October 14, 2005

Mr. William Levis
Senior Vice President & Chief Nuclear Officer
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: EMERGENCY CORE COOLING SYSTEM ACCUMULATORS (TAC NOS. MC6416 AND MC6417)

Dear Mr. Levis:

The Commission has issued the enclosed Amendment Nos. 267 and 249 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 (TSs) in response to your application dated March 4, 2005, as supplemented by letter dated August 2, 2005.

These amendments extend the completion time (CT) from 1 hour to 24 hours for Actions “a” and “b” of TS 3.5.1, “Accumulators,” which defines requirements for the emergency core cooling system accumulators. Actions “a” and “b” of TS 3.5.1 specify a CT to restore an accumulator to operable status when it has been declared inoperable for reasons other than boron concentration in the accumulator not being within the required range.

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/

Stewart N. Bailey, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 267 to License No. DPR-70
2. Amendment No. 249 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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Salem Nuclear Generating Station, Unit Nos. 1 and 2

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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. 267
License No. DPR-70

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

   A. The application for amendment filed by PSEG Nuclear LLC (PSEG) on behalf of PSEG and Exelon Generation Company, LLC (the licensees) dated March 4, 2005, as supplemented on August 2, 2005, 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. 267, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 14, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 267

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of the Appendix A, Technical Specifications, 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 Page	Insert Page
3/4 5-1	3/4 5-1

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. 249
License No. DPR-75

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

   A. The application for amendment filed by PSEG Nuclear LLC (PSEG) on behalf of PSEG and Exelon Generation Company, LLC (the licensees) dated March 4, 2005, as supplemented on August 2, 2005, 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. 249, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 14, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 249

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of the Appendix A, Technical Specifications, 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 Page	Insert Page
3/4 5-1	3/4 5-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 267 AND 249 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 letter dated March 4, 2005, as supplemented by letter dated August 2, 2005, PSEG Nuclear LLC (the licensee) submitted a request for changes to the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, Technical Specifications (TSs). The requested changes would extend the completion time (CT) from 1 hour to 24 hours for Actions "a" and "b" of TS 3.5.1, "Accumulators," which defines requirements for the emergency core cooling system (ECCS) accumulators. Actions "a" and "b" of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for reasons other than boron concentration in the accumulator not being within the required range. The application and supplement may be found in the Nuclear Regulatory Commission's (NRC or the Commission) Agencywide Documents Access and Management System (ADAMS) using Accession Nos. ML050740454 and ML052210497. The August 2, 2005, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination, as published in the Federal Register on May 24, 2005 (70 FR 29800).

2.0 REGULATORY EVALUATION

The Westinghouse Owners Group (WOG) submitted Topical Report WCAP-15049, "Risk-Informed Evaluation of an Extension to Accumulator Completion Times," to the NRC on August 20, 1998. The WCAP evaluates the risk associated with extending the accumulator CT from 1 hour to 24 hours when an accumulator is declared inoperable for reasons other than its boron concentration being out of specification. The NRC staff approved the topical report in a letter dated February 19, 1999. The WOG subsequently submitted the approved version of the topical report, WCAP-15049-A, by its letter dated May 18, 1999.

Wolf Creek was the lead plant for the WOG program and received plant-specific approval for changes to the TSs on April 27, 1999 (License Amendment No. 124). In the NRC letter of February 19, 1999, the staff indicated that it will not repeat its review of the matters described in

Topical Report WCAP-15049 when the report appears as a reference in license applications, except to ensure that the material presented applies to the specified plants involved.

The WOG, through the industry's Technical Specification Task Force (TSTF), proposed a generic change to the standard technical specifications (STSs) for Westinghouse plants (NUREG-1431). This proposed generic TSs change, identified as TSTF-370, revises the CT from 1 hour to 24 hours for Condition B of TS 3.5.1, and its associated Bases. Condition B of TS 3.5.1 in the STS currently specifies a CT of 1 hour to restore a reactor coolant system (RCS) accumulator to operable status when it has been declared inoperable due to any reason except not being within the required boron concentration range. Following its review of TSTF-370, the NRC staff, following the consolidated line item improvement process (CLIIP), issued a notice of opportunity for comment on this model Safety Evaluation (SE) and a model no significant hazards consideration determination (July 15, 2002, 67 FR 46542). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications using the CLIIP (March 12, 2003, 68 FR 11880).

