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

January 12, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: REVISE THE DEFINITION OF FULLY WITHDRAWN FOR THE ROD CLUSTER CONTROL ASSEMBLIES (TAC NOS. ME0900 AND ME0901)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment Nos. 292 and 276 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 22, 2009.

The amendments revise the definition of the fully withdrawn position of the Rod Cluster Control Assemblies (RCCAs) to minimize localized RCCA wear. Previously, the fully withdrawn position for the RCCAs was defined in the TSs as being within the interval of 222 to 228 steps withdrawn (i.e., steps above rod bottom). The approved change allows the fully withdrawn position to be defined as being within the interval of 222 to 230 steps withdrawn.

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

Sincerely,

RB Ennis

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 292 to License No. DPR-70
2. Amendment No. 276 to License No. DPR-75
3. Safety Evaluation

cc w/encls: 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. 292
License No. DPR-70

1. The 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 March 22, 2009, 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 Appendices A and B, as revised through Amendment No. 292, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to entering Mode 2 following refueling outage 1R20 (currently scheduled for spring 2010).

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and the Technical Specifications

Date of Issuance: January 12, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 292

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

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

Remove	Insert
Page 4	Page 4

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
1-3	1-3
3/4 1-19	3/4 1-19
3/4 1-21	3/4 1-21

(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

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 292, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(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 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 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 Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

> PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 230 steps withdrawn, inclusive. FULLY WITHDRAWN will be specified in the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2.1 The shutdown and control rod position indication systems shall be OPERABLE and capable of determining the actual and demanded rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

All Shutdown Banks: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

Control Bank A: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

Control Bank B: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 160-230 steps.

Control Bank C and D: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-230 steps.

b. Group demand counters; ± 2 steps of the pulsed output of the Slave Cycler Circuit over the withdrawal range of 0-230 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours* and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

* During Cycle 14, the position of Rod 1SB2 will be determined indirectly by the movable incore detectors within 8 hours following its movement until the repair of the indication system for this rod. During reactor startup, the fully withdrawn position will be determined by current traces and subsequently verified by the movable incore detectors prior to entry into Mode 1.

b. With two or more analog rod position indicators per bank inoperable, within one hour restore the inoperable rod position indicator(s) to OPERABLE status or be in HOT STANDBY within the next 6 hours. A maximum of one rod position indicator per bank may remain inoperable following the hour, with Action (a) above being applicable from the original entry time into the LCO.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 230 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ≤71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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. 276
License No. DPR-75

1. The 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 March 22, 2009, 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 Appendices A and B, as revised through Amendment No. 276, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to entering Mode 2 following refueling outage 2R18 (currently scheduled for spring 2011).

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and the Technical Specifications

Date of Issuance: January 12, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 276

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

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

Remove	Insert
Page 4	Page 4

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
1-3	1-3
3/4 1-16	3/4 1-16
3/4 1-18	3/4 1-18

(2) Technical Specifications and Environmental Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 276, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Special Low Power Test Program

PSE&G shall complete the training portion of the Special Low Power Test Program in accordance with PSE&G's letter dated September 5, 1980 and in accordance with the Commission's Safety Evaluation Report "Special Low Power Test Program", dated August 22, 1980 (See Amendment No. 2 to DPR-75 for the Salem Nuclear Generating Station, Unit No. 2) prior to operating the facility at a power level above five percent.

Within 31 days following completion of the power ascension testing program outlined in Chapter 13 of the Final Safety Analysis Report, PSE&G shall perform a boron mixing and cooldown test using decay heat and Natural Circulation. PSE&G shall submit the test procedure to the NRC for review and approval prior to performance of the test. The results of this test shall be submitted to the NRC prior to starting up following the first refueling outage.

(4) Initial Test Program

PSE&G shall conduct the post-fuel-loading initial test program (set forth in Chapter 13 of the Final Safety Analysis Report, as amended) without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

(a) Elimination of any test identified in Chapter 13 of the Final Safety Analysis Report, as amended, as essential;

(b) Modification of test objectives, methods or acceptance criteria for any test identified in Chapter 13 of the Final Safety Analysis Report, as amended, as essential;

(c) Performance of any test at a power level different by more than five percent of rated power from there described; and

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 230 steps withdrawn, inclusive. FULLY WITHDRAWN will be specified in the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2.1 The shutdown and control rod position indication systems shall be OPERABLE and capable of determining the actual and demanded rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

All Shutdown Banks: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

Control Bank A: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

Control Bank B: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 160-230 steps.

