September 16, 2004

Mr. A. Christopher Bakken, III
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: REQUEST FOR RELAXATION OF TECHNICAL SPECIFICATION REQUIREMENTS APPLICABLE DURING MOVEMENT OF IRRADIATED FUEL (TAC NOS. MB5710 AND MB5711)

Dear Mr. Bakken:

The Commission has issued the enclosed Amendment Nos. 263 and 245 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 29, 2002, as supplemented March 28, and May 1, 2003, and August 20, 2004.

These amendments revise the TS requirements for containment closure associated with the equipment hatch and personnel airlocks during CORE ALTERATIONS and movement of irradiated fuel within the containment. These changes are based on a revised analysis of the fuel handling accident (FHA) using selective implementation of alternate source term (AST) methodology. Such selective implementation of an AST methodology for a postulated FHA was previously approved by Amendment Nos. 251 and 232 for Salem Unit Nos. 1 and 2, respectively, by letter dated October 10, 2002.

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/

Daniel S. Collins, Senior 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. 263 to License No. DPR-70
2. Amendment No. 245 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

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PDI-2 Reading	JClifford	WBeckner	GHill(4)	DCollins
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ACCESSION NUMBER: ML042450476 *Se Input provided

OFFICE	PDI-2/PM	PDI-2/LA	DIPM/IROB	SPSB/SC	SPSB/SC*	SPLB/SC*	OGC	PDI-2/SC
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DATE	9/14/04	9/1/04	9/7/04	9/1/04	10/04/2002	06/02/2003	913/04	9/16/04

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Salem Nuclear Generating Station, Unit Nos. 1 and 2

cc:

Mr. Michael H. Brothers
Vice President - Site Operations
PSEG Nuclear - X15
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. John T. Carlin
Vice President - Nuclear Assessments
PSEG Nuclear - N10
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. David F. Garchow
Vice President - Eng/Tech Support
PSEG Nuclear - N28
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. Steven Mannon
Acting Manager - Licensing
PSEG Nuclear - N21
P.O. Box 236
Hancocks Bridge, NJ 08038

Jeffrie J. Keenan, Esquire
PSEG Nuclear - N21
P.O. Box 236
Hancocks Bridge, NJ 08038

Ms. R. A. Kankus
Joint Owner Affairs
PECO Energy Company
Nuclear Group Headquarters KSA1-E
200 Exelon Way
Kennett Square, PA 19348

Lower Alloways Creek Township
c/o Mary O. Henderson, Clerk
Municipal Building, P.O. Box 157
Hancocks Bridge, NJ 08038

Dr. Jill Lipoti, Asst. Director
Radiation Protection Programs
NJ Department of Environmental
Protection and Energy
CN 415
Trenton, NJ 08625-0415

Brian Beam
Board of Public Utilities
2 Gateway Center, Tenth Floor
Newark, NJ 07102

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector
Salem Nuclear Generating Station
U.S. Nuclear Regulatory Commission
Drawer 0509
Hancocks Bridge, NJ 08038

Mr. Carl J. Fricker
Plant Manager
PSEG Nuclear - N21
P.O. Box 236
Hancocks Bridge, NJ 08038

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. 263
License No. DPR-70

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

   A. The application for amendment filed by PSEG Nuclear, LLC and Exelon Generation Company, LLC (the licensee) dated July 29, 2002, as supplemented March 28, and May 1, 2003, and August 20, 2004, 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. 263, 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 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Richard J. Laufer for/

James W. Clifford, 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: September 16, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 263

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

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 Pages	Insert Pages
IX	IX
1-2	1-2
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 7-18	3/4 7-18
3/4 7-19	3/4 7-19
3/4 9-4	3/4 9-4
3/4 9-9	3/4 9-9
3/4 9-12	3/4 9-12
3/4 9-13	3/4 9-13
3/4 9-14	3/4 9-14
B 3/4 3-1a	B 3/4 3-1a
B 3/4 3-2	B 3/4 3-2
B 3/4 7-5c	B 3/4 7-5c
B 3/4 9-1c	B 3/4 9-1c
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
B 3/4 9-4	B 3/4 9-4

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. 245
License No. DPR-75

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

   A. The application for amendment filed by PSEG Nuclear LLC and Exelon Generation Company, LLC (the licensee) dated July 29, 2002, as supplemented March 28, and May 1, 2003, and August 20, 2004, 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. 245, 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 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Richard J. Laufer for/

James W. Clifford, 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: September 16, 2004

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 245</u>

<u>FACILITY OPERATING LICENSE NO. DPR-75</u>

<u>DOCKET NO. 50-311</u>

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 Pages	Insert Pages
IX	IX
1-2	1-2
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 7-15	3/4 7-15
3/4 7-16	3/4 7-16
3/4 9-4	3/4 9-4
3/4 9-10	3/4 9-10
3/4 9-13	3/4 9-13
3/4 9-14	Page Deleted
3/4 9-15	Page Deleted
B 3/4 3-1a	B 3/4 3-1a
B 3/4 3-2	B 3/4 3-2
B 3/4 7-5c	B 3/4 7-5c
B 3/4 9-1c	B 3/4 9-1c
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
B 3/4 9-4	B 3/4 9-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 263 AND 245 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. INTRODUCTION

By letter dated July 29, 2002, as supplemented March 28, and May 1, 2003, and August 20, 2004, PSEG Nuclear, LLC (PSEG or 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 revise the TS requirements for containment closure associated with the equipment hatch and personnel airlocks during CORE ALTERATIONS and movement of irradiated fuel within the containment. The March 28, and May 1, 2003, and August 20, 2004, letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration determination as published in the *Federal Register* on August 20, 2002 (67 FR 53989).

These changes are based on a revised analysis of a fuel handling accident (FHA) using selective implementation of alternate source term (AST) methodology as permitted by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident Source Term." Such selective implementation of the AST is defined in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents [DBAs] at Nuclear Power Reactors."

The specific proposed changes are as follows.

1.1 Revise the current TS Definition 1.9, "Core Alterations," for consistency with the standard technical specifications (STS), NUREG-1431. As editorial changes, this definition is being renumbered as 1.8 and the current Definition 1.9a is being renumbered as 1.9.

1.2 Revise TS Table 3.3-6, Item 2.a.1.a to delete operability and surveillance requirements (SRs) for the containment gaseous activity process monitor, purge and pressure vacuum relief isolation during Mode 6.

Revise TS Table 3.3-6, Item 2.a.2.a to delete operability and SRs for the containment particulate activity process monitor, purge and pressure relief isolation during Mode 6.

These changes delete requirements for the automatic isolation of the containment purge system during fuel movement within containment, such that in the event of an FHA, airflow will be into containment allowing continuous monitoring of the containment atmosphere until containment closure is accomplished.

Additionally, the proposed change would delete Action 22, which states “With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9."

1.3 Revise TS Table 4.3-3, Items 2.a.1.a and 2.a.2.a to reflect the corresponding changes made to Table 3.3-6. The change to Item 2.a.2.a also deletes the SR for the containment particulate activity process monitor in conformance with a previously approved change to Table 3.3-6, Item 2.a.2.a.

