results to Mr. Tom Simril and other members of the licensee’s staff. Additional inspection results were discussed on March 16, 2017. The inspectors verified no proprietary information was retained or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
T. Simril, Site Vice President
C. Curry, Plant Manager
C. Bigham, Organization Effectiveness Director
C. Fletcher, Regulatory Affairs Manager
L. Keller, Engineering General Manager
S. Myers, Design Engineering Director
D. Kulla, Nuclear Design Engineering Manager
S. Andrews, Senior Engineer
D. Yang, Engineer
M. Dickson, Lead Engineer

NRC personnel
F. Ehrhardt, Chief, Projects Branch 1, Division of Reactor Projects
J. Austin, Senior Resident Inspector, Division of Reactor Projects
C. Scott, Resident Inspector, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000413, 414/2017007-01	NCV	Failure to Translate Design Requirements into Operating Procedures for NW System

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations

A/R 01342540, Atmospheric Steam Dump Isolation for Greater than 90 days, dated 1/30/2014
A/R 01346137, Revise the UFSAR Section 5.2.2 and TS Bases 3.4.12, dated 1/22/15
A/R 01962944, Revision to Methodology Report DPC-NE-2009-P-A to implement Revision 1-A to WCAP-8963-P-A Addendum 1-A , dated 11/3/2015
EC 105813, Upgrade Unit 1 Main Feedwater Control & Bypass Valve Positioners, Rev.1
EC 105814, Upgrade Unit 2 Main Feedwater Control and Bypass Valve Positioners, Rev.1
EC 105974, Installation of SSPS Boards Containing Complex Programmable Logic Device (CPLD) Technology, Rev. 0

10 CFR 50.59 Screenings

A/R 2087532, PT/2/A/4600/002A – 215862 Mode 1 Periodic Surveillance Items Rev. 196, dated 1/4/2017
EC 105829, Replace Valves 1NV312A and 1NV314B with 09J-644, Rev. 6
EC 106445, Install 5 Vents on Unit 2 KC System Piping as Recommended by SER 02-05 Evaluation U2-19, Rev. 0
EC 108309, Replace Valves and Operator 2NI009A and 2NI010B due to Margin Issues, Rev. 3
EC 109611, Piping Reroute Downstream of 1NM006A to Minimize Crud Collection, Rev. 1
EC 110541, Revise Unit 1 Accelerated Sequence Voltage Permissive Circuit, Rev. 3
EC 110934, (FKS) Install Backup Spent Fuel Pool Level Indication for Unit 1 Required for NRC Order EA-12-051, Rev. 3
EC 110955, Add Fuses In Series With Breakers in 2EDE and 2EDF Distribution Centers to Improve Coordination, Rev. 0
EC 110958, Resolve Breaker Coordination for Eight Unit 2 600vac Essential MCC Incoming Breakers Which Have an Unacceptable PRA Risk for Fire and Require Resolution for Transition to NFPA 805, Rev. 0
EC 110965, Add Fuses In Series With Breakers in 1EDE and 1EDF Distribution Centers to Improve Coordination, Rev. 0
EC 111298, FKS Provide MCC Cable Plug in Connections for Phase 1 Loads Unit 1, Rev. 4
EC 111337, Replace 2ERPA Panelboard Subpanel, Dead Front Panel and External Door, Rev. 3
EC 111923, 1D NCP Motor Replacement: Configuration Management, Rev. 13
EC 112410, Reroute 3CA Cables in Train A SWGR Room for Fire CDF Improvement, Rev.0
EC 112411, Reroute 3CA Cables in Train A SWGR Room for Fire CDF Improvement, Rev. 3
EC 115007, Add Hanger to Provide Support for 2NS-18A Valve Operator, Rev. 1
EC 115370, 2EIA: TM For U2 PZR Level Indications Temp EC Expected Removal 2EOC21, Rev. 0
EC 406259, SSF Indication For NC Loop B & C Wide Range Cold Leg Temperature, Rev.0
EC 97797, Relocate and Replace Valves 2NV32B and 2NV39A, Rev. 3

10 CFR 50.59 Applicability Determinations

EC 111080, Install Red Range on C/R Gages 1(2)NWP5040 and 5050 Showing High Pressure, Rev. 0

Calculations

CNC -1206.02-83-2010, Unit 1 NVF Stress Analysis, Revision 12
CNC -1206.12-29-1006, Unit 1 NVF Support/Restraint Design, Revision 3
CNC-1206.02-83-0004, NM System Stress Problem NM-03, Revision 20
CNC-1206.12-24-1013, Support Restraint Calculation for Math Model NIB, Revision 2

