UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 14, 2017

Mr. Peter P. Sena, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS RE: ACCIDENT MONITORING INSTRUMENTATION (CAC NOS. MF8859 AND MF8860; EPID L-2016-LLA-0041)

Dear Mr. Sena:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment Nos. 320 and 301 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the technical specifications in response to your application dated November 17, 2016, as supplemented by letters dated August 7, 2017, and October 18, 2017.

The amendments revise technical specification requirements regarding accident monitoring instrumentation. Specifically, the amendments modify the list of instruments required to be operable based on implementation of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." In addition, allowed outage times and required actions for inoperable accident monitoring instrumentation channels have been revised to be consistent with NUREG-1431, Revision 4.0, "Standard Technical Specifications – Westinghouse Plants."

A copy of our related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Carleen J. Parker, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 320 to Renewed DPR-70
2. Amendment No. 301 to Renewed DPR-75
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 320
Renewed License No. DPR-70

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees), dated November 17, 2016, as supplemented by letters dated August 7, 2017, and October 18, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 320, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: November 14, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 320

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of Renewed Facility Operating License No. DPR-70 with the attached revised page as indicated. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55
3/4 3-56	3/4 3-56
3/4 3-56a	3/4 3-56a
3/4 3-57	3/4 3-57
3/4 3-57a	3/4 3-57a
6-24b	6-24b

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 320, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

(3) Deleted Per Amendment 22, 11-20-79

(4) Less than Four Loop Operation

PSEG Nuclear LLC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this renewed license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this renewed license.

(5) PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

---NOTE---
Separate Condition entry is allowed for each Function.
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a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-11.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. Deleted			
12. Deleted			

TABLE 3.3-11 (CONTINUED)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. Deleted			
14. Deleted			
15. Deleted			
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9
20. Containment High Range Accident Radiation Monitor	2	2	10
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	1/MS Line	1/MS Line	10
22. Wide Range Neutron Flux Monitors	2	1	1, 2
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level	2	1	1, 2
24. Containment Isolation Valve Position Indication	2 per penetration flow path [(a)(b)]	1/valve[(c)]	1, 2

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) Action 2 not required for penetration flow paths with only one installed control room indication channel.

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 1 With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 2 With the number of OPERABLE accident monitoring channels less than the MINIMUM Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 3 deleted

ACTION 4 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel; otherwise, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 5 deleted

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 6 With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 7 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed until the next CHANNEL CALIBRATION (which shall be performed upon the next entry into MODE 5, COLD SHUTDOWN).

ACTION 8 With one RVLIS channel inoperable, restore the RVLIS channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 9 With both RVLIS channels inoperable, restore one channel to OPERABLE status within 7 days or submit a special report in accordance with Specification 6.9.4.

ACTION 10 With the number of OPERABLE Channels less than required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the actions taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK[(1)]	CHANNEL CALIBRATION[(1)]	CHANNEL FUNCTIONAL TEST[(1)]
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)			N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)			N.A.
3. Reactor Coolant Pressure (Wide Range)			N.A.
4. Pressurizer Water Level			N.A.
5. Steam Line Pressure			N.A.
6. Steam Generator Water Level (Narrow Range)			N.A.
7. Steam Generator Water Level (Wide Range)			N.A.
8. Refueling Water Storage Tank Water Level			N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#		N.A.
11. Deleted			

Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK[(1)]	CHANNEL CALIBRATION[(1)]	CHANNEL FUNCTIONAL TEST[(1)]
12. Deleted			
13. Deleted			
14. Deleted			
15. Deleted			
16. Containment Pressure - Wide Range			N.A.
17. Containment Water Level - Wide Range			N.A.
18. Core Exit Thermocouples			N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)			N.A.
20. Containment High Range Accident Radiation Monitor			
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor			
22. Wide Range Neutron Flux Monitors			N.A.
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level			N.A.
24. Containment Isolation Valve Position Indication			N.A.

Table Notation

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined,

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 1, 4, 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 301
Renewed License No. DPR-75

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees), dated November 17, 2016, as supplemented by letters dated August 7, 2017, and October 18, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 301, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating
License and Technical Specifications

Date of Issuance: November 14, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 301

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of Renewed Facility Operating License No. DPR-75 with the attached revised page as indicated. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 3-50	3/4 3-50
3/4 3-51	3/4 3-51
3/4 3-51a	3/4 3-51a
3/4 3-51b	3/4 3-51b
3/4 3-51c	3/4 3-51c
3/4 3-52	3/4 3-52
3/4 3-52a	3/4 3-52a
6-24b	6-24b

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at steady state reactor core power levels not in excess of 3459 megawatts (thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 301, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

---NOTE---

Separate Condition entry is allowed for each Function.

----

a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-11.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. Deleted			
12. Deleted			

TABLE 3.3-11 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. Deleted			
14. Deleted			
15. Deleted			
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9
20. Containment High Range Accident Radiation Monitor	2	2	10
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	1/MS Line	1/MS Line	10
22. Wide Range Neutron Flux Monitors	2	1	1, 2
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level	2	1	1, 2
24. Containment Isolation Valve Position Indication	2 per penetration flow path [(a)(b)]	1/valve[(c)]	1, 2

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated valve, closed manual valve, blind flange, or check valve with flow through the valve secured.
(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
(c) Action 2 not required for penetration flow paths with only one installed control room indication channel.

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 1 With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 2 With the number of OPERABLE accident monitoring channels less than the Minimum Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 3 deleted

ACTION 4 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operations may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate Channel; otherwise, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 5 deleted

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 6 With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 7 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed until the next CHANNEL CALIBRATION (which shall be performed upon the next entry into MODE 5, COLD SHUTDOWN).

ACTION 8 With one RVLIS channel inoperable, restore the RVLIS channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 9 With both RVLIS channels inoperable, restore one channel to OPERABLE status within 7 days or submit a special report in accordance with Specification 6.9.4.