The plant-specific TSs for Salem, Units 1 and 2, have not been converted into the STS format. The equivalent of Condition B of the STS is addressed in Actions "a" and "b" of the Salem TS 3.5.1. Action "a" addresses the condition of an accumulator declared inoperable for reasons other than as a result of a closed isolation valve or boron concentration. Action "b" addresses the condition of an accumulator declared inoperable as a result of an isolation valve being closed. Condition B of STS 3.5.1 addresses an accumulator declared inoperable for any reason other than boron concentration being outside its limits.

3.0 TECHNICAL EVALUATION

Deterministic Evaluation

The purpose of the ECCS accumulators is to supply water to the reactor vessel during the blowdown phase of a loss-of-coolant accident (LOCA). The accumulators are large volume tanks, filled with borated water and pressurized with nitrogen. The cover pressure is less than that of the RCS so that following an accident, when the RCS pressure decreases below tank pressure, the accumulators inject the borated water into the RCS cold-leg piping. The current deterministic safety analysis has not been changed and, therefore, the limiting condition for operation (LCO) (i.e., the lowest functional capability required for safe operation) continues to be:

LCO 3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,
b. A contained volume of between 6223 and 6500 gallons of borated water,
c. A boron concentration of between 2200 and 2500 ppm, and
d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

Applicability: Modes 1 and 2, Mode 3*

*Pressurizer pressure above 1000 psig.

TS Actions allow for limited deviations from the LCO. Historically, these Actions and associated CTs have been set using judgment and are not part of the deterministic safety analysis discussed above. Currently, the TS allows for one accumulator to be inoperable for 1 hour for reasons other than boron concentration not within limits during Modes 1, 2, and in Mode 3 with pressurizer pressure greater than 1000 psig. The WCAP, as well as this TSTF, proposes to increase this CT to 24 hours. The proposed CT of 24 hours is an extension of the current ACTION statements "a" and "b" in the Salem TS. CTs are determined by considering risk. The risk implications of extending the CT are reviewed in the following section.

Risk Evaluation

A three-tiered approach, consistent with Regulatory Guide (RG) 1.177[1], was used by the NRC staff to evaluate the risk associated with the proposed accumulator CT, or allowed outage time (AOT), extension from 1 hour to 24 hours. The need for the proposed change was that the current 1 hour CT would be insufficient in most cases for licensees to take a reasonable action when an accumulator was found to be inoperable.

Tier 1: Quality of Probabilistic Risk Assessment (PRA) and Risk Impact

Westinghouse used a reasonable approach to assess the risk impact of the proposed accumulator CT extension. The approach is generally consistent with the intent of the applicable NRC RGs 1.174[2] and 1.177. The quantitative risk measures addressed in the topical report included the change in core damage frequency (CDF) and incremental conditional core damage probability (ICCDP[3]) for a single CT. The change in large early release frequency (LERF) and incremental conditional large early release probability (ICLERP[4]) for a single CT was qualitatively addressed. Representative calculations were performed to determine the risk impact of the proposed change. Various accumulator success criteria were considered in these calculations to encompass the whole spectrum of Westinghouse plants, e.g., two-, three- and four-loop plants. A reasonable effort was also made to address the differences in other components of risk analysis such as initiating event (IE) frequency and accumulator unavailability among Westinghouse plants.

Westinghouse considered a comprehensive range of IEs in the risk analysis. LOCAs in all sizes - large, medium and small - were included, and reactor vessel failure and interfacing

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[1]RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," September 1998.

[2]RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

[3]ICCDP = (conditional CDF with the subject equipment out-of-service) - (baseline CDF with nominal expected equipment unavailabilities) x (duration of single CT under consideration).