CNC-1223.03-00-0040, Low Temperature Overpressure Protection (LTOP) Failure Modes and Effects Analysis (FMEA), Rev. 1
CNC-1223.19-00-0004, NW System Setpoint Calculation, Rev. 7
CNC-1381.05-00-0135, U1/U2 125VDC Vital I&C Power System (EPL) Short Circuit Analysis, Rev. 7
CNC-1381.05-00-0163, 125 VDC Vital Instrumentation & Control Battery Load Study Summary, Rev. 1
CNC-1381.05-00-0251, Catawba Nuclear Station, UNITS 1 and 2 NFPA 805 Circuit Breaker And Fuse Coordination Study, Rev. 17
CNC-1381.05-00-0251, Units 1 and 2 NFPA 805 Circuit Breaker and Fuse Coordination Study. Rev. 017 Date: 12/5/2016
CNC-2206.02-82-2001, Pipe Stress Calculation for Math Model NIB, Revision 8
CNM 1205.00-2191.001, Design/Seismic/Weak Link Report for Item 09J-644, Revision 2

Corrective Action Documents

01449472	01912139
01513020	01912453
01897325	02087532
01898043	02087705
01898180	

Drawings
CN-1702-05.02, One Line Diagram, Essential & Blackout Auxiliary Power Systems, 4.16KV/600V Systems EPC, EPE & ETC, Rev. 9
CN-1878-01, Electrical Equipment Location, Auxiliary Building, Below El. 611'-0", Rev. 57
CN-1904-06, Electrical Equipment Layout, Misc. Electrical Equipment Seismic mounting Details, Rev. 0
CN-20705-01, One Line Diagram, 120VAC Vital Instrumentation and Control Power System (EPG), Rev. 17
CN-20705-01.02-01, One Line Diagram, 120VAC Vital Instrumentation and Control Power System (EPG), Rev. 16E
CNEE-0114-00.08, Elementary Diagram Diesel Generator No. 1A Load Sequencer (Part 8) Committed & Accelerated Sequence Circuits, Rev. 11
CNEE-0114-00.10, Elementary Diagram Diesel Generator No. 1A Load Sequencer (Part 10) Loading Relays, Rev. 2
CNEE-0114-00.11, Elementary Diagram Diesel Generator No. 1A Load Sequencer (Part 11) Loading Relays, Rev. 7

Licensing Bases Documents
DUKE-QAPD-001 A, Duke Energy Corporation Topical Report, Quality Assurance Program Description, Operating Fleet, Amendment 41
SER and Supplements
TS, Current
UFSAR, Current

Miscellaneous Documents
CD200007, Duke Power Company Engineering Change, NM to correct NC pump motor design documentation, Unit 2, approved date 1/31/2005
CN -1904-06, EC0000111298, Electrical Equipment Layout Misc. Electrical, Equipment Seismic Mounting Details, Rev. 000

CN -2705-01.02, One Line Diagram, 120 VAC Vital Instrumentation and Control Power System EPG, Rev. 6
CN -2705-01.02-01, EC0000111337, VTO One Line Diagram 120 VAC Vital Instrumentation And Control Power System EPG, Rev. 017
CN-1381.12, Memo to File, DC-1.02 and Backfeed Cables for QA Conditions 1 MCC's, dated November 2, 2014
CNBM-1752-01.02-01, EC0000111298, Bill of Materials Fukushima Extended Loss of AC Power Backfeed Receptacles, Rev. 000,
CNLT-1752-01.01, EC0000108310, EC0000110962, EC0000111298, Electrical One Line and Bill of Material List for 600v Essential Motor Control Center 1EMXA, Rev. 047
CNLT-1752-01.02, EC0000403822, EC0000404862, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXB, Rev. 038
CNLT-1752-01.03, EC0000110962, EC0000111298, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXC, Rev. 026
CNLT-1752-01.04, EC0000110962, EC0000111298, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXD, Rev. 024
CNLT-1752-01.07, EC0000407060, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXG, Rev. 032
CNLT-1752-01.08, EC0000403792, Electrical One Line and Bill of Material List for 600v Essential Motor Control Center 1EMXI, Rev. 030
CNLT-1752-01.09, EC0000404771, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXJ, Rev. 042
CNLT-1752-01.10, EC0000404082, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXK, Rev. 030
CNLT-1752-01.11, EC0000110962, EC0000111298, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXL, Rev. 033
CNLT-1752-01.15, EC0000111298, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXS, Rev. 020
CNLT-1752-01.15, EC0000111298, Electrical One Line and Bill of Material List for 600V Essential Motor Control Center 1EMXS, Rev. 020
CNM 1201.01-0157.001, Reactor Coolant Pump I/B, Rev. 057
CNM 1201.01-0157.001, Vendor/Duke manual certification form, Reactor Coolant Pump I/B, Rev. D54
CNM 1393.01-0012.001, Environment and Seismic Qualification of 120 VAC VITAL Panelboard Parts, Qualification Report QR 15-03, 128-9248689-001, Rev. 000
CNM 2314.01-0159.001, 120 VAC I&C Power Panel Board Main Pan O/L & Connection, Diagram, EC0000111337, 30 Circuit Panel Overall Layout "QA Condition 1", Rev. 011
CNM 2314.01-0159.002, 120 VAC I&C Power Panel Board # 2ERPA B/M, EC0000111337, 30 circuit panel schedules "QA Condition 1", Rev. 011
CNS 1314.01-00-0002, Replacement of 120 VAC Vital and Non-Vital Panelboard Subpanels and Doors, Rev. 002
CNS 1318.11-00-0003, Reactor Coolant Pump Motor Stator Replacement, Rev. 1
CNS-106.01-EPG-0001, EC0000111337, Design Basis Specification 120VAC VITAL Instrumentation and Control Power System (EPG), Rev. 009
CNS-1318.11-00-0004, EC0000406738, Refurbishment/Repair Specification for Westinghouse Reactor Coolant Pump Motors, Rev. 001
CNS-1560.SS-00-0001, Design Basis Specification for the Standby Shutdown Facility (SSF), Rev. 1
CNS-1569.NW-00-0001, Containment Valve Injection Water System (NW) Design Basis Specification, Rev. 18
CNS-SLC-16.7-9, Standby Shutdown System (SSS)

