April 15, 2003

Mr. Roy A. Anderson
Chief Nuclear Officer
PSEG Nuclear LLC-X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: CONTAINMENT REQUIREMENTS DURING FUEL HANDLING AND REMOVAL OF CHARCOAL FILTERS (TAC NO. MB5548)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 28, 2002, as supplemented December 18, 2002, January 18, 2003, and February 25, 2003. The amendment makes use of the alternate source term in the analysis of the fuel-handling accident and the loss-of coolant accident to relax certain TSs for containment isolation and to remove the Filtration Recirculation and Ventilation System - Recirculation Subsystem charcoal filters from the TSs.

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/

George F. Wunder, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 146 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

Hope Creek Generating Station

cc:

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

Hope Creek Resident Inspector
U.S. Nuclear Regulatory Commission
Drawer 0509
Hancocks Bridge, NJ 08038

Mr. David F. Garchow
Vice President - Operations
PSEG Nuclear - X10
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. Gabor Salamon
Manager - Nuclear Safety and Licensing
PSEG Nuclear - N21
P.O. Box 236
Hancocks Bridge, NJ 08038

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

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

Richard Hartung
Electric Service Evaluation
Board of Regulatory Commissioners
2 Gateway Center, Tenth Floor
Newark, NJ 07102

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

April 15, 2003

Mr. Roy A. Anderson
Chief Nuclear Officer
PSEG Nuclear LLC-X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: CONTAINMENT REQUIREMENTS DURING FUEL HANDLING AND REMOVAL OF CHARCOAL FILTERS (TAC NO. MB5548)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 28, 2002, as supplemented December 18, 2002, January 18, 2003, and February 25, 2003. The amendment makes use of the alternate source term in the analysis of the fuel-handling accident and the loss-of coolant accident to relax certain TSs for containment isolation and to remove the Filtration Recirculation and Ventilation System - Recirculation Subsystem charcoal filters from the TSs.

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/
George F. Wunder, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 146 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC	JClifford	OGC	GMeyer, RGN-I	WBeckner
PDI-2 Reading	GWunder	ACRS	CRaynor	RDennig
Tech Branch	GHill (2)	SRichards		

ADAMS Accession Number: ML030760293, TS(s): ML , Package: ML
*See previous concurrence

OFFICE	PDI-2/PM	PDI-2/LA	DSSA/SPSB**	DSSA/SPLB***	DRIP/RORP*	OGC*	PDI-2/SC
NAME	GWunder	CRaynor	MReinhardt	EWeiss	RDennig	AHodgdon	JBoska for JClifford
DATE	4/11/03	4/11/03				04/9/03	4/11/03

OFFICIAL RECORD COPY

*Safety Evaluation dated February 27, 2003
** Safety Evaluation dated March 5, 2003

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
License No. NPF-57

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

   A. The application for amendment filed by PSEG Nuclear LLC dated June 28, 2002, as supplemented December 18, 2002, January 18, 2003, and February 25, 2003, 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 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. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 146, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by JBoska 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: April 15, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 146

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

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

Remove	Insert
1-2	1-2
1-3	1-3
1-4	1-4
3/4 3-16a	3/4 3-16a
3/4 3-31	3/4 3-31
3/4 6-47	3/4 6-47
3/4 6-49	3/4 6-49
3/4 6-51	3/4 6-51
3/4 6-51a	3/4 6-51a
3/4 6-52	3/4 6-52
3/4 6-52a	3/4 6-52a
3/4 6-53	3/4 6-53
3/4 6-53a	3/4 6-53a
3/4 7-1	3/4 7-1
3/4 7-3	3/4 7-3
3/4 7-5	3/4 7-5
3/4 7-6	3/4 7-6
3/4 8-11	3/4 8-11
3/4 8-17	3/4 8-17
3/4 8-23	3/4 8-23
B 3/4 3-2g	B 3/4 3-2g
B3/4 6-13	B3/4 6-13

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated June 28, 2002, as supplemented December 18, 2002, January 18, 2003, and February 25, 2003, PSEG Nuclear LLC (PSEG or the licensee) submitted a request for changes to the Hope Creek (Hope Creek) Generating Station Technical Specifications (TSs). The amendment would make use of the alternate source term (AST) in the analysis of the fuel-handling accident and the loss-of coolant accident to relax certain TSs for containment isolation and to remove the Filtration Recirculation and Ventilation System - Recirculation Subsystem charcoal filters from the TSs. The February 25, 2003, submittal did not change the staff's proposed finding of no significant hazards or expand the scope of the original Federal Register notice.