[4]ICLERP = (conditional LERF with the subject equipment out-of-service) - (baseline LERF with nominal expected equipment unavailabilities) x (duration of single CT under consideration).

system LOCAs were also considered. Modeling of accumulators for mitigation of events other than large-, medium-, and small-LOCAs was identified to have insignificant risk impact; therefore, the analysis was performed only on accumulator injection in response to large-, medium-, and small-LOCA events. The success criteria considered are summarized as follows:

LOCA Category	No. of Loops	Success Criteria
Large	4	3 accumulators to 3 of 3 intact loops (3/3) 2 accumulators to 2 of 3 intact loops (2/3) no accumulators required (0/3)
	3	2 accumulators to 2 of 2 intact loops (2/2) 1 accumulator to 1 of 2 intact loops (1/2) no accumulators required (0/2)
	2	1 accumulator to 1 of 1 intact loop (1/1) no accumulators required (0/1)
Medium and Small	4	3 accumulators to 3 of 3 intact loops (3/3)
	3	2 accumulators to 2 of 2 intact loops (2/2)
	2	1 accumulator to 1 of 1 intact loop (1/1)

The success criteria considered in this analysis were comprehensive and considered conservative in many cases. For example, many plants indicated the accumulator success criteria for medium- and small-LOCA events resulted from their role in an alternate success path, in which high-pressure injection (HPI) had already failed. Additionally, the NRC staff's review of a number of the original individual plant examinations (IPEs) indicated that no accumulator was needed at all for many medium-LOCA sequences and for most small-LOCA sequences.

The fault trees that model accumulator unavailabilities were evaluated. The assumptions made in the fault tree modeling were detailed and were found to be reasonable. For example, the model assumed that the total CT would be used for each corrective maintenance, and this was considered conservative. A comprehensive list of failure mechanisms was considered, and potential common cause failures for check valves and motor-operated valves were also included. Westinghouse used the Multiple Greek Letter technique to determine the common cause failure contributions to the accumulator injection failure.

The component failure rates were taken from the Advanced Light Water Utility Requirements Document.[5] Accumulator unavailabilities due to boron concentration out-of-limit and due to other reasons were calculated based on a survey of a number of Westinghouse plants. The values for component failure rates and accumulator unavailabilities were within reasonable ranges. The common cause factors used were also comparable to those used in other PRAs. The accumulator fault trees were quantified using the WesSAGE computer code. The code

[5]"Advanced Light Water Utility Requirements Document," Volume II, ALWR [advanced light-water reactor] Evolutionary Plant, Chapter 1, Appendix A, PRA Key Assumptions and Ground Rules, Rev. 5, Issued December 1992.

provided information on the unavailability and cutsets related to the component failures and maintenance activities modeled in the fault trees. A separate hand calculation was used to determine the unavailability due to potential common cause failures. An evaluation of some of the cutsets provided in the topical report did not reveal any unexpected results.

The NRC staff examined the accident sequence identification for each LOCA category. The probability of the sequence leading to core damage involving accumulator failure is summarized for each LOCA category as follows:

Large-LOCA	(Large-LOCA IE frequency) x (accumulator unavailability)
Medium-LOCA	(Medium-LOCA IE frequency) x (unavailability of HPI) x (accumulator unavailability)
Small-LOCA	(Small-LOCA IE frequency) x (unavailability of HPI) x (accumulator unavailability)

The LOCA IE frequencies used for WCAP-15049 are summarized below. Also listed are the LOCA frequencies used in NUREG/CR-4550[6] (the NUREG-1150 study) for pressurized-water reactors and those in the original IPEs.