1.4 Revise TS 3.9.4, “Containment Building Penetrations,” to allow the containment equipment hatch and personnel air locks to be open during movement of irradiated fuel assemblies within containment, provided they are capable of being closed within one hour under administrative controls. A new SR will be added to verify this closure capability prior to the start of irradiated fuel movement within the containment. Corresponding changes to the bases were proposed.

As discussed in Section 1.6, the requirement to verify that each containment purge isolation valve actuates closed on a manual actuation signal once per 18 months would be added to TS 3/4.9.4 as SR 4.9.4.3

1.5 Revise TS 3/4.7.6, “Control Room Emergency Air Conditioning System,” to delete “CORE ALTERATIONS” from the applicability and actions.

1.6 Delete TS 3/4.3.9.9, “Refueling Operations, Containment Purge and Pressure-Vacuum Relief Isolation System.” Part of SR 4.9.9 will be relocated to TS 3/4.9.4 such that the verification that each containment purge isolation valve actuates closed on a manual actuation signal at least once-per-18-months would be retained. This proposed change will implement consistency with the improved technical specifications (ITSs).

1.7 Revise TS 3/4.3.9.12, “Fuel Handling Area Ventilation,” to delete the spent fuel pool (SFP) filtration system and its associated surveillances. Also, Action A is being revised to delete the requirement for suspending crane operation with loads over the storage pool.

1.8 Though not a change to the license, PSEG has proposed a number of changes to its procedures and other administrative controls. These changes are discussed in more detail below.

In summary, the proposed amendment would allow movement of sufficiently decayed irradiated fuel within the containment building with the equipment hatch, personnel air locks and containment penetrations open. Operation of the containment purge exhaust system (CPES) is not required during movement of sufficiently decayed fuel provided that the auxiliary building ventilation system (ABVS) is in operation and taking suction from the containment via the open containment airlocks.

2.0 REGULATORY EVALUATION

The construction permits for both Salem Unit Nos. 1 and 2 were issued by the Atomic Energy Commission (AEC) on September 25, 1968. The plant was designed and constructed based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as “draft GDC”). The AEC published the final rule that added Appendix A to 10 CFR Part 50, “General Design Criteria for Nuclear Power Plants,” in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as “final GDC”).

Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the Nuclear Regulatory Commission’s (NRC or the Commission) Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (Agencywide Document Access and Management System (ADAMS) Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and that the GDC were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission. Because the Salem construction permits were issued prior to May 21, 1971, the requirements applicable to the Salem facilities are those of the draft GDC.

The regulatory requirements that the NRC staff considered in its review of this amendment application included those contained in:

- 10 CFR 50.34, “Contents of applications; technical information”
- 10 CFR 100, “Reactor Site Criteria”
- 10 CFR 50.67, “Accident Source Term”
- 10 CFR 50.36, “Technical Specifications”
- Draft GDC 17, “Monitoring Radioactivity Releases”
- Draft GDC 18, “Monitoring Fuel and Waste Storage”
- Draft GDC 69, “Protection Against Radioactivity Release From Spent Fuel and Waste Storage”
- Draft GDC 70, “Control of Release of Radioactivity to the Environment”

- Final GDC 19[1], "Control Room"

The NRC staff also considered the regulatory guidance contained in:

- RG 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," dated July 2000

- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3

- Technical Specification Task Force Traveler (TSTF) 51, Revision 2.

- NRC Information Notice (IN) No. 90-77, "Inadvertent Removal of Fuel Assemblies from the Reactor Core"

<u>10 CFR 50.34, Part 100, and 10 CFR 50.67</u>

Applications for nuclear facility operating licenses are required, per 10 CFR 50.34, to include, in a final safety analysis report, all current information related to site evaluation factors identified in 10 CFR Part 100. That information must demonstrate that, in the event of an accident, radiation doses to persons onsite and offsite will continue to meet applicable acceptance criteria. Regulatory guidance for these evaluations is provided in the form of RGs and standard review plans. Fundamental to these evaluations is the source term -- the assumptions related to the radioactive material available for release to the environment. DBA analyses have traditionally used the source term provided in the 1962 document "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844.

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," which utilized this research to provide more physically-based estimates of the AST that could be applied to the design of future light-water power reactors. These revised source terms are described in terms of radionuclide composition and magnitude, physical and chemical form, and timing of release. In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST. The staff also issued regulatory guidance in using the AST in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

A licensee seeking to use an AST is required, pursuant to 10 CFR 50.67, to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the submittal. PSEG's evaluation supporting this amendment request was submitted in a separate amendment request regarding the minimum decay time required prior to fuel

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[1]Final GDC 19 contains requirements that apply to *all* holders of operating licenses who use an AST under 10 CFR 50.67.

movement. The NRC staff's evaluation of that submittal and its applicability to this amendment request is discussed further in section 3.0 of this Safety Evaluation (SE).

10 CFR 50.36, "Technical Specifications"

The NRC staff's evaluation of the acceptability of some of the proposed TS changes is based upon 10 CFR 50.36. Section 50.36(c)(2)(ii) of 10 CFR requires that a TS limiting condition for operation (LCO) of a nuclear reactor must 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.

A licensee seeking to delete a functional unit from the TS LCO must demonstrate that these criteria no longer apply to the functional unit to be deleted.

Section 50.36(c)(3) of 10 CFR defines "Surveillance Requirements" as 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.

Draft GDCs and Final GDC 19

Draft GDC 17, "Monitoring Radioactivity Releases"

Draft GDC 17 requires that means be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Draft GDC 18, "Monitoring Fuel and Waste Storage"

Draft GDC 18 requires that monitoring and alarm instrumentation be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Draft GDC 69, "Protection Against Radioactivity Release From Spent Fuel and Waste Storage"

Draft GDC 69 requires that containment of fuel and waste storage be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Draft GDC 70, "Control of Release of Radioactivity to the Environment"

Draft GDC 70 specifies that facility design shall include those means necessary to maintain control over gaseous, liquid, and solid radioactive plant radioactive effluents. It further specifies that appropriate holdup capacity be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified (a) on the basis of 10 CFR Part 20 requirements for normal operations and transient situations that might reasonably be anticipated to occur and (b) on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reductions of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Final GDC 19, "Control Room"

Final GDC 19 specifies, in part, that holders of operating licenses who use an AST under 10 CFR 50.67 shall provide a control room from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection must be provided to permit access to and occupancy of the control room under accident conditions to ensure that personnel receiving radiation exposure shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.

<u>Regulatory Guidance Documents</u>

RG 1.183 establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, Section 11.3.6.5, "Containment - Primary (PWR)/Secondary (BWR)" states the following:

> In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of TS requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays fairly rapidly. The basis of the TS operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases.

- A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

The licensee used TSTF 51 as a guide for developing their submittal. The licensee is not requesting any TS changes under the TSTF. The NRC approved TSTF-51, Revision 2 on October 15, 1999. TSTF-51 allows the removal of TS requirements for engineered safety features (ESF) to be OPERABLE after sufficient radioactive decay has occurred to ensure off-site doses remain below a small fraction of 10 CFR Part 100 limits. The NRC staff has allowed the use of TSTF-51 where the licensee is using the AST guidance if exclusion area boundary (EAB) and low population zone (LPZ) dose limits in 10 CFR 50.67 are not exceeded. Fuel that is not sufficiently decayed to allow relaxation of OPERABILITY requirements is referred to as “recently” irradiated fuel. Recently irradiated fuel could still be moved but the appropriate ESF systems need to be OPERABLE. TSTF-51 also allows the deletion of OPERABILITY requirements for ESF mitigation features during CORE ALTERATIONS.