ACTION 10 With the number of OPERABLE Channels less than required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the actions taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)			N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)			N.A.
3. Reactor Coolant Pressure (Wide Range)			N.A.
4. Pressurizer Water Level			N.A.
5. Steam Line Pressure			N.A.
6. Steam Generator Water Level (Narrow Range)			N.A.
7. Steam Generator Water Level (Wide Range)			N.A.
8. Refueling Water Storage Tank Water Level			N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#		N.A.
11. Deleted			

Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
12. Deleted			
13. Deleted			
14. Deleted			
15. Deleted			
16. Containment Pressure - Wide Range			N.A.
17. Containment Water Level - Wide Range			N.A.
18. Core Exit Thermocouples			N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)			N.A.
20. Containment High Range Accident Radiation monitor			
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor			
22. Wide Range Neutron Flux Monitors			N.A.
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level			N.A.
24. Containment Isolation Valve Position Indication			N.A.

Table Notation

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 1, 4, 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 320 AND 301 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated November 17, 2016, as supplemented by letters dated August 7, 2017, and October 18, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16323A279, ML17219A160, and ML17291A766, respectively), PSEG Nuclear LLC (PSEG, the licensee) submitted a license amendment request (LAR) for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The amendments would revise technical specification (TS) requirements regarding accident monitoring instrumentation. Specifically, the amendments would modify the list of instruments required to be operable based on implementation of Regulatory Guide (RG) 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 (ADAMS Accession No. ML060750525). In addition, allowed outage times (AOTs) and required actions for inoperable accident monitoring instrumentation channels would be revised to be consistent with NUREG-1431, Revision 4.0, "Standard Technical Specifications – Westinghouse Plants" (ADAMS Accession Nos. ML12100A222 and ML12100A228).

The supplemental letters dated August 7, 2017, and October 18, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission's (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 17, 2017 (82 FR 4931).

2.0 REGULATORY EVALUATION

2.1 Background

The TS requirements for the Salem accident monitoring instrumentation are specified in TS 3/4.3.3.7, "Accident Monitoring Instrumentation." In accordance with Limiting Condition for Operation (LCO) 3.3.3.7, the instrumentation channels shown in TS Table 3.3-11, "Accident Monitoring Instrumentation," shall be operable in Modes 1, 2, and 3. Table 3.3-11 also specifies AOTs and required actions for inoperable accident monitoring instrumentation channels.

In accordance with Surveillance Requirement (SR) 4.3.3.7, each accident monitoring instrumentation channel shall be determined operable by performance of channel checks, channel calibrations, and channel functional tests, at the frequencies specified in the surveillance frequency control program, unless otherwise noted in TS Table 4.3-11, "Surveillance Requirements for Accident Monitoring Instrumentation." The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

On September 4, 2015, the NRC issued Amendment No. 310 for Salem, Unit No. 1 (ADAMS Accession No. ML15245A636). This amendment approved the removal of the pressurizer power-operated relief valve (PORV) position indication from TS Tables 3.3-11 and 4.3-11. As discussed in the licensee's application dated November 17, 2016, since the LAR associated with Amendment No. 310 was requested and approved on an emergency basis, the LAR was only for the specific PORV line item for Salem, Unit No. 1. As such, PSEG has submitted the current LAR to align the remainder of the Salem, Unit Nos. 1 and 2, accident monitoring instrumentation channels specified in TS Tables 3.3-11 and 4.3-11 with RG 1.97, Revision 2, and NUREG-1431, Revision 4.0.

2.2 Regulatory Requirements and Guidance

The NRC staff identified the following regulatory requirements and guidance as being applicable to the LAR:

2.2.1 Technical Specification Requirements

The NRC's regulatory requirements related to the content of the TSs are specified in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications." Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

As stated in 10 CFR 50.36(c)(2)(i), LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met. The LCO action requirements establish those remedial actions that must be taken when the requirements of an LCO are not met.

As stated in 10 CFR 50.36(c)(3), SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement), which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953, July 19, 1995). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As discussed in the *Federal Register* notice for the final rule dated July 19, 1995 (60 FR 36955):

> LCOs that do not meet any of the criteria, and their associated actions and surveillance requirements, may be proposed for relocation from the technical specifications to licensee-controlled documents, such as the FSAR [Final Safety Analysis Report]. The criteria may be applied to either standard or custom technical specifications.

As discussed in the Final Policy Statement (58 FR 39138):

> When licensees submit amendment requests based on this Policy Statement, they should identify the location of and controls for the technical and administrative requirements of the relocated requirements. The NRC staff will carefully review these submittals to ensure the accountability and the acceptability of controls for each relocated requirement. Many of the requirements will be relocated to the FSAR and will be enforceable through 10 CFR 50.59. Other requirements will be relocated to more appropriate documents (e.g., Security Plan, QA Plan) and controlled by the applicable regulatory requirements. The adequacy of controls for relocated requirements which do not fit in the above categories will be reviewed and approved by the NRC staff on a case-by-case basis to determine, among other things, whether an enforceable control method will need to be established.

As discussed in the licensee's letter dated August 7, 2017, the amendments would relocate certain TS requirements (as discussed in Section 2.3 of this safety evaluation (SE)) to the Salem Updated Final Safety Analysis Report (UFSAR). Accordingly, further changes to the relocated requirements would be controlled via the requirements in 10 CFR 50.59, "Changes, tests and experiments."

2.2.2 Regulatory Guide 1.97

RG 1.97, Revision 2, describes a method acceptable to the NRC staff for complying with the NRC's regulations to provide instrumentation to monitor plant variables and system during and following an accident in a light-water-cooled nuclear power plant. On page 3 of Attachment 1 to the licensee's application dated November 17, 2016, PSEG noted that:

> Salem was designed prior to the issuance of RG 1.97. The UFSAR was subsequently updated to demonstrate compliance with the intent of RG 1.97, Revision 2. In various site-specific evaluations of PAM [post-accident monitoring] instruments, Revision 3 was used by Salem as more clearly presenting guidance for PWR [pressurized-water reactor] variables while at the same time being essentially equivalent to Revision 2 guidance. The current RG 1.97 (Revision 4) states that it is primarily intended for new reactors and that previous versions of the RG remain in effect for licensees of current operating reactors. Revision 2 remains the Salem licensing basis for RG 1.97 compliance.