The proposed changes would revise the Hope Creek TSs based upon application of the alternate source term (AST) to design basis analyses. Specifically, the proposed changes would revise:

A. Definition 1.7 - Core Alterations

The licensee proposes to alter the definition by stating that the movement of a control rod, provided that there are no fuel assemblies in the associated core cell, is not considered a core alteration.

B. TS 3/4 3.2 - Isolation Actuation Instrumentation

The licensee proposes to modify the Applicable Operational Condition associated with a note in TS Tables 3.3.2-1 and 4.3.2-1. Certain instrumentation is required to be operable when handling any irradiated fuel and during core alterations. The proposed change would eliminate the operability requirement during core alterations and would specify that operability would be required when handling recently irradiated fuel.

C. TS 3.6.5.1 - Secondary Containment Integrity

The licensee proposes to modify the applicability and action statements to remove the reference "... during CORE ALTERATIONS..." and to clarify by stating that the Limiting Condition for Operation (LCO) is applicable when handling recently irradiated fuel.

D. TS 3.6.5.2 - Secondary Containment Automatic Isolation Dampers

The licensee proposes to modify the applicability and action statements to remove the reference "... during CORE ALTERATIONS..." and to clarify by stating that the LCO is applicable when handling recently irradiated fuel.

E. TS 3.6.5.3 - Filtration, Recirculation, and Ventilation System (FRVS)

The licensee proposes to modify the applicability and action statements to remove the reference "... during CORE ALTERATIONS..." and to clarify by stating that the LCO applies when handling recently irradiated fuel. The licensee also proposes to modify these statements, as appropriate, to eliminate references to the FRVS charcoal filters and heaters and to change several surveillance requirements.

F. TS 3.7.1 - Service Water Systems

The licensee proposes to modify the applicability and action statements by stating that the LCO applies when handling recently irradiated fuel.

G. TS 3.7.2 - Control Room Emergency Filtration System

The licensee proposes to modify the applicability and action statements by stating that the LCO applies when handling recently irradiated fuel.

H. TS 3.8.1 - A.C. Sources - Shutdown

The licensee proposes to modify certain applicability and action statements by stating that the LCO applies when handling recently irradiated fuel.

I. TS 3.8.2 - D.C. Sources - Shutdown

The licensee proposes to modify certain applicability and action statements by stating that the LCO applies when handling recently irradiated fuel.

J. TS 3.8.3 Onsite Power Distribution Systems

The licensee proposes to modify certain applicability and action statements by stating that the LCO applies when handling recently irradiated fuel.

The licensee also proposes to modify the TS BASES as appropriate. The bases will state that recently irradiated fuel is any fuel assembly which has been in a critical part of the core within the past 24 hours.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954 (the Act), as amended, requires applicants for nuclear power plant operating licenses to include TSs, which are derived from the plant safety analyses, as part of the license. In general, licensees cannot justify TS changes solely on the basis of having adopted the model standard TS. As a part of its review, the staff makes a determination that the proposed changes maintain adequate safety. Changes that result in relaxation (less restrictive condition) of current TS requirements require detailed justification. Such changes may be supported by evidence that the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. This amendment changes the design basis in that the licensee is adopting the AST into the design basis for the fuel-handling accident (FHA) and is also implementing the guidance contained in Technical Specification Task Force Item No. 51 (TSTF-51).

The Hope Creek TSs have a number of operational restrictions during shutdown conditions. The shutdown conditions requiring TS Operability are captured in the Applicability statements of the TSs. The standard wording of the Applicability statements during shutdown is:

> "...when irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel."

Following reactor shutdown, the radioactive decay of certain short-lived fission products results in a great reduction in the overall fission product inventory in the irradiated fuel. The proposed TSs take advantage of this reduction in the fission product inventory and apply the AST to their analysis of an FHA, the postulated accident during fuel handling and core alterations. The specific decay time assumed for Hope Creek was 24 hours. After 24 hours, active containment systems are no longer necessary to mitigate an FHA. Fuel that has not decayed for at least 24 hours is termed by the TS Bases to be "recently irradiated."