NRC IN No. 90-77, “Inadvertent Removal of Fuel Assemblies from the Reactor Core,” discusses the potential for movement of reactor internal components, such as the upper guide structure, to result in inadvertent movement of fuel. That IN suggested that, “licensees may ... wish to consider the need to carefully inspect the upper core support structure as it is initially raised from the reactor vessel to ensure that no core components are suspended.”

3.0 TECHNICAL EVALUATION

By letter dated June 28, 2002, PSEG requested license amendments for Salem Unit Nos. 1 and 2 that proposed (1) revision of the TS requirements for minimum fuel decay time prior to movement of irradiated fuel; and, (2) a selective implementation of the AST to replace the accident source term used in the FHA radiological analyses pursuant to 10 CFR 50.67. The NRC approved that request and issued Amendment No. 251 to License No. DPR-070 for Salem Unit No. 1, and Amendment No. 232 to License No. DPR-075 for Salem Unit No. 2, by letter dated October 10, 2002. In support of that amendment request, PSEG performed radiological consequence analyses of a design-basis FHA. The assumptions and inputs used in those analyses encompass the plant configuration that will result from the TS changes proposed in PSEG’s July 29, 2002, application. As such, those analyses are relevant to the current amendment request.

3.1 FHA Radiological Consequence Analysis

Because the NRC staff previously approved selective implementation of an AST for the Salem FHA radiological analysis, the NRC staff relied upon its earlier review and finding of acceptability in approving this amendment. The NRC staff's complete evaluation of that selective implementation was issued by letter dated October 10, 2002, (ADAMS Accession No. ML022770181). However, because that review is relevant to this application, Sections 3.1.1, 3.1.2, and 3.1.3 below provide a summary of the portions of that SE regarding the radiological consequences of an FHA.

3.1.1 FHA Evaluation

The licensee evaluated the consequences of an event in which a spent fuel assembly is dropped during refueling, damaging all of the fuel rods in the assembly. This accident is postulated to occur inside the containment or in the fuel-handling building (FHB). The licensee considered two potential release points for the containment release and three potential release points for the FHB. The licensee reported the limiting case for an FHA in either the containment or the FHB.

The inventory of fission products in the reactor core is a function of the reactor power, the duration of the at-power operation, and the time after shutdown prior to spent fuel movement. PSEG determined the core inventory assuming a power level of 3600 megawatts thermal (MWt) (greater than 102% of the rated thermal power), an extended period of operation sufficient for significant radionuclides to reach equilibrium, and a decay period of 96 hours following shutdown. To account for differences in power distribution across the core, a peaking factor of 1.7 is applied to the average inventory. The majority of the fission products produced during operation are contained within the fuel pellet; however, some migrate to void spaces, known as "gaps," within the fuel rods. PSEG assumed that 8% of the I-131 inventory of the core was in the fuel rod gap, along with 10% of the Kr-85, 12% for alkali metals, and 5% of all other iodines and noble gases.

PSEG assumed that a single fuel assembly is dropped over the reactor vessel, or over the SFP. The Salem reactor cores contain 193 assemblies each. The radionuclides are assumed to be released from the damaged fuel rods, pass through the water in the reactor cavity or SFP, and enter the building atmosphere instantaneously. As the released gases rise through the overlaying water, halogens are scrubbed by the water column, resulting in an effective halogen decontamination factor of 200. No decontamination of noble gases or organic iodine forms was assumed. The fission products are assumed to be released to the environment over a 2-hour period via the open containment equipment hatch (CEH), personnel air locks, and other penetrations. The CEH provides a direct release path to the environment. Releases via the other paths are collected and released via the plant vent (PV). Since PSEG assumes a 100% release over 2 hours and has taken no credit for engineered safeguards features for isolation or filtration of releases to the environment, the only parameter that differentiates the release points in the containment and FHB cases is the atmospheric dispersion. To simplify the calculational effort, PSEG selected the limiting χ/Q values for each case and performed the dose analysis once. The releases in both cases were treated as ground level releases for determining the EAB, LPZ, and control room doses. PSEG assumed a release rate based on the release of 99% of the radionuclides in the containment or FHB to the environment over a 2-hour period.

The licensee evaluated the dose to operators in the control room assuming that the control room ventilation system automatically realigns within 2 minutes into an emergency configuration and assuming an unfiltered inleakage flow rate of 4000 cfm. The NRC staff found that the analyses are sufficiently conservative to provide adequate assurance that the radiation doses to the control room personnel will not impede response actions necessary to protect the public. Furthermore, the doses estimated by the licensee for the postulated FHA was found to be acceptable.

3.1.2 Atmospheric Relative Concentration Estimates

PSEG calculated new χ/Q estimates for the CEH at both units for both control room intakes and calculated new values for the Salem Unit No. 1 PV and the Salem Unit No. 1 FHB roll-up door for both control room intakes. These values bound the values for the Salem Unit No. 2 PV and the Salem Unit No. 2 FHB roll-up door. PSEG used the guidance of DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," in developing these values. PSEG did not credit the ability to preferentially select a control room intake even though that capability is available.

All of the values were determined as ground level releases using meteorological data collected for the years 1988 through 1994, using the NRC-sponsored ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). The meteorological data were obtained from the site meteorological tower that services both Salem and Hope Creek Generating Station. The NRC staff performed a series of statistical checks on the meteorological data to evaluate its suitability, and compared the ARCON96 code inputs used by PSEG to the site release point and intake configuration. The inputs were found to be acceptable. Additionally, the NRC staff found the χ/Q values used by PSEG to be acceptable.

3.1.3 NRC Staff's Conclusion

The NRC staff reviewed the AST implementation proposed by PSEG for Salem. The staff found that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183. The staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff found, with reasonable assurance, that the licensee's estimates of the TEDE due to FHA accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. Although the staff did confirmatory analyses, the staff's approval of that amendment was based on the information docketed by the licensee as well as on the staff's finding that the methods, inputs, and assumptions used in the licensee's analyses are acceptable.

By letter dated March 28, 2003, PSEG notified the NRC that they had identified an error in the radiological dose consequences for the EAB in their calculations associated with the previously issued Amendment Nos. 251 and 232 for Salem Unit Nos. 1 and 2, respectively. The nature of the error was such that the EAB dose reported in the application was a factor of 10 higher than the corrected calculation. Because the corrected dose was lower than the estimated doses that the NRC staff had reviewed, and well within the regulatory limits of 10 CFR 50.67, the error does not invalidate the results of the NRC staff's previous review.

3.2 Proposed Administrative Controls

As recommended in NUMARC 93-01, PSEG proposed administrative controls that will be implemented as contingency methods to promptly establish containment closure in the event of an FHA. In its July 29, 2002, submittal, PSEG proposed the following:

Containment Building Closure:

The following requirements shall be maintained to ensure defense-in-depth. Closure Controls are in effect whenever the affected Containment is open during operations within containment involving movement of irradiated fuel assemblies. The definition of an open containment penetration is a penetration that provides direct access from the containment atmosphere to the outside environment.