Accordingly, the NRC staff reviewed the LAR in accordance with the guidance in RG 1.97, Revision 2. RG 1.97 lists five types (Types A-E) of variables to help designers select the appropriate accident monitoring instrumentation. The types are as follows:

Type A: Those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design-basis accident (DBA) events.

Type B: Those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control).

Type C: Those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Type D: Those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Type E: Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

Tables 1 and 2 in RG 1.97 list the specific variables for each of the five types listed above for boiling-water reactors and PWRs, respectively. Regulatory Positions 1.3 and 1.4 in RG 1.97 provide design and qualification criteria for the instrumentation used to measure the various variables in Tables 1 and 2. The design and qualification criteria are separated into three categories that provide a graded approach. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status. Category 3 is intended to provide requirements that will ensure high-quality, off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. It is also used where the state of the art will not support requirements for higher qualified instrumentation.

2.2.3 <u>NUREG-1431</u>

NUREG-1431, Revision 4 was used by the NRC staff as guidance regarding the PAM instrumentation that should be included in the TSs. As discussed in the Bases for Standard

Technical Specification (STS) 3.3.3, "Post Accident Monitoring (PAM) instrumentation," the instrument channels required to be operable by the PAM LCO include two classes of parameters identified during plant-specific implementation of RG 1.97. Specifically, the two classes include Type A variables and Category 1 variables. Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs. Category 1 variables are the key variables deemed risk-significant because they are needed to:

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

2.2.4 NUREG-0737

Following the accident at Three Mile Island Nuclear Station (TMI), Unit 2, on March 28, 1979, the NRC staff developed a number of proposed requirements to be implemented on operating reactors and on plants under construction. The requirements included:

- NUREG-0737, "Clarification of TMI Action Plan Requirements" published November 1980 (ADAMS Accession No. ML051400209).
- NUREG-0737, Supplement No. 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," published January 1983 (ADAMS Accession No. ML102560009).

The above documents provide requirements regarding PAM instrumentation, including clarification of RG 1.97, Revision 2, guidance.

2.2.5 Salem UFSAR

Salem UFSAR, Revision 29 (ADAMS Package No. ML17046A230), Section 7.5, "Safety-Related Display Instrumentation," discusses Salem compliance with RG 1.97.

2.2.6 Other References

In addition to the regulatory requirements and guidance listed above, the NRC staff used the following Salem plant-specific precedent in its review of this LAR:

- Letter from NRC (R. Ennis) to PSEG (T. Joyce) dated March 21, 2011, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 - Issuance of Amendments Re: Relocation of Specific Surveillance Frequencies to a Licensee-Controlled Program Based on Technical Specification Task Force (TSTF) Change TSTF-425," Amendment Nos. 299 and 282 (ADAMS Accession No. ML110410691).
- Letter from NRC (C. Parker) to PSEG (R. Braun) dated September 4, 2015, "Salem Nuclear Generating Station, Unit No. 1 - Issuance of Emergency Amendment Regarding Removal of

Pressurizer Power Operated Relief Valve Position Indication Instrumentation from Technical Specifications," Amendment No. 310 (ADAMS Accession No. ML15245A636).

2.3 Proposed TS Changes

2.3.1 Proposed Changes to TS Tables 3.3-11 and 4.3-11

PSEG described the methodology used to assess the proposed changes to TS Tables 3.3-11 and 4.3-11 in Section 4.1 of Attachment 1 to the licensee's application dated November 17, 2016. The licensee stated, in part, that:

> The intent is to align the instruments in the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11 to be consistent with the scope of the NUREG-1431 reviewer's note which states that the TS should include all RG 1.97 Type A instruments and all RG 1.97, Category 1, non-Type A instruments in accordance with the Unit's RG 1.97 Safety Evaluation Report. The instruments proposed for removal from the TS tables are not being removed from the plant and will continue to satisfy their RG 1.97 requirements.

Accordingly, the licensee has proposed to remove certain instruments from TS Tables 3.3-11 and 4.3-11 because they are not RG 1.97 Type A or Category 1, in accordance with the Salem, Unit Nos. 1 and 2, licensing basis pertaining to RG 1.97. The requirements proposed for removal from the TSs would be relocated to the UFSAR.

In addition, the licensee has proposed to add instruments to TS Tables 3.3-11 and 4.3-11 because, in accordance with the Salem, Unit Nos. 1 and 2, licensing basis, they are RG 1.97 Type A or Category 1 and are not currently shown in the tables. The proposed changes are summarized in Table 1 below.

Table 1

Instrument	Unit Nos.	Action
Reactor Coolant System Subcooling Margin Monitor	1 and 2	Remove
PORV Position Indicator	2 only (Note 1)	Remove
PORV Block Valve Position Indicator	1 and 2	Remove
Pressurizer Safety Valve Position Indicator	1 and 2	Remove
Containment Pressure - Narrow Range	1 and 2	Remove
Wide Range Neutron Flux Monitors	1 and 2	Add
Auxiliary Feedwater Storage Tank (Condensate) Storage Tank) Water Level	1 and 2	Add
Containment Isolation Valve Position Indications	1 and 2	Add

Notes

1) Removal of the PORV Position Indicator is only being requested for Salem, Unit No. 2, because removal of this indication from the Salem, Unit No. 1, TSs was approved by the NRC staff in Salem, Unit No. 1, Amendment No. 310.

2) For the instruments proposed to be removed from TS Tables 3.3-11 and 4.3-11, the associated footnotes for those instruments would also be removed from the tables.

2.3.2 Proposed Changes to AOTs and Actions for Inoperable Channels

To be consistent with NUREG-1431, the LAR also proposed to revise the AOTs and Actions for inoperable channels in the TABLE NOTATION of Table 3.3-11. These changes are shown in Table 2 below. Bold text in Table 2 is used to help identify the wording changes.