The original guidance for the use of source terms following design-basis accidents (DBAs) at nuclear power plants was issued by the U.S. Atomic Energy Commission in 1962 in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." 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. In 1995, the staff published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 provides estimates of the accident source term that are more physically based and that can be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment.

The staff considered the applicability of the revised source terms in NUREG-1465 to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The staff also determined that some licensees might wish to use an AST in their analyses to support cost-beneficial licensing actions. The staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design basis radiological consequence analyses. The results and findings of an evaluation of the impact of implementing the alternative source term for operating reactors are presented in SECY-98-154, "Results of the Revised Source Term Rebaselining for Operating Reactors."

The Commission approved the use of the AST at operating reactors in a staff requirements memorandum dated December 8, 1999, stating that "this action would allow interested licensees to pursue cost-benefit licensing actions to reduce unnecessary regulatory burden without compromising the safety of facility. Many of the alternative source term applications may provide concurrent improvements in overall safety and in reduced occupational exposures." These initiatives resulted in the development and issuance of 10 CFR 50.67 and Regulatory Guide (RG) 1.183.

A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR 50.67 to voluntarily revise its current accident source term used in design basis radiological consequence analyses for a license amendment under 10 CFR 50.90.

In addition to the use of AST in re-analysis of DBAs, the licensee is proposing the removal of heaters in both the ventilation sub-system and recirculation sub-system and the removal of charcoal adsorbers in the recirculation sub-system. In assessing these changes, the staff evaluated the licensee's request against RG 1.52 Revision 2, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," which provides guidance on design, testing, and maintenance of engineered safety feature air filtration and adsorption systems; General Design Criterion (GDC) 19, which provides requirements and regulations on maintaining a habitable control room and includes limitations on radiological dose that may be received by control room operators; Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear Grade Charcoal," which provides requirements for charcoal tested in accordance with American Society for Testing and Materials (ASTM) D3803-1989; ASTM D3803-89, "Standard Test Method for Nuclear Grade Activated Carbon," which provides methodology for charcoal testing at 30°C and 95% relative humidity; and American National Standards Institute (ANSI) N510-1975, "Testing of Nuclear Air Cleaning Systems," which provides requirements for air flow capacity testing through the ESF filtration system.

3.0 TECHNICAL EVALUATION

The licensee re-analyzed and submitted the radiological consequence analyses for the following two DBAs:

- FHA, and
- Loss-of-Coolant Accident (LOCA).

3.1 Fuel-Handling Accident

During refueling operations, the most restrictive DBA requiring containment operability is the FHA. By re-analyzing the FHA, the licensee justified the relaxation of certain containment operability requirements when handling irradiated fuel that had decayed for at least 24 hours. The current radiological consequence analysis for the postulated design basis FHA is based on the accident source term described in TID-14844 and it is provided in the Hope Creek Updated

Final Safety Analysis Report (UFSAR) Section 15.7.4. The licensee re-analyzed the radiological consequences of a postulated FHA in the containment with no credit taken for containment isolation using the AST. The FHA is postulated to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in a drop of a raised fuel assembly onto stored fuel assemblies in the reactor core. The licensee assumed a total of 124 fuel rods are damaged. The staff has accepted 124 fuel rod failures as the licensing basis for this accident in the staff's safety evaluation (SE) in NUREG-1048, "Safety Evaluation Report Related to the Operation of Hope Creek Generating Station." The fuel rod failure mechanism is described in the Hope Creek UFSAR Section 17.7.4.7. As the use of AST does not affect the postulated failure mechanism for this accident, the staff considers the assumption of 124 failed rods to continue to be acceptable.

Instantaneous release of all noble gases and iodine vapors from the fuel rod gaps from the damaged fuel rods occurs as gas bubbles up through the water covering the fuel. All fission products reaching the reactor building atmosphere are released directly to the environment within 2 hours without filtration. The licensee concluded that the radiological consequences resulting from the postulated FHA in the containment with no credit taken for containment isolation are within the dose acceptance criteria specified in Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and GDC 19.