1. *The equipment necessary to implement containment closure shall be appropriately staged prior to maintaining any containment penetration open including airlock doors and the containment equipment hatch.*

2. *Hoses and cables running through any open penetration, airlock, or equipment hatch should be configured to facilitate rapid removal in the event that containment closure is required.*

3. *The containment personnel airlock may be open provided the following conditions exist:*

    a. *One door in each airlock is capable of being closed*
    b. *Hoses and cables running through the airlock shall employ a means to allow safe, quick disconnection or severance.*
    c. *The airlock door is not blocked in such a way that it cannot be expeditiously closed. Protected covers used to protect the seals/airlock doors or devices to keep the door open/supported do not violate this provision.*
    d. *Personnel are designated and available with the responsibility for expeditions closure (within 1 hour) of at least one door on the containment airlocks following the FHA.*

4. *The containment equipment hatch may be open provided the following conditions exist:*
    a. *The containment equipment hatch is capable of being closed or an equivalent closure device is available and can be closed within 1 hour.*
    b. *Hoses and cables running through the equipment hatch shall employ a means to allow safe, quick disconnection or severance.*
    c. *The equipment hatch is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/equipment hatch or devices to keep the hatch open supported do not violate this provision.*
    d. *Necessary tools to install the equipment hatch and tighten at least four equipment hatch closure bolts are available or other methods*

to close the equipment hatch opening (i.e., restrict air flow out of the containment), such as a refueling hatch closure device, is staged at the work area along with the necessary installation tools.

e. A sufficient number of personnel are designated and available with the responsibility for expeditious closure (within 1 hour) of the containment equipment hatch opening following the FHA.

5. If containment closure would be hampered by an outage activity, compensatory actions will be developed.

6. Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished, or close the equipment hatch (or an equivalent closure device is installed) and personnel airlocks. Periodic verification (once per shift) of this administrative control will ensure that air flow will be directed from containment to the Auxiliary Building or the Plant Vent where continuous monitoring will be in effect thus minimizing the potential for unmonitored releases out the open containment hatch following the FHA.

7. Personnel responsible for Containment Building Closure shall be trained and knowledgeable in using the procedure for executing containment closure. Walkdowns should be considered to demonstrate the closure capability including compensatory actions in the event of loss of electrical power.

<u>Fuel Handling Building Closure</u>:

The following requirements shall be maintained to ensure defense-in-depth. Closure Controls are in effect during operations within the Fuel Handling Building involving movement of irradiated fuel assemblies.

1. The Fuel Handling Building doors shall be maintained closed except for normal entry and exit unless a designated person is available to close the open door(s) should a[n] *FHA occur within the Fuel Handling Building.*

2. The FHAVS [fuel handling area ventilation system]*, with associated radiation release monitoring will be available for the release flow path. If for any reason operation of the fuel handling area ventilation system flow path must be discontinued and the fuel building is open to the outside environment, fuel movement within the openings to the outside environment are closed.*

3. If the Fuel Handling Building closure would be hampered by an outage activity, compensatory actions will be developed.

Control Room Emergency Air Conditioning System (CREACS)

During movement of irradiated fuel assemblies, both CREACS normal outside air intakes should normally be open. If one intake is closed, movement of irradiated fuel assemblies will be suspended until the intake is reopened. These controls are governed by existing action requirements under Technical Specification 3.7.6.1. The actuation of CREACS during a[n] *FHA is performed by the radiation monitors located in the normal outside air intakes. Exceeding the setpoints of these radiation monitors will cause dampers to reposition to isolate the normal ventilation system from the Control Room Envelope, start the CREACS fans and open the appropriate outside emergency air intake. The radiation monitors in the Control Room normal outside intake are required to be OPERABLE during movement of irradiated fuel assemblies as governed by TS Table 3.3-6.*
[Italics added to denote quoted material.]

These controls would be implemented in plant procedures controlled in accordance with 10 CFR 50.59. Additionally, PSEG proposed that actions to develop and implement these administrative controls be established as a regulatory commitment associated with implementation of these amendments.

In requests for additional information (RAIs) dated March 18, 2003, and July 16, 2004, the NRC staff asked PSEG for several clarifications regarding these controls. The following discussion addresses NRC RAI questions specific to these proposed controls; questions regarding other aspects of the amendment request are addressed elsewhere in this SE.

1. The NRC staff asked PSEG to describe analyses performed to verify the ability of the CPES and the ABVS to draw a negative pressure in containment (March 18, 2003, RAI question 2).

   PSEG responded that the ability to draw a negative pressure on containment has been demonstrated by previous operating experience. PSEG provided anecdotal information from an occurrence during the Salem Unit No. 2, 12th refueling outage in which the ABVS was used to reduce high airborne radioactivity levels inside containment.

2. The NRC staff asked PSEG to describe any expected outage activities that could prevent establishment of containment closure and compensatory actions that would be taken (March 18, 2003, RAI question 4).

   PSEG responded that there are no “expected” outage activities that would prevent establishment of containment closure. The provisions in the administrative controls regarding outage activities that might hamper containment closure, in conjunction with the newly added SR 4.9.4.2, are intended to address any off-normal activities during defueling and ensure: (1) that any needed compensatory actions are identified, and (2) the capability to establish closure within 1 hour.

3. The NRC staff asked PSEG to describe any expected outage activities that could prevent establishment of FHB closure and compensatory actions that would be taken (March 18, 2003, RAI question 5, and July 16, 2004, RAI question 3)

   PSEG responded that they intend to maintain the FHB door closed during fuel movement except normal entry/exit and off-normal or plant modification conditions. In instances where the door is maintained open, there will be a designated person to close the door in the event of an FHA in the fuel building. Additionally, in instances where the door is maintained open, fuel handling would either be terminated or compensatory measures (e.g., use of a temporary atmospheric pressure ventilation barrier) would be established in accordance with the proposed administrative controls.

4. The NRC staff asked PSEG to define what criteria are used to determine whether a device is an equivalent closure device to the equipment hatch (July 16, 2004, RAI question 2).

   In its response, PSEG indicated that an equivalent closure device must be sufficient to provide an atmospheric ventilation barrier to restrict radioactive material released from an FHA. Additionally, the design and fabrication of the equivalent device will be governed by ASME Boiler and Pressure Vessel Code, Section III for Class B Vessels 1968. The licensee's response also noted that use of an equivalent closure device that meets these criteria is already specifically allowed by the Salem Unit Nos. 1 and 2 TSs. That allowance was incorporated into the Salem TSs by Amendment Nos. 217 and 199 for Salem Unit Nos. 1 and 2, respectively and, therefore, is already part of the Salem current licensing basis.

5. The NRC staff asked PSEG to explain what measures are in place to close the equipment hatch in the event of a loss of alternating current (AC) power, and the number of closure bolts required to be installed to prevent release of radioactivity (July 16, 2004, RAI question 4).