Table 2

Action	Current Wording	Proposed Wording
Action 1	With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within **7 days**, or **be in HOT SHUTDOWN within the next 12 hours.**	With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within **30 days**, or **submit a special report in accordance with Specification 6.9.4**.
Action 2	With the number of OPERABLE accident monitoring channels less than the MINIMUM Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within **48 hours** or be in HOT SHUTDOWN within the **next 12 hours**.	With the number of OPERABLE accident monitoring channels less than the MINIMUM Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within **7 days** or be in **HOT STANDBY within the next 6 hours and in** HOT SHUTDOWN within the **following 6** hours.
Action 4	With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel**.**	With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel**; otherwise, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.**
Action 6	With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the **next 12** hours.	With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be **in HOT STANDBY within the next 6 hours and in** HOT SHUTDOWN within the **following 6** hours.

2.3.3 Other Proposed TS Changes

In addition to the proposed changes to TS Tables 3.3-11 and 4.3-11 described in Sections 2.3.1 and 2.3.2 of this SE, the licensee has proposed the following TS changes:

- TS 3.3.3.7 would be revised to add a note which states, "Separate Condition entry is allowed for each Function."

- TS 6.9.4, "Special Reports," currently reads, in part, that, "When a report is required by ACTION 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days." This sentence would be revised to read: "When a report is required by ACTION 1, 4, 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days."

3.0 TECHNICAL EVALUATION

3.1 NRC Staff Approach

The NRC staff evaluated the LAR using the regulatory requirements and guidance shown in Section 2.2 of this SE. The staff used the following approach to determine the acceptability of the LAR:

- For instruments proposed to be removed from TS Tables 3.3-11 and 4.3-11, confirm that the associated function: (1) does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and (2) is not considered a Type A or Category 1 variable, in accordance with the Salem licensing basis related to RG 1.97.

- For instruments proposed to be added to TS Tables 3.3-11 and 4.3-11, confirm that the associated function is considered a Type A or Category 1 variable, in accordance with the Salem licensing basis related to RG 1.97.

- For the proposed changes to AOTs and actions for inoperable channels, confirm whether the changes are consistent with NUREG-1431.

3.2 Proposed Changes to TS Tables 3.3-11 and 4.3-11

3.2.1 Reactor Coolant System Subcooling Margin Monitor

As discussed in the LAR, the reactor coolant system (RCS) subcooling margin monitor (SMM) indication provides information to the operators related to satisfying one of the safety injection termination criteria following a steamline break of steam generator tube rupture accident. The inputs to the RCS SMM are the core exit thermocouples for RCS temperature and the wide range RCS pressure indication for RCS pressure.

As discussed Section 2.3.1 of this SE, the licensee proposes to remove the RCS SMM instrument function from TS Tables 3.3-11 and 4.3-11 (item 11 in both tables) for Salem, Unit Nos. 1 and 2. The licensee provided the following justification with respect to the criteria in 10 CFR 50.36 (c)(2)(ii):

> The RCS SMM does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is

intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. The instrumentation that satisfies Criterion 1 is contained in Salem Unit 1 TS 3.4.6.1 and Unit 2 TS 3.4.7.1 "Reactor Coolant System Leakage, Leakage Detection Systems."

RCS sub-cooling instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

The RCS SMM is not part of a primary success path as indicated in Criterion 3.

The loss of the RCS SMM instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The NRC staff has reviewed the licensee's submittal and agrees with the licensee's conclusion that the RCS SMM instrument function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) requiring inclusion of this item as a TS LCO.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97, Table 2, "PWR Variables," indicates that the RCS SMM function ("Degrees of Subcooling") is a Type B, Category 2 variable provided for verification and analysis of plant conditions.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," does not include the RCS SMM instrument function.

With respect to the Salem licensing basis, UFSAR Table 7.5-3, "Index Type 'A' Variables," currently identifies the "Degrees of Subcooling" function as an RG 1.97, Type A variable. The LAR indicates that the licensee has evaluated this classification and has determined that the RCS SMM should be reclassified as a Type B, Category 2 variable, consistent with RG 1.97. Accordingly, the LAR states that the UFSAR and associated design documents will be revised under the provisions of 10 CFR 50.59 to reflect that the RCS SMM is not a Type A variable.

Based on its review and the preceding evaluation, the NRC staff concludes that the proposed deletion of the RCS SMM instrument function from TS Tables 3.3-11 and 4.3-11 (item 11 in both tables) and the associated footnote in Table 4.3-11 for Salem, Unit Nos. 1 and 2, is acceptable because the instrumentation: (1) does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and (2) is not considered a Type A or Category 1 variable per the proposed change to the Salem licensing basis. Consistent with the licensee's letter dated August 7, 2017, the requirements being deleted from the TSs will be relocated to the Salem UFSAR. Further changes to the relocated requirements will be controlled via the requirements in 10 CFR 50.59.

3.2.2 <u>PORV Position Indicator</u>

As discussed in the LAR, the RCS is protected against over-pressurization by control and protective circuits such as the pressurizer pressure high reactor trip and by the PORVs connected to the top of the pressurizer. The PORVs provide the means for pressurizer venting. Each PORV has two limit switches that provide open and closed indication (i.e., lights) of the PORV position in the control room. The LAR also stated that the Salem design-basis accident

(DBA) analysis for the inadvertent opening of a PORV does not credit operator diagnosis and closure of the PORV or block valve. The DBA analysis assumes that automatic safety injection actuation will provide adequate protection.