DPS 1205.00-00-0002; Nuclear Safety Related Carbon and Stainless Steel Forged and Cast Electric Motor Operated Gate Valves, Revision 3
Meltric Corporation, DR series plugs and receptacles manufacturer catalog sheets, 2017
Operability Determination 2097222; NRC 2017 50.59 Seismic Qualification of NV Valves; dated February 2, 2017
WCAP-17867-P-A, Westinghouse SSPS Board replacement Licensing Summary Report, Rev. 1

Procedures
AD-EG-ALL-1103. Procurement Engineering Products, Rev. 4
AD-EG-ALL-1110, Design Review Requirements, Rev. 3
AD-EG-ALL-1117, Design Analyses and Calculations, Rev. 4
AD-EG-ALL-1132, Preparation and Control of Design Change Engineering Changes, Rev. 6
AD-EG-ALL-1133, Preparation and Control of Equivalent Change Engineering Changes, Rev. 2
AD-EG-ALL-1134, Preparation and Control of Evaluation Engineering Changes, Rev. 0
AD-EG-ALL-1135, Preparation and Control of Configuration Management Update Engineering Changes, Rev. 0
AD-EG-ALL-1136, Preparation and Control of Commercial Controls Engineering Changes, Rev. 0
AD-LS-ALL-0005, UFSAR Updates, Rev. 4
AD-LS-ALL-0007, Applicability Determination Process, Rev. 2
AD-LS-ALL-0008, 10 CFR 50.59 Review Process, Rev. 0
AD-LS-ALL-0010, Commitment Management, Rev. 2
CNLT-1780-03.03, Environmental Qualification Criteria Manual (EQCM), Rev. 34
FG/0/A/CFLX/FSG-20, Flex Electrical Distribution, Rev. 3
IP/0/A/4974/039, Termination and Splicing of Medium Voltage Cable (Rated 5kV through 15kV), Rev. 018
OP/0/B/6100/013, Standby Shutdown Facility Operations, Rev. 55
OP/1/A/6100/001, Controlling Procedure for Unit Startup, Rev. 240
OP/1/A/6100/002, Controlling Procedure for Unit Shutdown, Rev. 186
OP/1/A/6200/001, Chemical and Volume Control System, Rev. 153
OP/1/A/6200/004, Residual Heat Removal System, Rev. 143
OP/1/A/6200/019, Containment Valve Injection Water System, Rev. 36
OP/1/A/6350/001, Normal Power Checklist, Rev. 082
OP/1/A/6350/005, Alternate AC Power Sources, Rev. 078
PT/1/A/4600/003A, Monthly Surveillance Items, Enclosure 13.6, Rev. 135

Condition Reports (ARs) generated as a result of the inspection
2095923, Revise UFSAR Sections 15.3.1.3 and 15.3.2.3
2096366, Revise CNS-1560.SS-00-0001 for SSF Equipment Functionality
2096392, Update and Complete EC 111080 and PRRs
2096781, Revise NW DBD, CNS-1569.NW-00-0001
2096798, Revise AR 2087532 screen to treat pressurizer level as a design function for Pressurizer Level Channel 3
2096835, Error found in CNM1205.00-2191 Seismic Report
2097222, Seismic Qualification of Valves 1NV312A, 1NV314B, 2NV312A, and 2NV314B