   The licensee responded that the refueling equipment is powered from offsite power and fails safe on a loss of power. Therefore, the licensee asserts, loss of all AC power coincident with an FHA is unlikely. Additionally, PSEG noted that the current licensing basis FHA analysis does not credit containment closure and the dose limitations of 10 CFR 50.67 are still met. In response to another question, PSEG noted that they have already demonstrated their ability to establish closure within 1 hour using the outage equipment hatch, which is a temporary closure devise that contains a hinged door that meets the criteria of an "equivalent closure device." PSEG provided portions of an existing plant procedure that indicates that the hinged door of the outage equipment hatch is manually operated.

   Finally, the licensee stated that methods for closing the containment equipment hatch in the event of a loss of AC power will be developed as part of implementing this amendment or the containment equipment hatch will remain closed during movement of irradiated fuel.

6. The NRC staff asked PSEG to explain why the proposed controls contain no provisions to isolate the CPES or ABVS after an FHA (July 16, 2004, RAI question 5).

PSEG responded, in its May 1, 2003, and August 20, 2004, letters, that the provisions to maintain the ventilation lineup through either the ABVS or CPES will ensure that any releases to the environment are monitored in order to comply with the requirements of the draft GDC.

7. The NRC staff asked PSEG to provide the criterion used to decide if the containment personnel airlock and the containment equipment hatch are capable of being closed within 1 hour (July 16, 2004, RAI question 6).

   The licensee responded that the proposed administrative controls provide the criteria that must be met. These include:

   - airlock doors or equipment hatch opening is not blocked
   - cables or hoses running through the airlock or equipment hatch opening contain isolation valves and/or quick-disconnect fitting.
   - needed tools to remove cables/hoses and establish closure are pre-staged
   - designated closure team is available and team members do not have assigned duties that would interfere with them responding immediately.

   The licensee also noted that such administrative controls have already been implemented and tested for establishing timely closure during mid-loop operations. The licensee provided a partial copy of their plant procedure, SC.MD-FR.CAN-0001(Q), "Outage Equipment Hatch Installation, Removal, Seal Replacement and Door Manipulation for Containment Closure" where these controls have been incorporated.

8. In view of the fact that the licensee would use "designated" rather than a "dedicated" crew to close the containment equipment hatch in the event of an FHA, the NRC staff asked PSEG to discuss other duties that the designated crews will have and where they will be stationed relative to the airlock doors (July 16, 2004, RAI question 10).

   PSEG's response stated that use of a designated crew to ensure the ability to establish timely containment closure has already been incorporated into plant procedures for mid-loop operations and has demonstrated the ability to establish closure within 1 hour.

The NRC staff has reviewed the proposed administrative controls and finds that they are adequate to ensure the ability to establish containment closure in a timely manner in the unlikely event of an FHA. The use of equivalent closure devices and the minimum four-bolt closure requirement on the containment equipment hatch are already part of the Salem licensing bases and are not being changed by these amendments.

3.3 Technical Specification Changes

3.3.1 TS Definitions 1.8, 1.9, 1.9a

Proposed Change

PSEG proposed revision of the current TS Definition 1.9, "Core Alterations," as well as renumbering this definition to 1.8; the current Definition 1.9a is also being renumbered as 1.9.

> The current definition of CORE ALTERATION :
>
> "...movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position."
>
> would be revised to read:
>
> "...movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position."

This proposed revision would result in a definition of CORE ALTERATIONS that is consistent with NUREG-1431. Associated with this change, the term "CORE ALTERATIONS" would be deleted from the applicability and actions for TS 3/4.7.6, "Control Room Emergency Air Conditioning System, and TS 3.9.4, "Containment Building Penetrations." These associated changes to TSs 3/4.7.6 and 3.9.4 are discussed in Sections 3.3.5 and 3.3.4.5, respectively.

Evaluation

TSTF-51, Revision 2 identifies inadvertent criticality, FHAs, and loading of a fuel element in the wrong location as postulated accidents that could occur during core alterations. PSEG states that, of these accidents, the FHA is most limiting. The proposed definition is consistent with the FHA being the limiting event.

Noting the industry experience discussed in IN 90-77, in an RAI dated July 16, 2004, the NRC staff asked why the proposed TS change should not include applicability for an FHA occurring during movement of core components other than fuel, sources, or reactivity control components. In its response, dated August 20, 2004, PSEG clarified that the current licensing basis FHA analysis does not specify the means by which the hypothetical FHA occurs. In other words, it is independent of whether the postulated fuel bundle falls from the refueling equipment or from movement of another core component. Furthermore, the estimated radiological consequences of an FHA - which assumes a 100% release over 2 hours and no credit for engineered safeguards features for isolation or filtration of releases to the environment - are within the regulatory limits specified in 10 CFR 50.67. Therefore, the potential for an inadvertent removal of fuel assemblies from the reactor core leading to an FHA is bounded by the current licensing basis FHA analysis.

In spite of the fact that the radiological consequences of an inadvertent removal of fuel assemblies from the reactor core leading to an FHA are addressed by the analysis, PSEG has instituted procedural controls to guard against this scenario. In its August 20, 2004, RAI response, PSEG provided a copy of the applicable portion of its procedure SC.MD-FR.FH-0011(Q) that requires personnel to perform a video inspection of the upper internals assembly when it is 12-14 inches clear of the reactor vessel to ensure that no core components (such as fuel assemblies or fuel inserts) are attached. The procedure specifies immediate actions to be taken in the event that core components are inadvertently moved with the upper internals.

Because the proposed change is bounded by the current licensing basis FHA analysis and the licensee has implemented appropriate procedural controls to protect against an inadvertent removal of fuel assemblies from the reactor core, the NRC staff finds the proposed change to the definition of CORE ALTERATION to be acceptable. It should be noted that the proposed change is consistent with NUREG-1431.

The proposed renumbering of the definitions is administrative and does not alter the content of any TS requirements. Therefore, the proposed renumbering is acceptable.

3.3.2 TS Table 3.3-6

PSEG proposed changes to:

a) Revise TS Table 3.3-6, Item 2.a.1.a to delete operability and SRs for the containment gaseous activity process monitor during Mode 6.

b) Revise TS Table 3.3-6, Item 2.a.2.a to delete operability and SRs for the containment particulate activity process monitor during Mode 6.

TS Table 3.3-6, Item 2.a.1.a, “Containment Gaseous Activity, Purge and Pressure Vacuum Relief Isolation,” (page 3/4 3-36) requires a minimum of one containment purge valve isolation channel to be OPERABLE during plant operation in Mode 6. This signal is generated on increasing airborne gaseous radioactivity levels within containment.

TS Table 3.3-6, Item 2.a.2.a, “Containment Air Particulate Activity, Purge and Pressure Vacuum Relief Isolation,” (page 3/4 3-36) requires a minimum of one containment purge valve isolation channel to be OPERABLE during plant operation in Mode 6. This signal is generated on increasing airborne particulate radioactivity levels within containment.

These changes delete requirements for the automatic isolation of the containment purge system during fuel movement within containment. This change is being sought to maintain outside airflow into containment via open hatches and penetrations and allow the continuous monitoring of the containment atmosphere until containment closure is established following an FHA. An automatic isolation of the purge system could result in unmonitored releases via the open equipment hatch.