As discussed in Section 2.3.1 of this SE, the licensee proposes to remove the PORV position indicator instrument function from TS Tables 3.3-11 and 4.3-11 (item 12 in both tables) for Salem, Unit No. 2.[1] The licensee provided the following justification with respect to the criteria in 10 CFR 50.36 (c)(2)(ii):

> The PORVs themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a DBA and therefore meet Criterion 3 of the NRC Final Policy Statement. For example, they are credited in Salem UFSAR Section 15.2.1 4, Spurious Operation of the Safety Injection System at Power. The operability of the PORVs is therefore required by TS 3.4.3, "Relief Valves." However, PORV position indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. PORV position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.
>
> While the function of the PORVs themselves is part of the primary success path in the UFSAR, PORV position indication is not part of the primary success path. UFSAR accident analysis assumes that the PORVs open as designed to reduce reactor pressure and no operator action based on PORV position indication is required. Therefore, PORV position indication is not part of the primary success path as indicated in Criterion 3.
>
> The loss of PORV position indication instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The NRC staff has reviewed the licensee's submittal and agrees with the licensee's conclusion that the PORV position indicator instrument function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) requiring inclusion of this item as a TS LCO.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97 Table 2, "PWR Variables," indicates that the PORV position indicator instrument function ("Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines") is a Type D, Category 2 variable provided for operational status and to monitor loss of coolant.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," does not include the PORV position indicator instrument function.

With respect to the Salem licensing basis, UFSAR Table 7.5-3, "Index Type "A" Variables," does not include the PORV position indicator instrument function as one of the Salem RG 1.97

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[1] As noted in Section 2.3.1 of this SE, removal of the PORV position indicator is only being requested for Salem, Unit No. 2, because removal of this indication from the Salem, Unit No. 1, TSs was approved by the NRC staff in Salem, Unit No. 1, Amendment No. 310.

Type A variables. UFSAR Table 7.5-4, "Summary of Instrumentation Compliance with Regulatory Guide 1.97," indicates that this function (i.e., "Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines") is considered Compliance Level 1. UFSAR Section 7.5.3.3 indicates that "Compliance Level 1" instrumentation meets the intent of RG 1.97. As such, the Salem licensing basis considers the PORV position indicator instrument function as a Type D, Category 2 variable.

Based on its review and the preceding evaluation, the NRC staff concludes that the proposed deletion of the PORV position indicator instrument function from TS Tables 3.3-11 and 4.3-11 (item 12 in both tables) and the associated footnote in Table 3.3-11, for Salem, Unit No. 2, is acceptable because the instrumentation: (1) does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and (2) is not considered a Type A or Category 1 variable per the Salem licensing basis. Consistent with the licensee's letter dated August 7, 2017, the requirements being deleted from the TSs will be relocated to the Salem UFSAR. Further changes to the relocated requirements will be controlled via the requirements in 10 CFR 50.59.

3.2.3 PORV Block Valve Position Indicator

As discussed in the LAR, the PORVs can be isolated by the PORV block valves which are connected in line with the PORVs. The PORV block valves provide closure redundancy to the PORVs, to isolate PORVs with excessive seat leakage or that stick open. Use of the PORV block valves in the event of a PORV malfunction can prevent a severe depressurization of the RCS with potential for uncovering of the reactor core. The PORV block valve position indication provides information to the control room operators on the position of the pressurizer PORV block valves. The LAR states further that this instrumentation is not needed for manual operator action necessary for safety systems to accomplish their safety function for the design-basis events.

As discussed above in SE Section 2.3.1, the licensee proposes to remove the PORV block valve position indicator instrument function from TS Tables 3.3-11 and 4.3-11 (item 13 in both tables) for Salem, Unit Nos. 1 and 2. The licensee provided the following justification with respect to the criteria in 10 CFR 50.36 (c)(2)(ii):

> The PORVs themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. In order to provide this function, the PORV block valves must be open or capable of being manually opened. The operability of the PORV block valves is therefore required by Unit 1 TS 3.4.3 and Unit 2 TS 3.4.5, "Relief Valves." However, PORV block valve position indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 is not to be interpreted to include instrumentation installed to identify the source of actual leakage, for example valve position indicators. The instrumentation that satisfies Criterion 1 is contained in Salem Unit 1 TS 3.4.6.1 and Unit 2 TS 3.4.7.1 "Reactor Coolant System Leakage, Leakage Detection Systems."

PORV block valve position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

While the function of the PORVs themselves is part of the primary success path in the UFSAR, the PORV block valve position indication is not part of the primary success path. UFSAR accident analysis assumes that the PORVs open as designed to reduce reactor pressure. In the event a PORV block valve is closed to isolate PORV valve seat leakage as allowed by the technical specifications, the emergency operating procedures direct the operators to manually open the PORV and associated block valve. This action does not rely on the PORV block valve position indication. Therefore, PORV block valve position indication is not part of the primary success path as indicated in Criterion 3.

The loss of PORV block valve position indication instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The NRC staff has reviewed the licensee's submittal and agrees with the licensee's conclusion that the PORV block valve position indicator instrument function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) requiring inclusion of this item as a TS LCO.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97, Table 2, "PWR Variables," indicates that the PORV block valve position indicator instrument function ("Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines") is a Type D, Category 2 variable provided for operational status and to monitor loss of coolant.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," does not include the PORV block valve position indicator instrument function.

With respect to the Salem licensing basis, UFSAR Table 7.5-3, "Index Type "A" Variables," does not include the PORV block valve position indicator instrument function as one of the Salem RG 1.97, Type A variables. UFSAR Table 7.5-4, "Summary of Instrumentation Compliance with Regulatory Guide 1.97," indicates that this function (i.e., "Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines") is considered Compliance Level 1. UFSAR Section 7.5.3.3 indicates that "Compliance Level 1" instrumentation meets the intent of RG 1.97. As such, the Salem licensing basis considers the PORV block valve position indicator instrument function as a Type D, Category 2 variable. Based on its review and the preceding evaluation, the NRC staff concludes that the proposed deletion of the PORV block valve position indicator instrument function from TS Tables 3.3-11 and 4.3-11 (item 13 in both tables) and the associated footnotes in both tables for Salem, Unit Nos. 1 and 2, is acceptable because the instrumentation: (1) does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and (2) is not considered a Type A or Category 1 variable per the Salem licensing basis. Consistent with the licensee's letter dated August 7, 2017, the requirements being deleted from the TSs will be relocated to the Salem UFSAR. Further changes to the relocated requirements will be controlled via the requirements in 10 CFR 50.59.

3.2.4 Pressurizer Safety Valve Position Indicator

As discussed in the LAR, the pressurizer safety valve position indication provides information to the control room operators on the position of the pressurizer safety valves. These valve position indicators can also be used to diagnose high RCS pressures or a stuck open safety valve (i.e., loss-of-coolant accident) at lower RCS pressures.