The licensee reached this conclusion based on their analysis using the following assumptions:

(1) implementing the AST,
(2) taking no credit for containment isolation,
(3) taking no credit for fission product removal by the reactor building FRVS Recirculation System (FRVS-RS), the reactor building FRVS Ventilation System (FRVS-VS), and control room emergency filtration system,
(4) using an overall decontamination factor of 200 for iodine in elemental and particulate forms in the spent fuel pool water with minimum water depth of 23 feet consistent with the guidelines provided in RG 1.183,
(5) releasing all fission products within 2 hours,
(6) assuming all fuel rods in one fuel assembly with an axial power peaking factor of 1.5 are damaged to the extent that the entire gap activity inventory of the damaged fuel rods is released instantaneously to the surrounding water,
(7) using a fission product decay period of 24 hours (time period from the reactor shutdown to the first fuel movement),
(8) using the guidance provided in Appendix B to RG 1.183, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident."

The staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the postulated FHA. The radiological consequences calculated by the staff are within the dose criterion specified in GDC 19 (5 rem total effective dose equivalent (TEDE) in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1 (6.3 rem TEDE at the exclusion area boundary (EAB)).

Even though the staff performed its confirmatory dose calculations, the staff's acceptance is based on our review of the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the staff are listed in Table 3. The radiological consequences at the EAB, at the low-population zone (LPZ), and in the control room as calculated by the licensee are also within the dose criterion specified in GDC 19 and meet the dose acceptance criterion specified in the SRP 15.0.1, and are, therefore, acceptable.

3.2 Control Room Habitability

The licensee normally maintains the Hope Creek control room at a slightly positive pressure to prevent the introduction of air into the control room from sources other than the 1000 cubic feet per minute (cfm) outdoor air makeup flow. The licensee proposed to manually isolate the control room air intakes no later than 30 minutes after the initiation of the postulated LOCA. During this 30-minute period, the licensee assumed an unfiltered air inleakage rate of 500 cfm. Once the air intakes are isolated, the control room atmosphere is recirculated through the control room emergency filtration (CREF) system at 3600 cfm with 1000 cfm of makeup air. The licensee also assumed 350 cfm of unfiltered air inleakage to the control room beginning 30 minutes into the accident and continuing throughout the 30-day accident period.

The results of the licensee's control room radiological consequence calculations are given in Table 1. The major parameters and assumptions used by the staff in its confirmatory dose calculation and by the licensee in its dose calculation are listed in Tables 2 through 4. The radiological consequences to the control room operator calculated by the licensee and confirmed by the staff are within the dose criterion specified in 10 CFR 50.67 and, therefore, are acceptable.

3.3 Atmospheric Relative Concentrations at Control Room Air Intake, EAB, and LPZ

The meteorological data used in the relative concentration (X/Q) calculations for this amendment are discussed in the SE associated with Amendment No. 134, dated October 3, 2001. The X/Q values listed below for the EAB and LPZ were also previously approved as part of that amendment.

For this FHA dose assessment, the licensee calculated control room X/Q values using the ARCON96 methodology (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake"), with modifications to the surface roughness length and averaging sector width constant recommended by the staff for generic application based on further consideration of nuclear power plant sites and the calculational procedure. The licensee assumed a ground level point release from the reactor building truck bay door, with no forced flow, and used the shortest horizontal straight-line distance between the door and the intake. Staff qualitatively reviewed the inputs to the code and found them to be reasonably consistent with site configuration drawings and staff practice. Based on this review, the staff finds the X/Q values acceptable for the postulated FHA release. The X/Q values are listed in Table 4 of this SE.

3.4 Loss-of-Coolant Accident

To demonstrate the adequacy of the Hope Creek engineered safety features (ESF) to mitigate the radiological consequences of design-basis LOCA after the removal of the FRVS-RS charcoal filters, the licensee recalculated the offsite and control room radiological doses from a postulated LOCA at a reactor core power level of 3857 megawatts thermal (MWt). This power level is 15.5% above the current licensed power level of 3339 MWt. In its dose calculations, the licensee used the RADionuclide Transport and Removal And Dose Estimation (RADTRAD) computer code, Version 3.02. The RADTRAD code was developed by Sandia National Laboratories, the Nuclear Regulatory Commission's (NRC's) technical contractor, for the staff to use in establishing fission product transport and removal models and in estimating radiological doses at selected receptors at nuclear power plants. The licensee submitted the inputs to, and outputs from, the code, along with the resulting radiological consequences at the EAB, in the LPZ, and the control room.