	WCAP-15049	NUREG-1150	IPE Average (High; Low)
Large-LOCA	$3x10^{-4}$/yr	$5x10^{-4}$/yr	$3.3x10^{-4}$/yr ($5x10^{-4}$/yr; $1x10^{-5}$/yr)
Medium-LOCA	$8x10^{-4}$/yr	$1x10^{-3}$/yr	$7.9x10^{-4}$/yr ($2.6x10^{-3}$/yr; $1x10^{-4}$/yr)
Small-LOCA	$7x10^{-3}$/yr	$1x10^{-3}$/yr	$8.9x10^{-3}$/yr ($2.9x10^{-2}$/yr; $3.7x10^{-4}$/yr)

Westinghouse indicated that the IE frequencies for WCAP-15049 were based on the plant-specific information contained in the WOG probabilistic safety assessment Comparison Database, which documented the PRA modeling methods and results of the updated PRAs for Westinghouse plants. The mean IE frequencies were used for the risk analysis. These were comparable to the values used for the NUREG-1150 study and the average values in the original IPEs. The NRC staff also found that the IE frequency values in high range among the original IPEs were not much higher than those used for this topical report. The HPI unavailability values used were $7x10^{-3}$ and $1x10^{-3}$/yr for medium- and small-LOCA events, respectively. The NRC staff's examination revealed that the HPI unavailability values were generally comparable to those used in other PRAs, and were generally conservative.

The risk measures calculated to determine the impact on plant risk were based on three different cases. The risk measures considered in each case included the impact on CDF and ICCDP for a single CT, and the impact on LERF and ICLERP for a single CT were qualitatively considered. The three cases considered were:

[6]NUREG/CR-4550, "Analysis of Core Damage Frequency: Internal Events Methodology," Vol. 1, Rev. 1, January 1990.

<u>Design basis case</u>. This case required accumulator injection only for mitigation of large-LOCA events (3/3 for 4-loop, 2/2 for 3-loop, and 1/1 for 2-loop).

<u>Case 1</u>. This case credited realistic accumulator success criteria (2/3 for 4-loop, 1/2 for 3-loop, and 0/1 for 2-loop) for large-LOCA events and credited the use of accumulators in responding to medium- and small-LOCA events (3/3, 2/2, and 1/1 for 4-loop, 3-loop, and 2-loop, respectively) following failure of HPI.

<u>Case 2</u>. This case credited more realistic improved accumulator success criteria (no accumulator required) for large-LOCA events and credited the use of accumulators in responding to medium- and small-LOCA events (3/3, 2/2, and 1/1 for 4-loop, 3-loop, and 2-loop, respectively) following failure of HPI.

The results were summarized as follows:

Case	LOCA CDF(/yr) (Current)	LOCA CDF(/yr) (Proposed)	^{a}CDF	ICCDP
4-loop Design Basis	6.93×10^{-7}	9.24×10^{-7}	2.31×10^{-7}	8.20×10^{-7}
4-loop Case 1	6.23×10^{-8}	7.77×10^{-8}	1.54×10^{-8}	5.53×10^{-8}
4-loop Case 2	4.57×10^{-8}	6.09×10^{-8}	1.52×10^{-8}	5.41×10^{-8}
3-loop Design Basis	4.62×10^{-7}	6.18×10^{-7}	1.56×10^{-7}	8.21×10^{-7}
3-loop Case 1	4.27×10^{-8}	5.31×10^{-8}	1.04×10^{-8}	5.48×10^{-8}
3-loop Case 2	3.05×10^{-8}	4.08×10^{-8}	1.03×10^{-8}	5.42×10^{-8}
2-loop Design Basis	2.31×10^{-7}	3.09×10^{-7}	7.80×10^{-8}	8.21×10^{-7}
2-loop Case 1	1.52×10^{-8}	2.04×10^{-8}	5.20×10^{-9}	5.42×10^{-8}
2-loop Case 2	1.52×10^{-8}	2.04×10^{-8}	5.20×10^{-9}	5.42×10^{-8}

For both realistic cases, the ÎCDFs and ICCDPs were very small for 2-loop, 3-loop, and 4-loop plants, and were much below the numerical guidelines in RGs 1.174 and 1.177. The NRC staff also noted that the values were considered still bounding in the sense that the risk analysis used a multitude of conservative assumptions and data in the modeling. For many Westinghouse plants, the realistic impact on risk would be much smaller than the values above.