As discussed in Section 2.3.1 of this SE, the licensee proposes to remove the pressurizer safety valve position indicator instrument function from TS Tables 3.3-11 and 4.3-11 (item 14 in both tables) for Salem, Unit Nos. 1 and 2. The licensee provided the following justification with respect to the criteria in 10 CFR 50.36 (c)(2)(ii):

> The Pressurizer safety relief valves themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. The operability of the Pressurizer safety valves is therefore required by Unit 1 TS 3.4.2 and Unit 2 TS 3.4.2 & 3.4.3, "Safety Valves." However, the Pressurizer safety valve position indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 is not to be interpreted to include instrumentation installed to identify the source of actual leakage, for example valve position indicators. The instrumentation that satisfies Criterion 1 is contained in Salem Unit 1 TS 3.4.6.1 and Unit 2 TS 3.4.7.1 "Reactor Coolant System Leakage, Leakage Detection Systems."
>
> Pressurizer safety relief valve position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.
>
> While the function of the Pressurizer safety relief valves themselves is part of the primary success path in the UFSAR, the valve position indication is not part of the primary success path. UFSAR accident analysis assumes that the Pressurizer safety relief valves open as designed to reduce reactor pressure and no operator action based on valve position indication is required. Therefore, the Pressurizer safety valve position indication is not part of the primary success path as indicated in Criterion 3.
>
> The loss of Pressurizer safety valve position indication instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The NRC staff has reviewed the licensee's submittal and agrees with the licensee's conclusion that the pressurizer safety valve position indicator instrument function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) requiring inclusion of this item as a TS LCO.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97, Table 2, "PWR Variables," indicates that the pressurizer safety valve position indicator instrument function ("Primary System Safety Relief Valve Positions (including

PORV and code valves) or Flow Through or Pressure in Relief Valve Lines") is a Type D, Category 2 variable provided for operational status and to monitor loss of coolant.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," does not include the pressurizer safety valve position indicator instrument function.

With respect to the Salem licensing basis, UFSAR Table 7.5-3, "Index Type "A" Variables," does not include the pressurizer safety valve position instrument function as one of the Salem RG 1.97, Type A variables. UFSAR Table 7.5-4, "Summary of Instrumentation Compliance with Regulatory Guide 1.97," indicates that this function (i.e., "Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines") is considered Compliance Level 1. UFSAR Section 7.5.3.3 indicates that "Compliance Level 1" instrumentation meets the intent of RG 1.97. As such, the Salem licensing basis considers the pressurizer safety valve position instrument function as a Type D, Category 2 variable.

Based on its review and the preceding evaluation, the NRC staff concludes that the proposed deletion of the pressurizer safety valve position instrument function from TS Tables 3.3-11 and 4.3-11 (item 14 in both tables) and the associated footnotes in Table 3.3-11, for Salem, Unit Nos. 1 and 2, is acceptable because the instrumentation: (1) does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and (2) is not considered a Type A or Category 1 variable per the Salem licensing basis. Consistent with the licensee's letter dated August 7, 2017, the requirements being deleted from the TSs will be relocated to the Salem UFSAR. Further changes to the relocated requirements will be controlled via the requirements in 10 CFR 50.59.

3.2.5 Containment Pressure (Narrow Range)

As discussed in the LAR, the containment pressure indication provides information for assessing an inadequate containment cooling condition and for determining the potential challenge to the containment pressure retaining integrity. Salem, Unit Nos. 1 and 2, TS Tables 3.3-11 and 4.3-11 list both the narrow range containment pressure indication (item 15) and the wide-range containment pressure indication (item 16). However, only the wide-range instrument is used in the emergency operating procedures to define the potential for a challenge to containment integrity due to over-pressurization.

As discussed Section 2.3.1 of this SE, the licensee proposes to remove the containment pressure (narrow range) instrument function from TS Tables 3.3-11 and 4.3-11 (item 15 in both tables) for Salem, Unit Nos. 1 and 2. The licensee provided the following justification with respect to the criteria in 10 CFR 50.36 (c)(2)(ii):

> The containment pressure narrow instrument channels themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. The operability of the containment pressure narrow range channels is therefore required by Unit 1 and 2 TS 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation." However, containment pressure narrow range indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 is not to be interpreted to include instrumentation

installed to identify the source of actual leakage, for example valve position indicators.

Containment pressure narrow range indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

While the function of the containment pressure narrow range channels themselves is part of the primary success path in the UFSAR, the pressure indication is not part of the primary success path. UFSAR accident analyses assume automatic actuation of the safety injection and containment spray system based on containment narrow range pressures and no operator actions based on containment pressure narrow range indication are required. Therefore, the containment pressure narrow range indication is not part of the primary success path as indicated in Criterion 3.

The Containment Pressure narrow range indication has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The NRC staff has reviewed the licensee's submittal and agrees with the licensee's conclusion that the containment pressure (narrow range) instrument function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) requiring inclusion of this item as a TS LCO.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97, Table 2, "PWR Variables," indicates that containment pressure is a Type B, Category 1 variable provided for function detection, accomplishment of mitigation, and verification. Table 2 also indicates that containment pressure is a Type C, Category 1 variable provided for detection of potential for or actual breach and accomplishment of mitigation.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," does not include the containment pressure (narrow range) instrument function. However, Table 3.3.3-1 does include the containment pressure (wide-range) instrument function.

As discussed in the LAR, based on UFSAR Table 7.5-3, the current Salem licensing basis considers the containment pressure indication function as a Type A, Category 1 variable. However, no distinction is made between the use of wide range or narrow range instruments. The LAR indicates that the licensee has evaluated this classification and has determined that the narrow range containment pressure instrumentation is not a Type A variable. Accordingly, the LAR states that the UFSAR and associated design documents will be revised under the provisions of 10 CFR 50.59 to reflect that the narrow range containment pressure instrumentation is not a Type A variable.