In its submittal, the licensee concluded that the existing Hope Creek ESF systems, with this license amendment, would still provide adequate assurance that the radiological consequences of a postulated LOCA at the EAB, in the LPZ, and in the control room would be within the dose criteria specified in 10 CFR 50.67. The licensee calculated the radiological consequences for the following three potential fission product release pathways after the postulated LOCA:

(1) containment leakage;
(2) post-LOCA leakage from ESF systems outside containment; and
(3) main steam isolation valve (MSIV) leakage.

These three potential fission product release pathways are evaluated in Sections 3.4.1, 3.4.2, and 3.4.3 of this SE.

Table 1 summarizes the results of the licensee's radiological consequence calculations, while Tables 2 and 4 list the major parameters and assumptions used by the licensee in its radiological consequence calculations and by the staff in its confirmatory dose calculations.

3.4.1 Containment Leakage Pathway

The FRVS consists of two ESF subsystems, the FRVS-VS and the FRVS-RS. The FRVS-VS processes and filters air from the containment before it is released to the environment. The FRVS-RS cleans contaminated air recirculated through the reactor building. The licensee is proposing to remove the charcoal filters from the FRVS-RS; this amendment does not remove the filters from the FRVS-VS.

The licensee evaluated the radiological consequences resulting from containment leakage following a postulated design-basis LOCA at a reactor core power level of 3857 MWt. The licensee used a containment leak rate of 0.5% per day based on the allowable Hope Creek TS limit for the first 24 hours and a 0.25% per day leak rate for the remaining 29 days of the accident period, consistent with the guideline provided in RG 1.183. The licensee also assumed that the source term in the primary containment mixes instantaneously and homogeneously throughout the free air volume of the primary containment. Hope Creek has a General Electric Mark 1 type containment. In addition, because the reactor building is not maintained at a 0.25-inch water gauge negative pressure relative to adjacent areas during the first 375 seconds of the accident, the licensee assumed that all containment leakage is released unfiltered to the environment. After this initial 375-second period, the licensee assumed that primary containment leakage is processed by the FRVS before being released to the environment.

The FRVS-RS consists of 6 25% capacity trains, each of which has a flow capacity of 30,000 cubic feet per minute (cfm). Of the 6 trains, 4 are normally in operation, with a total combined flow capacity of 120,000 cfm. Therefore, the licensee assumed a combined containment air mixing flow rate of 108,000 cfm by 4 trains (90% of the rated capacity of each train, or 27,000 cfm each). The licensee did not credit any iodine removal by the charcoal adsorbers in the FRVS-RS.

The FRVS-VS is designed to exhaust sufficient air from the reactor building to maintain a negative pressure in that building and to remove airborne radioactive materials before discharging the air to the environment. The FRVS-VS takes suction only from the discharge duct of the FRVS-RS. The licensee assumed a reactor building air mixing efficiency of 50%. To simulate the 50% air mixing in the reactor building, the licensee doubled the FRVS-VS release rates to the environment. The licensee's evaluation of radiological consequences used a 90% iodine removal efficiency by charcoal adsorbers in the FRVS-VS.

The licensee did not credit the safety-related drywell spray system for removal of fission products. Instead, the licensee assumed aerosol removal in the unsprayed area of the containment by natural deposition, using the model provided in the RADTRAD code with a 10th percentile uncertainty distribution.

The radiological consequence contribution from this release pathway resulting from the postulated LOCA, as calculated by the licensee, is shown in Table 1. The overall radiological consequences from the combined contributions from all release pathways are evaluated in Section 3.4.4 of this SE.