The FHA in containment event was analyzed with the assumption that 99% of the radionuclides released from the damaged fuel assembly in containment would be released over a 2-hour period without automatic isolation of the containment purge system. The dose analysis was performed using the more limiting χ/Q value for the plant vent or the containment equipment hatch. Even with CPES unisolated, no regulatory limits are expected to be exceeded in the event of an FHA. As such, the analysis bounds the proposed configuration. Therefore, the change is acceptable from an accident radiological consequence perspective.

The proposed changes to Table 3.3-6, Items 2a1a, “Containment Gaseous Activity Purge and Pressure/Vacuum Relief Isolation,” and 2a2a, “Containment Air Particulate Activity Purge and Pressure/Vacuum Relief Isolation,” will allow the ventilation lineups to maintain a negative pressure inside containment. This will not only allow continuous monitoring of post-FHA containment activity until containment closure is established, but will reduce the radiological release to the environment. The capability to manually isolate the CPES is unaffected.

The NRC staff notes that the containment purge and purge isolation signal is not a form of instrument or a process variable, design feature or operational restriction that is an initial condition of a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; nor is it a structure, system or component that is part of a primary success path. Therefore, Criterion 1, 2 and 3 of 10 CFR 50.36(c)(2)(ii) do not apply.

Items 2.a.1.a and 2.a.2.a., which provide the operability criteria for the instrumentation that automatically actuates the containment purge isolation valves, have been shown not to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. The subject instrumentation is no longer credited to ensure that the radiological dose criteria are met for the EAB, LPZ, and control room. Thus, the operability of the instrumentation is not risk significant; therefore, Criterion 4 of 10 CFR 50.36(c)(2)(ii) does not apply.

Additionally, the proposed change would delete Action 22, which states “With the number of channels OPERABLE less than required by the minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9." The provisions of Action 22 were only applicable to those LCOs whose removal was discussed above, and are no longer necessary.

Based on the above, the NRC staff considers that the proposed deletion of TS Table 3.3-6, Items 2.a.1.a, 2.a.2.a, and Action 22 is acceptable.

3.3.3 TS Table 4.3-3

PSEG proposed changes to TS Table 4.3-3, Items 2.a.1.a and 2.a.2.a to reflect the corresponding changes made to Table 3.3-6. The change to Item 2.a.2.a also deletes the SR for the containment particulate activity process monitor in conformance to a previously approved change to Table 3.3-6, Item 2.a.2.a.

TS Table 4.3-3 specifies requirements for radiation monitoring instrumentation. Items 2.a.1.a and 2.a.2.a correspond to TS Table 3.3-6, Items 2.a.1.a and 2.a.2.a discussed above. The change to Item 2.a.1.a deletes the SR for the air particulate activity radiation monitor in Mode 6. Since the instrumentation is no longer required to be in the TS LCO, as discussed above, the associated Mode 6 SR is also no longer required. Section 50.36(c)(3) of 10 CFR no longer applies and this SR may be removed.

Regarding the proposed change to TS Table 4.3-3, Item 2.a.2.a, removal of the requirements for this SR in Modes 1, 2, 3 and 4 was previously approved by Amendment Nos. 79 and 53 for Salem Unit Nos. 1 and 2, respectively. Those amendments removed the TS LCO requirements for the containment air particulate monitor to be operable in Modes 1, 2, 3, and 4. However, due to an oversight, corresponding changes to the associated SRs were not made. Since the LCO requirements for Modes 1, 2, 3 and 4 no longer apply, it is acceptable to delete the associated SRs. As discussed above for Item 2.a.1.a, since the instrumentation is no longer required to be included in a TS LCO for Mode 6, it is acceptable to delete the Mode 6 SR.

3.3.4 TS 3.9.4

PSEG proposed revision of TS 3.9.4, “Containment Building Penetrations,” to allow the containment equipment hatch and personnel air locks to be open during movement of irradiated fuel assemblies within containment provided they are capable of being closed within 1 hour under administrative controls. A new SR will be added to verify this closure capability prior to movement of irradiated fuel within the containment. Corresponding changes to the bases were proposed.

The FHA in containment event was analyzed with the assumption that 99% of the radionuclides released from the damaged fuel assembly in containment would be released over a 2-hour period without automatic isolation of the containment purge system. The dose analysis was performed using the more limiting χ/Q value for the plant vent or the containment equipment hatch. As such, the analysis bounds the proposed configuration.

The licensee proposes the following changes.

3.3.4.1 LCO 3.9.4.a

Change LCO 3.9.4.a to read : “The equipment hatch inside door is capable of being closed and held in place by a minimum of four bolts, or an equivalent closure device installed and capable of being closed.”

The FHA accident analysis takes no credit for containment closure. Even with the equipment hatch open, no regulatory limits are expected to be exceeded in the event of an FHA. Therefore, the requirement to have the equipment hatch inside door or an equivalent closure device closed no longer meets the criterion in 10 CFR 50.36. Although closing the equipment hatch is not necessary to meet the requirements of 10 CFR 50.67, and is not required by 10 CFR 50.36, the NRC staff has determined that this measure is an important element of defense-in-depth that serves to manage the consequences of an FHA, further reducing the release. Commensurate with the revision of the LCO, the phrase “capable of being closed” is inserted into the proposed SR 4.9.4.2 which specifies an installation time of within 1 hour after the FHA. Therefore, the proposed change is acceptable.

3.3.4.2 LCO 3.9.4.b

Change LCO 3.9.4.b to read: “A minimum of one door in each airlock is capable of being closed.”

The FHA accident analysis takes no credit for containment closure. Even with the personnel airlocks open, no regulatory limits are expected to be exceeded in the event of an FHA. Therefore, the requirement to have a minimum of one door in each airlock closed no longer meets the criterion in 10 CFR 50.36. Although closing the personnel airlock is not necessary to meet the requirements of 10 CFR 50.67, and is not required by 10 CFR 50.36, the NRC staff has determined that this measure is an important element of defense-in-depth that serves to manage the consequences of an FHA, further reducing the release. Commensurate with the revision of the LCO, the phrase “capable of being closed” is tied to administrative controls which specify a closure time of within 1 hour after the FHA. Therefore, the proposed change is acceptable.

3.3.4.3 LCO 3.9.4.c

Change LCO 3.9.4.c to read: “Each penetration providing direct access for the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by the Containment Purge and Pressure-Vacuum Relief Isolation System.”

The FHA accident analysis takes no credit for containment closure. Even with the containment penetrations open, no regulatory limits are expected to be exceeded in the event of an FHA. Although not necessary to meet the requirements of 10 CFR 50.67, and not required by 10 CFR 50.36, the NRC staff has determined that isolating the penetrations is an important element of defense-in-depth that serves to manage the consequences of an FHA, further reducing the release. Commensurate with the revision of the LCO, the phrase “capable of being closed” is tied to administrative controls which specifies a closure time of within 1 hour after the FHA. Therefore, the proposed change is acceptable.

3.3.4.4 LCO Note

Add the following Note to the LCO:

> “Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.”

The licensee committed to develop administrative controls to ensure that containment closure is accomplished within 1 hour following an FHA within containment even though the containment fission product control function is not required to meet acceptable dose consequences. Closing penetration flow path(s) is a consequence management action that further reduces the release due to the FHA and supports defense-in-depth. The administrative controls do not need to be in the TS Bases because, as the licensee states, the changes fall under the requirements of 10 CFR 50.59. Based on the licensee’s commitment to ensure containment closure in a timely fashion, unisolating penetration flow path(s) under administrative controls is acceptable.