Based on its review and the preceding evaluation, the NRC staff concludes that the proposed deletion of the containment pressure (narrow range) instrument function from TS Tables 3.3-11 and 4.3-11 (item 15 in both tables), for Salem, Unit Nos. 1 and 2, is acceptable because the instrumentation: (1) does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), and (2) is not considered a Type A or Category 1 variable per the proposed change to the Salem licensing basis. Consistent with the licensee's letter dated August 7, 2017, the requirements being deleted from the TSs will be relocated to the Salem UFSAR. Further changes to the relocated requirements will be controlled via the requirements in 10 CFR 50.59.

3.2.6 Wide Range Neutron Flux Monitors

As discussed Section 2.3.1 of this SE, the licensee proposes to add the wide range neutron flux monitors instrument function to TS Tables 3.3-11 and 4.3-11 (item 22 in both tables) for Salem, Unit Nos. 1 and 2.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97, Table 2, "PWR Variables," indicates that neutron flux is a Type B, Category 1 variable provided for function detection and accomplishment of mitigation. The table states that neutron flux should be measured over a range of 10^{-6} percent to 100 percent full power.
- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," includes two neutron flux instrument functions: power range neutron flux and source range neutron flux. The Bases for NUREG-1431 state, in part, that:

  > Power Range and Source Range Neutron Flux indication is provided to verify reactor shutdown. The two ranges are necessary to cover the full range of flux that may occur post accident.
  >
  > Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

As discussed in the LAR, the Gamma-Metrics Post-Accident Neutron Monitoring (PANM) system was installed at Salem, Unit Nos. 1 and 2, to provide reliable neutron flux monitoring in a harsh environment from plant shutdown to full power. The licensee stated that its evaluation determined that this instrument function should be classified as Category 1. The licensee also stated that in the wide-range mode, the PANM output range meets the requirement of RG 1.97.

The NRC staff concludes that the proposed addition of the wide range neutron flux monitor instrument function to TS Tables 3.3-11 and 4.3-11 (item 22 in both tables) is consistent with NUREG-1431 since the licensee's site-specific evaluation determined that this instrument function should be classified as RG 1.97, Category 1. The staff further concludes that since the wide range mode can monitor neutron flux from plant shutdown to full power, separate source range and power range entries in the TSs (as shown in NUREG-1431) are not needed. Based on its review and the preceding evaluation, the NRC staff concludes that the proposed change is acceptable.

3.2.7 Auxiliary Feedwater Storage Tank (Condensate Storage Tank) Water Level

As discussed above in Section 2.3.1 of this SE, the licensee proposes to add the auxiliary feedwater storage tank (condensate storage tank) instrument function to TS Tables 3.3-11 and 4.3-11 (item 23 in both tables) for Salem, Unit Nos. 1 and 2.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97, Table 2, "PWR Variables," indicates that condensate storage tank water level is a Type D, Category 1 variable provided to ensure water supply for auxiliary feedwater. The table notes that this function can be Category 3 if the tank is not the primary source of auxiliary feedwater.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," includes condensate storage tank level as one of the instrument functions. The Bases for NUREG-1431 state, in part, that at some plants, this function is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the operator.

With respect to the Salem licensing basis, UFSAR Table 7.5-3, "Index Type "A" Variables," lists the auxiliary feedwater storage tank (condensate storage tank) instrument function as a Type A variable. In addition, this function is listed as a Type A, Category 1 variable in the licensee's site-specific evaluation.

The NRC staff concludes that the proposed addition of the auxiliary feedwater storage tank (condensate storage tank) instrument function to TS Tables 3.3-11 and 4.3-11 (item 23 in both tables) is consistent with NUREG-1431 since the licensee's site-specific evaluation determined that this instrument function should be classified as RG 1.97, Type A, Category 1. Based on its review and the preceding evaluation, the NRC staff concludes that the proposed change is acceptable.

3.2.8 Containment Isolation Valve Position

As discussed in the licensee's application dated November 17, 2017, all instruments identified as Type A variables, or non-Type A but Category 1, were evaluated by the licensee and were proposed to be added into the Salem, Unit Nos. 1 and 2, TS Tables 3.3-11 and 4.3-11, with the following two exceptions: (1) Containment Isolation Valve (CIV) Position; and (2) Containment Hydrogen Concentration. However, in response to an NRC staff request for additional information dated September 11, 2017 (ADAMS Accession No. ML17254A738), the licensee subsequently determined that the CIV position instrument function, and associated notes, should be added to TS Tables 3.3-11 and 4.3-11 (item 24 in both tables) for Salem Unit Nos. 1 and 2, consistent with NUREG-1431. The proposed addition of this function to the TSs is discussed in the licensee's supplement dated October 18, 2017, and is evaluated below. The NRC staff's evaluation regarding the containment hydrogen concentration instrument function is discussed in Section 3.2.9 of this SE.

With respect to the guidance in RG 1.97 and NUREG-1431:

- RG 1.97 Table 2, "PWR Variables," indicates that the CIV position (excluding check valves) instrument function is shown as a Type B, Category 1 variable. The table indicates the purpose of this variable is to verify accomplishment of isolation.

- NUREG-1431, Table 3.3.3-1, "Post Accident Monitoring Instrumentation," includes "Penetration Flow Path Containment Isolation Valve Position" as one of the instrument functions. The Bases for NUREG-1431 states, in part, that this function is provided for verification of containment operability.

Salem UFSAR Table 7.5-4, "Summary of Instrumentation Compliance with Regulatory Guide 1.97," indicates that this function (i.e., "Containment Isolation Valve Position (excluding check valves) – Limit Switches") is considered Compliance Level 1. UFSAR Section 7.5.3.3 indicates that "Compliance Level 1" instrumentation meets the intent of RG 1.97. As such, the Salem licensing basis considers the CIV position instrument function as a Type B, Category 1 variable.

As noted in Section 2.2.2 of this SE, Type B variables provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control). For the CIV position instrument function, maintaining containment integrity is the applicable safety function.