A set of sensitivity cases was also calculated using higher IE frequencies for small- and medium-LOCAs. The results of the sensitivity calculations did not cause the overall risk impact to increase significantly.

Westinghouse indicated that accumulator success or failure does not directly impact the containment performance; therefore, the change in LERF would be proportional to the increased CDF due to accumulator failures. Westinghouse concluded that, since the impact on CDF was small, the impact on LERF would also be small. The NRC staff found the Westinghouse argument to be acceptable, and concluded that the impact on LERF and ICLERP for a single CT was very small.

One of the potential benefits of the proposed extended CT was the averted risk associated with avoiding a forced plant shutdown and startup. The risk associated with a forced plant shutdown and ensuing startup due to the inflexibility in current TSs could be significant in comparison with the risk increase due to the proposed accumulator CT increase.

Based on the NRC staff's Tier 1 review, the quality of risk analysis used to calculate the risk impact of the proposed accumulator CT extension was reasonable and generally conservative. It was also found that the risk impact of the proposed change was below the NRC staff guidelines in RGs 1.174 and 1.177.

Tiers 2 and 3: Configuration Risk Control

Tier 2 of RG 1.177 addresses the need to preclude potentially high-risk configurations which could result if certain equipment is taken out of service during implementation of the proposed TS change (in this case, accumulator CT). If such configurations are identified, the licensee should also identify appropriate measures to avoid them.

The accumulators are always needed to mitigate large-LOCAs. Large-LOCAs require accumulators to inject as analyzed under Tier 1 in order to avoid core damage. This means that if a large-LOCA occurs without the accumulator function, the core will be damaged independently of whether other systems, such as HPI, function properly or not. However, the probability that a large-LOCA occurs in the 24-hour CT is extremely small (in the order of 1E-7 or less). Furthermore, no compensatory or other measures are available to mitigate the risk associated with the loss of accumulator function. Due to the negligible risk increase associated with this scenario and the fact that there are no measures to take once a large-LOCA occurs, no "high risk" configurations are associated with this scenario.

In general, medium-LOCAs do not require accumulators if at least one HPI train is available. This means that if a medium-LOCA occurs when minimum accumulator functionality is unavailable and at the same time HPI is unavailable, the core will be damaged. However, the probability that a medium-LOCA occurs in the 24-hour CT and at the same time both trains of HPI are unavailable is extremely small (in the order of 1E-8 or less), because it is assumed that the plant is not operating at power with both HPI trains out-of-service. This assumption is based on current STS that limit operation at power with no HPI capability. Therefore, no Tier 2 restrictions beyond those currently in the STSs are deemed necessary.

Tier 3 calls for a program to identify "risk significant" configurations beyond those identified in Tier 2 resulting from maintenance or other operational activities and take appropriate compensatory measures to avoid such configurations. Because the accumulator sequence modeling is relatively independent of that for other systems, the Tier 2 analysis by itself is sufficient.

Furthermore, Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.65(a)(4) (Maintenance Rule) requires that licensees assess the risk any time maintenance is being considered on safety-related equipment. This requirement serves the objectives of Tier 3.

In summary, the Tier 2 evaluation did not identify the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration. The current TS provisions were found to be sufficient to address the Tier 2 issue. Because the accumulator sequence modeling is relatively independent of that for other systems and the implementation of the Maintenance Rule, the NRC staff concluded that application of Tier 3 to the proposed accumulator CT was not necessary.

The NRC staff finds that the proposed changes will allow safe operation with the changes in CT from 1 hour to 24 hours for Condition B of TS LCO 3.5.1. The NRC staff also finds that the proposed changes are consistent with the ICCDPs calculated in WCAP-15049 for the accumulator AOT increase and meet the criterion of 5E-07 in RGs 1.174 and 1.177. The analysis and acceptance provided in this SE, as demonstrated by WCAP-15049, covers all Westinghouse nuclear steam system supplier plants regardless of plant vintage and number of loops. The licensee confirmed the applicability of the analyses and the NRC staff's model SE in its application. The NRC staff, therefore, concludes that the proposed changes 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. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 29800; May 24, 2005). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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