3.3.4.5 Delete Core Alternations Applicability

The staff’s position is that the FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. This position is documented in TSTF-51. The proposed change to the applicability statement leaves the LCO and required actions applicable during activities which could result in an FHA with fuel damage and radiological release. Therefore, the deletion of CORE ALTERATIONS is acceptable.

3.3.4.6 <u>SR 4.9.4.1</u>

Change SR 4.9.4.1 to read:

> "Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by a <u>manual or</u> automatic containment isolation valve at least once per seven days."

The requirement to verify the status of containment penetrations within 100 hours prior to moving irradiated fuel will be deleted by this change. The provisions of SR 4.0.4 do not allow entry into an LCO unless all SRs are current. Thus, due to the retained weekly verification of isolation capability, the requirement to verify that all penetrations are closed or capable of being closed will be verified within 168 hours (i.e., once per seven days) prior to moving irradiated fuel. The staff considers 168 hours to be a reasonable period that corresponds to the longest decay time required before fuel movement per TS 3/4.9.3.

The addition of the phrase "manual or" in the SR makes the proposed SR consistent with the previously discussed revision to LCO 3.9.4.c.1. Therefore, the proposed revision to the SR is acceptable.

3.3.4.7 <u>SR 4.9.4.2</u>

Add SR 4.9.4.2 which reads:

> "<u>Once per refueling prior to the start of movement of irradiated fuel assemblies within the containment building, verify the capability to install, within 1 hour, the equipment hatch. Applicable only when the equipment hatch is open during the movement of irradiated fuel in the containment building.</u>"

This new SR sets the requirement to verify that during the movement of irradiated fuel that the equipment hatch can be installed within 1 hour in the event of an FHA. This verification provides the requisite assurance that the equipment hatch is able to be closed in the event of an FHA. Although closing the equipment hatch is not necessary to meet the requirements of 10 CFR 50.67, the NRC staff has determined that these measures are an important element of defense-in-depth that serves to manage the consequences of an FHA, further reducing the release. Therefore, the proposed change is acceptable.

The licensee states that the revised FHA analyses assumes that all of the radioactive material which could be released to the containment atmosphere exits the containment within 2 hours of accident initiation with no credit taken for the containment boundary closure. The licensee proceeds to state that, consistent with the philosophy of minimizing dose released to the environment, administrative controls will be established to ensure that the equipment access hatch, and other containment penetrations which provide direct access to the outside atmosphere, can be closed within 1 hour of accident initiation as a defense-in-depth measure to minimize the consequences of an FHA.

The containment penetrations being open during refueling is partially compensated by the licensee implementing administrative controls to close the containment in 1 hour using designated personnel after an FHA. As discussed above, the NRC staff determined that these administrative controls provide an important element of defense-in-depth, and with these administrative controls in place, it will assure that the licensee will manage the consequences of an FHA in a manner that will afford adequate protection to the public.

The licensee also states that the containment atmosphere is monitored during normal and transient operations of the reactor plant by the radiation monitors on the main PV or on the containment purge line. The NRC staff agrees that the use of existing radiation monitors on the main PV or on the containment purge line would provide sufficient monitoring to comply with the provisions of Draft GDC 17.

The staff also considered the implications of the proposed change on draft GDC 69 and 70, which require appropriate containment, confinement, and filtering of radioactive contaminants in areas where fuel is stored. The NRC staff considers the licensee's commitment to develop administrative controls that close the equipment hatch, terminate the purge, and isolate the containment to satisfy the requirements of draft GDC 69 and 70, and these controls will minimize any potential release to the public.

3.3.4.8 SR 4.9.4.3

Add SR 4.9.4.3 which reads:

> "Verify, once per 18 months, each required containment purge isolation valve actuates to the isolation position on a manual actuation signal."

The proposed SR 4.9.4.3 replaces the deleted SR 4.9.9 (see next section). In proposed SR 4.9.4.3, for open purge and exhaust penetrations, the periodic frequency for verifying the automatic closure capability is relaxed from once per 7 days to once per 18 months. The staff has previously accepted the relaxation of the periodic frequency from 7 days to 18 months as it maintains consistency with other engineered safety features actuation system instrumentation and valve testing TS requirements. This relaxation is reasonable given the assurances of the closure capability of any open purge and exhaust valves provided by SR 4.9.4.1 with its 7-day frequency. Therefore, the proposed TS 4.9.4.3 is acceptable.

The proposed SR 4.9.4.3 also removes the requirement in SR 4.9.9 to verify that containment purge and pressure-vacuum relief isolation system occurs on a high radiation test signal from each of the containment radiation monitoring instrumentation channels. This change is consistent with the changes in TS 3.3.6 discussed above that eliminate the automatic actuation of the containment purge and pressure-vacuum relief isolation system signal.

Based on the above evaluation, the NRC staff considers that the proposed changes to TS Section 3/4.9.4 are acceptable.

3.3.5 TS 3/4.7.6

Revise TS 3/4.7.6, "Control Room Emergency Air Conditioning System," to delete "CORE ALTERATIONS" from the applicability and actions.

The licensee proposes to delete applicability during CORE ALTERATIONS. The LCO remains applicable during Modes 1-4 and during movement of irradiated fuel assemblies.

The NRC staff's position is that the FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. This position is documented in TSTF-51. Under the revised FHA analyses, the potential for a radioactive release only exists during the movement of fuel within the containment or the SFP. The proposed change to the applicability statement leaves the LCO and required actions applicable during activities which could result in an FHA with fuel damage and radiological release. Therefore, the deletion of CORE ALTERATIONS is acceptable.

3.3.6 <u>TS 3/4.9.9</u>

Delete TS 3/4.9.9, "Refueling Operations, Containment Purge and Pressure-Vacuum Relief Isolation System," into TS 3.9.4, "Containment Building Penetrations." This proposed change will implement consistency with the ITSs. Part of SR 4.9.9 will be relocated to TS 3/4.9.4 such that the verification that each containment purge isolation valve actuates closed on a manual actuation signal at least once-per-18-months would be retained.

TS 3/4.9.9 "Containment Purge and Exhaust Isolation System," the entire section, would be deleted and the page is marked "intentionally left blank."

Removal of automatic isolation of the purge system is partially compensated by adding a footnote to TS 3.9.4, discussed below, which requires administrative controls to close the containment in 30 minutes using designated personnel after an FHA. The staff has determined that these administrative controls provide an important element of defense-in-depth, and with these administrative controls in place, it will assure that the licensee will manage the consequences of an FHA in a manner that will afford adequate protection to the public. As such, the staff finds that the removal of automatic isolation is acceptable with the addition of administrative controls to effect closure. The staff reviewed the request to delete TS 3/4.9.9 and agrees, as further explained below, that the TS section may be deleted since the purge isolation system is not credited in the DBA analysis.

The NRC staff notes that the containment purge and exhaust isolation system (CPEIS) is not a form of instrument or a process variable, design feature or operational restriction that is an initial condition of a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. Therefore, Criterion 1, and 2 of 10 CFR 50.36(c)(2)(ii) do not apply.