The NRC staff concludes that the proposed addition of the CIV position indication instrument function, and associated notes, to TS Tables 3.3-11 and 4.3-11 (item 24 in both tables) is consistent with NUREG-1431 since the licensee's site-specific evaluation determined that this instrument function should be classified as RG 1.97, Type B, Category 1. Based on its review and the preceding evaluation, the NRC staff concludes that the proposed change is acceptable.

3.2.9 Containment Hydrogen Concentration

As discussed in Section 3.2.8 of this SE, the licensee is not proposing to add the containment hydrogen concentration instrument function to TS Tables 3.3-11 and 4.3-11.

As discussed in UFSAR Section 6.2.5.3, a hydrogen monitoring system is provided at Salem, Units Nos. 1 and 2, for continuous measurement of hydrogen concentration at two sample locations within containment. Data from sample locations allows for diagnosing beyond DBA events.

In RG 1.97, Table 2, "PWR Variables," the containment hydrogen concentration instrument function is shown as a Type C, Category 1 variable. Although RG 1.97 lists this variable as Category 1, subsequent rulemaking relaxed the requirements regarding hydrogen monitoring. Specifically, in a final rule dated September 16, 2003 (68 FR 54123), the NRC revised 10 CFR 50.44, "Combustible gas control for nuclear power reactors," to eliminate requirements for hydrogen recombiners and hydrogen purge systems, and to relax requirements for hydrogen and oxygen monitoring equipment. The statement of consideration for the final rule stated, in part, that:

> Currently, RG 1.97 recommends classifying the hydrogen monitors in Category 1, defined as applying to instrumentation designed for monitoring key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. Because the hydrogen monitors no longer meet the definition of Category 1 in RG 1.97, the NRC believes that licensees' current commitments are unnecessarily burdensome. The NRC believes that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of significant beyond design basis accidents. Category 3 applies to high-quality, off-the-shelf backup and diagnostic instrumentation. As with the revision to oxygen monitoring, this relaxation may also require a license amendment at some facilities.

Following the rulemaking, the Standard Technical Specifications (including NUREG-1431) were revised accordingly. As such, Table 3.3.3-1 in NUREG-1431 currently does not list containment hydrogen concentration as one of the PAM instrument functions.

With respect to the Salem licensing basis, Amendment s Nos. 281 and 284, for Salem, Unit Nos. 1 and 2, respectively, dated April 19, 2007 (ADAMS Package Accession No. ML070930262), deleted TS requirements related to hydrogen recombiners and hydrogen analyzers, consistent with the 2003 final rule for 10 CFR 50.44. Consistent with the statement of consideration in the final rule cited above, the NRC staff's SE stated that Category 3, as defined in RG 1.97, is the appropriate categorization for the hydrogen monitors.

Based on its review and the preceding evaluation, the NRC staff concludes that not including the containment hydrogen concentration instrument function in TS Tables 3.3-11 and 4.3-11 is acceptable since it is consistent with NUREG-1431 and the Salem licensing basis.

3.3 Proposed Changes to AOTs and Actions for Inoperable Channels and Other Changes

Changes to AOTs and Actions for Inoperable Channels in TS Table 3.3-11

The licensee has proposed to revise the AOTs and Actions for inoperable channels in the TABLE NOTATION of TS Table 3.3-11. Specifically, TS Table 3.3-11, Actions 1, 2, 4, and 6 would be revised as shown in Section 2.3.2 of this SE.

An AOT is specified in a nuclear power plant's TSs as the maximum time for which certain safety equipment can be placed out of service without requiring the plant to be put into a safer operating state. The AOT covers one or more modes of plant operation and it is constant for all plant configurations. In the STS, AOT is referred to as completion time. In the STS, with one channel inoperable beyond 30 days, a special report is required to be submitted to the NRC within the following 14 days. This report outlines the accident monitoring preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrument channel to operable status. With two inoperable channels for more than 7 days, either a plant shutdown or submittal of a special report is required, depending on the particular channel that is out of service. The STS also contains provisions that permit a separate condition entry for each inoperable instrument function.

Based on its review of NUREG-1431, the NRC staff concludes that the proposed changes to the AOTs and actions for inoperable channels in TS Table 3.3-11 provide completion times and actions that are reasonable and provide better alignment with NUREG-1431 than the current Salem, Units Nos. 1 and 2, TSs. Therefore, the changes are acceptable.

Change to TS 3.3.3.7

As shown in the supplement dated August 7, 2017, the licensee has proposed to revise the TS 3.3.3.7 to add a note stating that "Separate Condition entry is allowed for each Function." This changes allows each instrument the full associated LCO time if multiple PAM instruments become inoperable at different times. This change is consistent with NUREG-1431 and, therefore, is acceptable.

Change to TS 6.9.4

TS 6.9.4, "Special Reports," currently requires that a special report be submitted within 14 days for TS Table 3.3-11, Actions 8 or 9. TS 6.9.4 would be revised to also require that a special report be submitted within 14 days for TS Table 3.3-11, Actions 1 or 4. This change is consistent with the proposed changes to Actions 1 and 4 in TS Table 3.3-11 shown in Section 2.3.2 of this SE. Therefore, this change is acceptable.

3.4 Technical Evaluation Conclusion

Based on the considerations discussed in Sections 3.2 and 3.3 of this safety evaluation, the NRC staff concludes that the proposed changes to the TSs for Salem, Unit Nos. 1 and 2: (1) will continue to meet the requirements of 10 CFR 50.36(c)(2) in that the LCOs describe the lowest functional capability or performance level of the accident monitoring instrumentation required for safe operation of the facility; and (2) will continue to meet the requirements of 10 CFR 50.36(c)(3) in that the SRs provide assurance that the accident monitoring instrumentation LCOs will be met. Therefore, the staff further concludes that the proposed amendments are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments on February 6, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (January 17, 2017; 82 FR 4931). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: H. Vu
R. Ennis

Date: November 14, 2017

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS RE: ACCIDENT MONITORING INSTRUMENTATION (CAC NOS. MF8859 AND MF8860; EPID L-2016-LLA-0041) DATED NOVEMBER 14, 2017

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