Control Banks C and D: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal range of 0-230 steps.

b. Group demand counters; ± 2 steps of the pulsed output of the Slave Cycler Circuit over the withdrawal range of 0-230 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per bank inoperable either:

    1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

    2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With two or more analog rod position indicators per bank inoperable, within one hour restore the inoperable rod position indicator(s) to OPERABLE status or be in HOT STANDBY within the next 6 hours. A maximum of one rod position indicator per bank may remain inoperable following the hour, with Action (a) above being applicable from the original entry time into the LCO.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 230 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 76% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 292 AND 276 TO 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 application dated March 22, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090920406), PSEG Nuclear LLC (PSEG or the licensee) submitted an amendment request for Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The proposed amendment would revise the definition of the fully withdrawn position of the Rod Cluster Control Assemblies (RCCAs) to minimize localized RCCA wear. Currently, the fully withdrawn position for the RCCAs is defined in the Technical Specifications (TSs) as being within the interval of 222 to 228 steps withdrawn (i.e., steps above rod bottom). The proposed change would allow the fully withdrawn position to be defined as being within the interval of 222 to 230 steps withdrawn. The following TSs would be revised by the proposed amendment:

1) TS Definition 1.13a, "FULLY WITHDRAWN"

2) TS 3.1.3.3, "Rod Drop Time"

3) TS 3.1.3.2.1, "Position Indication Systems - Operating"

In addition to revising the above TSs to change the step value from "228" to "230," other minor editorial changes are proposed in wording for consistency between the Salem Unit No.1 and Salem Unit No. 2 TSs.

The licensee's application dated March 22, 2009, provided the following background information regarding the reason for the proposed change:

> During Salem Unit 1 Fall outage 2005 (1 R17), RCCA examinations were conducted on 53 RCCAs by the Salem RCCA vendor AREVA NP. The examinations were conducted to investigate possible clad wear indications at

Enclosure

certain guide card locations, at the continuous guide block area and at the tip of each pin. These examinations were consistent with issues previously identified in NRC Information Notice 87-19, and with previous RCCA repositioning actions taken to address the issues [Salem Unit Nos. 1 and 2, Amendments 91 and 66 dated March 22, 1989 (ADAMS Accession No. ML011690019)]. Information Notice 87-19 identified the potential for RCCA clad wear when the RCCAs are left "parked" in the same FULLY WITHDRAWN position over several cycles. The wear is a result of core flow induced vibration which causes contact between the RCCAs cladding and the RCCA guide cards located in the upper reactor internals.

Both eddy current (ET) and ultrasound (UT) techniques were employed to measure the cladwear. The ET examinations provided an estimation of clad wear in terms of loss of cross-sectional area; the follow-up UT examinations provided more detail including depth, azimuth, circumferential extent and shape. The examination results indicated 49 of the 53 RCCAs had wear indications; none exceeded the acceptance criteria for continued use. In order to minimize the accrual of wear during future operation, AREVA NP has recommended that the axial repositioning program, initiated by [Salem Unit Nos. 1 and 2, Amendments 91 and 66], be expanded to include axial positions above 228. AREVA NP also recommended that the axial repositioning of RCCAs be continuously cycled each refuel outage to further minimize wear. This will maximize the wear area against the guide card and may also change the flow induced vibration characteristics of the rods tips in the guide tubes. The specific FULLY WITHDRAWN position will be specified in the reload analysis for each cycle (as per TS Definition 1.1 3a).

2.0 REGULATORY EVALUATION

The RCCAs, which are part of the reactivity control system, are discussed in Section 4 of the Salem Updated Final Safety Analysis Report (UFSAR). Fifty-three RCCAs are employed. Each RCCA consists of a group of individual neutron absorber rods (i.e., control rods) fastened at the top end to a common spider assembly. The absorber material in the control rods is sealed in stainless steel tubes which fit in the control rod guide tubes in the fuel assemblies.

The RCCAs are divided into two functional categories; control and shutdown. Each category has four individual groups (also called "banks"). The control banks compensate for reactivity changes due to variations in operating conditions of the reactor. The shutdown banks are in the fully withdrawn position during normal operation and provide a large negative reactivity insertion upon a reactor trip.