The licensee has shown, on the basis of their FHA design-basis analysis that operation of the CPEIS is not required to satisfy the dose values of 10 CFR 50.67. Thus, the system is not on the primary success path for a DBA. As such, Criterion 3 of 10 CFR 50.36(c)(2)(ii) does not apply.

Since the purge isolation system is not credited in the DBA analysis, it is not considered to be risk-significant to public health and safety by either operating experience or probabilistic safety assessment; therefore, Criterion 4 of 10 CFR 50.36(c)(2)(ii) does not apply. The staff finds that the purge isolation system does not meet the criteria contained in 10 CFR 50.36(c)(2)(ii) and its removal is, therefore, acceptable.

3.3.7 TS 3/4.3.9.12

Revise TS 3/4.3.9.12, "Fuel Handling Ventilation," to delete the SFP filtration system surveillances. Also, Action A is being revised to delete the requirement for suspending crane operation with loads over the storage pool.

The licensee has shown, on the basis of their FHA design-basis analysis that the fuel building exhaust filter system is not required to satisfy the dose values of 10 CFR 50.67 nor is it instrumentation used to detect and indicate a significant degradation of the reactor coolant pressure boundary. Thus, the system is not a part of the primary success path for a DBA. As such, Criterions 1 and 3 of 10 CFR 50.36 are not applicable. The fuel-handling DBA does have an assumption as to the time at which the accident occurs. This is because the licensee is restricted from moving the fuel before the requirements of the decay time specification TS 3/4.9.3, which limits the movement of irradiated fuel until the reactor has been subcritical for a pre-defined time period[2]. Since movement of fuel prior to this time is restricted by the TS, the inclusion of an LCO to satisfy the requirements of Criterion 2 of 10 CFR 50.36, which requires an LCO for process variables upon which a design-basis analysis depends, is not required. The CPEIS is not a form of instrument or a process variable, design feature, or operational restriction that is an initial condition of a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. The system is no longer credited to ensure that the radiological dose criteria are met for the EAB, LPZ, and control room. Thus, the operability of the system is not risk significant and Criterion 4 of 10 CFR 50.36(c)(2)(ii) is not applicable. Given that the four criterion are not applicable, removal from the TSs is acceptable.

The staff's finding of acceptability for this item is only associated with the removal of the item from the TS. Prior to removing any equipment or changing any procedure affecting the operation of engineering safeguards equipment, the licensee must use the appropriate modification process (10 CFR 50.59 or 10 CFR 50.90). The licensee has stated that "procedural guidance will be available for closing fuel building area atmosphere boundary penetrations if a[n] FHA occurs inside the fuel building." The use of this procedural guidance will be implemented "as a defense-in-depth measure to minimize actual releases to the outside atmosphere much lower than assumed in the AST FHA analyses dose calculations." The staff concurs that the development and implementation of procedural guidance will increase defense-in-depth and facilitate managing releases during an FHA. As such, it will provide additional assurance of protection to public health and safety and, therefore, the NRC staff concurs that the licensee may remove the section from their TS as the licensee has proposed.

3.4 Site Emergency Plan Considerations

As part of its review, the NRC staff considered the implications that the requested amendment might have on the site emergency plan. In its July 16, 2004, RAI, the NRC staff asked what criteria will be used to determine if closure is required in the event of adverse weather; whether the impact of wind on fuel handling has been evaluated and; what steps will be taken to

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[2]Between October 15 and May 15, the time period is 100 hours. Between May 16 and October 14, the time period is 168 hours.

minimize the impact of flying debris in the event of severe weather (July 16, 2004, RAI question 7).

In its August 20, 2004, response, PSEG stated that the existing site procedures for severe weather and adverse environmental conditions, which are based on National Weather Service Advisories or bulletins and actual measured weather conditions at the site, provide the criteria that will be used. PSEG also provided a partial copy of the Salem procedure SC.OP-AB.ZZ-0001, “Adverse Weather,” which contains steps to ensure that either the containment equipment hatch or outage equipment hatch are installed, and to evaluate the need to discontinue fuel movement. The licensee noted that procedure revisions that more explicitly address containment closure are to be included in the implementation of these amendments.

The NRC staff also asked if PSEG’s Emergency Plan will be updated to include accident release through the equipment hatch and whether the Emergency Operating Procedures will be updated to address specific details needed to respond to the accident scenario (July 16, 2004, RAI question 8).

PSEG responded that the Emergency Plan and its associated implementing procedures already address fuel handling events regardless of the manner in which they occur. Therefore, PSEG states that Emergency Plan or implementing procedure changes are not needed. However, as previously discussed, procedure revisions that address containment closure within 1 hour are to be included in the implementation of these amendments.

Finally, the NRC staff asked if State Emergency Response personnel will be informed of the FHA accident scenario (July 16, 2004, RAI question 9).

PSEG responded that this amendment request has already been discussed with representatives of the New Jersey Department of Environmental Protection Bureau of Nuclear Engineering. PSEG further stated that Delaware’s State Emergency Planning personnel will be notified of the amendment when it is approved.

Because the Emergency Plan and its associated implementing procedures contain provisions to ensure that containment closure will be established and evaluation of the need to secure fuel movement will be conducted, the NRC staff concludes that the proposed changes are acceptable.

3.5 SUMMARY

As described above, the NRC staff has previously reviewed the assumptions, inputs, and methods used by PSEG to assess the radiological impacts of an FHA that bound the proposed license amendment at Salem Unit Nos. 1 and 2. The staff found that PSEG used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The staff compared the doses estimated by PSEG to the applicable regulatory criteria and found, with reasonable assurance, that the licensee’s estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria.

With the NRC’s previous approval of a selective implementation of the AST, the selected characteristics of the AST and TEDE criteria became the design basis for the DBA FHA within the containment and outside the containment.

The proposed changes to the TSs identified in Section 3.3 were reviewed by the NRC staff and found to be in compliance with the NRC’s regulations. Thus, the licensee may implement these changes to their TSs. *The staff’s finding of acceptability for these changes is only associated with the removal of requirements from the TS.* Prior to removing any equipment or changing any procedure(s) affecting the operation of engineering safeguards equipment, the licensee must use the appropriate modification process (10 CFR 50.59 or 10 CFR 50.90). This will assure that the facility complies with all other commitments including draft GDC’s or their equivalents, the updated final safety analysis report, and other plant commitments and must demonstrate that safety margins and defense-in-depth are maintained. The licensee’s submittal demonstrates that, as a result of the partial implementation of AST methodology for the FHA, the TSs discussed above which the licensee has proposed to remove no longer meet the requirements of 10 CFR 50.36 for inclusion in the TSs. The licensee has proposed appropriate changes to SRs, and the implementation of appropriate administrative controls to ensure the ability to establish containment closure in a timely manner in the unlikely event of an FHA. These provide the basis upon which the NRC staff concludes the TS changes can be made.

In addition to the proposed TS changes, the licensee has proposed changes to the corresponding bases. The NRC staff has reviewed the proposed changes and found that they appropriately reflect the foundations for the TS requirements. The NRC staff does not have any objections to the proposed bases changes.

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 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 (67 FR 53989). 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: D. Cullison
S. La Vie
M. Blumberg
D. Collins

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