The licensee addressed the regulatory requirements applicable to the proposed amendment in Section 5.2 of Attachment 1 to the application dated March 22, 2009. The regulatory requirements, criteria, and guidance which the NRC staff applied in its review are discussed below.

As discussed in Section 3.1 of the Salem UFSAR, the general design criteria (GDC) followed in the design of Salem Units 1 and 2 are the Atomic Industrial Forum (AIF) version, as published in a letter to the Atomic Energy Commission (AEC) from E. A. Wiggin, AIF, dated October 2, 1967. As also discussed in Section 3.1 of the UFSAR, in addition to the AIF GDC, the Salem units were designed to comply with the intent of the AEC's proposed GDC dated July 1967. Section 3.1.2 of the UFSAR provides a discussion of the Salem's conformance with the AEC proposed GDCs. Section 3.1.3 of the UFSAR also states that the design of the Salem units conforms to the intent of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), "General Design Criteria for Nuclear Power Plants," dated July 7, 1971 (with several exceptions as discussed in this section of the UFSAR).

The following GDC in Appendix A to 10 CFR Part 50 is applicable to the proposed amendment:

- GDC 26, "Reactivity control system redundancy and capability," which states that:

  Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Section 3.1.3 of the Salem UFSAR does not list any exceptions to GDC 26.

NRC Information Notice (IN) 87-19, "Perforation and Cracking of Rod Cluster Control Assemblies," dated April 9, 1987 (ADAMS Accession No. ML031140513) notified all Westinghouse pressurized-water reactor (PWRs) facilities of operating experience at several PWRs that indicated excessive wear of the tubing in the RCCA rods. The IN noted that several licensees have been given approval to slightly change the position of the fully-withdrawn RCCA in order to distribute the wear among different locations on the RCCA tubing.

On March 22, 1989, the NRC staff issued Amendment Nos. 91 and 66 for Salem Unit Nos. 1 and 2, respectively (ADAMS Accession No. ML011690019). This amendment redefined the definition of the fully withdrawn position of the RCCAs from 228 steps to a band between 222 and 228 steps. Similar to the current amendment proposed by PSEG, the intent of the change in the definition of the fully withdrawn position of the RCCAs (approved in Amendment Nos. 91 and 66) was to minimize RCCA tubing wear.

3.0 TECHNICAL EVALUATION

The purpose of this amendment request is to change the definition of fully withdrawn to mean between steps 222 and 230 instead of steps 222 to 228. The change would allow the RCCAs to be left at steps 229 and 230 so that the wear on the cladding would not be in the same location and cause possible failure of the rod. Each step is 5/8-inch. As described in the licensee's application dated March 22, 2009, step 225 is at the top of the active fuel region.

Moving the RCCAs from step 228 to 230 will cause the RCCAs to be withdrawn an additional 1.25 inches from the active fuel region. As discussed in the licensee's application dated

March 22, 2009, since no change is being made to the lowest allowable position (i.e., step 222), prior assessments regarding minimal rod insertion into the active fuel region remain applicable and unchanged. As such, the NRC agrees with the licensee's assessment that the proposed amendment will have no impact on previously analyzed conditions for axial and radial power distributions, critical boron concentrations, and temperature-dependent shutdown margins.

Currently, TS 3.1.3.3 states that the rod drop time from 228 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry. The proposed amendment would change the step value from 228 to 230; however, the rod drop time limit would remain at 2.7 seconds. Since the proposed amendment would allow for the RCCAs to be withdrawn further out of the core, the actual insertion time would be longer. In accordance with TS surveillance requirement 4.1.3.3, rod drop times of full length rods are measured prior to criticality following a refueling outage. The licensee's application dated March 22, 2009, provided the following discussion regarding the expected impact of the proposed amendment on the actual rod drop time:

> Utilizing data from the most recent three Salem cycles (both Unit 1 and 2), a calculation was performed to determine the maximum rod drop time from a fully withdrawn position of 230.
>
> For each set of rod drop tests, the time to reach the dashpot was measured for each of the 53 RCCA locations. From the six sets of measurements the largest standard deviation was used along with a conservative tolerance factor to estimate the longest potential drop time from 228 steps to the dashpot at 1.4568 seconds.
>
> The number of rod steps from the current upper location of 228 steps to the dashpot is 196 rod steps. Taking the longest time over the distance from 228 to the dashpot results in 0.0074 sec/step (1.4568 / 196 steps). For the two additional steps to 230, the net additional time is 0.0148 seconds (0.0074 x 2). Therefore the longest expected total rod drop time from position 230 is: 1.4658 + 0.0148 = 1.4716 seconds (this bounds both Salem Units 1 and 2).
>
> The total maximum time of 1.4716 seconds is a small increase from the current measurements from 228 steps. The RCCA negative reactivity insertion characteristics from 230 steps withdrawn will remain bounded by the accident analysis assumption of 2.7 seconds, thus the proposed change does not result in a significant reduction in the margin of safety assumed in the UFSAR. Prior to each operating cycle the actual rod drop times are required to be confirmed as less than or equal to 2.7 seconds per TS Surveillance 4.1.3.3.

Based on the information discussed above, the NRC staff finds that the proposed amendment is acceptable with respect to the impact on control rod drop time.

As discussed in the application dated March 22, 2009, the licensee evaluated the impact of the proposed amendment on loss-of-coolant accident (LOCA) considerations. The licensee stated that since radial and axial power distributions are not impacted, there is no impact on small-break LOCA events. The licensee also stated that the proposed amendment does not impact the large-break LOCA analysis. Specifically, the licensee stated that the LOCA analysis used to determine the core response in terms of peak clad temperature and cladding oxidation does not take credit for control rod insertion to shut down the reactor. Post-LOCA long term cooling and core sub-criticality do take credit for insertion of control rods; however, the net control rod reactivity worth is not impacted by the proposed change. Based on the information provided by the licensee, the NRC staff finds that the proposed amendment is acceptable with respect to the impact on small-break and large-break LOCA analyses.

The licensee's application dated March 22, 2009, stated that the proposed amendment would not reduce the capability of the RCCAs to function mechanically during design basis events. The licensee stated, in part, that:

> The RCCAs remain inserted in the guide thimbles by a sufficient margin even at the fully withdrawn position of 230 steps (the insertion is only reduced by approximately 1.25 inches, with approximately 5.70 inches of insertion remaining). The RCCAs and protective guide tubes (and thimbles) are subjected to the same mechanical stresses in any position from Step 222 to Step 230 due to the fixed lateral support locations in both the upper internals and the fuel assemblies.

Based on the information discussed above, the NRC staff finds that the proposed amendment is acceptable with respect to the capability of the RCCAs to function mechanically during a design basis event.

In addition to the TS changes proposed in PSEG's application dated March 22, 2009, the NRC staff has made the following corrections to Salem Unit No. 1 TS page 3/4 1-19:

1) In TS 3.1.3.2.1.a, "All Shutdnwn Banks" is changed to "All Shutdown Banks"

2) In TS 3.1.3.2.1.a, "Control Rank A" is changed to "Control Bank A"

Both of the above typographical errors were introduced in Salem Unit No. 1 Amendment 230, dated May 26, 2000 (ADAMS Accession No. ML003719424) as a result of an incorrect mark up included in the licensee's application dated May 3, 2000 (ADAMS Accession No. ML003712864) for that amendment.

Technical Evaluation Conclusion

Based on the above evaluation, the NRC staff finds that there is reasonable assurance that the proposed amendment will not impact the capability of the RCCAs to function as designed and as assumed in the Salem UFSAR. Therefore, the NRC staff concludes that the proposed amendment is 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. 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. 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 (74 FR 26435). 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) 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: J. Miller
R. Ennis

Date: January 12, 2010

January 12, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: REVISE THE DEFINITION OF FULLY WITHDRAWN FOR THE ROD CLUSTER CONTROL ASSEMBLIES (TAC NOS. ME0900 AND ME0901)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment Nos. 292 and 276 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 22, 2009.

The amendments revise the definition of the fully withdrawn position of the Rod Cluster Control Assemblies (RCCAs) to minimize localized RCCA wear. Previously, the fully withdrawn position for the RCCAs was defined in the TSs as being within the interval of 222 to 228 steps withdrawn (i.e., steps above rod bottom). The approved change allows the fully withdrawn position to be defined as being within the interval of 222 to 230 steps withdrawn.

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

Sincerely,
/ra/
Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 292 to License No. DPR-70
2. Amendment No. 276 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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