3.4.2 Post-LOCA ESF System Leakage Pathway

With the exception of noble gases, the licensee assumed that all of the fission products that are released from the fuel to the containment, instantaneously and homogeneously mix with the suppression pool water at the time of release from the core. Any water leakage from ESF components located outside the primary containment releases fission products during the recirculating phase of long-term core cooling after a postulated LOCA. In the Hope Creek UFSAR, the licensee estimated this leakage to be less than 10 gallons per minute (gpm). In this license amendment request, the licensee proposed to reduce this leakage to 1 gpm from 10 gpm. As stated in their February 25, 2003, response to the staff's questions, historical leakage has been less than 1 gpm. In addition, the licensee has in place a TS required program to monitor and control such leakage. The staff, therefore, considers the use of 1 gpm to be acceptable. The licensee used 2 gpm (two times design basis leakage value) in its dose calculation for the entire duration of the accident (i.e., 30 days) as the staff did in its confirmatory dose calculation consistent with the guideline provided in RG 1.183.

The licensee assumed that 30% of the core iodine inventory mixes with the suppression pool water and circulates through the containment's external piping systems. The licensee also assumed that 10% of the iodine in the liquid leakage becomes airborne, and the airborne iodine is immediately released to the environment. In addition, the licensee assumed that radio iodine that is postulated to be available for release to the environment is 97% in elemental iodine form and 3% in organic iodine form. These assumptions are consistent with RG 1.183 and are, therefore, acceptable. The radiological consequence contribution from this release pathway resulting from the postulated LOCA, as calculated by the licensee, is shown in Table 1. The overall radiological consequences from the combined contributions from all release pathways are evaluated in Section 3.4.4 of this SE.

3.4.3 MSIV Leakage Pathway

Hope Creek has four main steam lines, each of which has both an inboard MSIV and an outboard MSIV. These valves isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. The licensee assumed a double-guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV. A total of 250 standard cubic feet per hour (scfh, the TS limit) is assumed to occur in the following ways: 150 scfh through the broken steam line, 50 scfh through a first-intact steam line, the remaining 50 scfh through a second intact steam line, and no leakage from a third-intact steam line. These leakage assumptions are current licensing bases as approved in Hope Creek License Amendment No. 134.

During the postulated LOCA, the main steam leakage flow pattern in the main steam lines could be plug flow, well-mixed flow, or some combination of the two. If temperature gradients exist along the length of the main steam line, then some degree of mixing would occur. For the same leakage rate into the main steam line, plug flow is expected to result in less offsite release than well-mixed flow, since the concentration of the fission product released to the environment is equal to the concentration of the fission product in the plug at the end of the main steam line. Plug flow effectively results in a longer fission product transport time in the steam line, with more aerosol deposition in the steam lines.

In its dose calculation for this release pathway, the licensee used the model developed and used by the staff in its review of a similar license amendment request for Perry Nuclear Power Plant, as described in the staff's Technical Report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," dated December 9, 1998. Although many of the systems at Perry Nuclear Power Plant and Hope Creek are of different designs, the aerosol deposition rates of fission products in the main steam system will be similar; therefore, the staff finds the use of this model to be acceptable for dose calculations for this release pathway. This model uses the RADTRAD code to calculate the resulting radiological consequences based on a plug flow model, supplemented with a separate calculation of aerosol settling velocities based on the well-mixed steam flow in the entire length of the main steam line. In AEB-98-03, the staff performed a Monte Carlo analysis to determine the distribution of aerosol settling velocities in the main steam lines. For the uncertainty analyses, the staff used the ranges and distributions provided in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," for aerosol density, diameter, viscosity, packing fractions, and shape factors.

In AEB-98-03, the staff stated in part, the following:

> Complete mixing (in the steamline) may not occur along the entire length of the pipe and, in some pipe segments, plug flow may exist. Given the conservatism associated with using a well-mixed model for the entire length of the pipe and a number of

> additional conservatisms inherent in the piping deposition analysis, use of a 10th percentile settling velocity with a well-mixed model is not appropriate. Additional conservatism includes additional (aerosol) deposition by thermophoresis, diffusiophoresis, and flow irregularities; additional deposition as a result of hydroscopicity; and possible plugging of the leaking MSIV by aerosols. Given the conservatism of the well-mixed assumption, we believe it is acceptable then to utilize median values (of 40th percentile uncertainty distribution) as compared to more conservative values for deposition parameters.

In its radiological consequence analysis, the licensee selected and used the aerosol settling velocity in the 40^{th} percentile uncertainty distribution (as the staff justified in AEB-98-03) to calculate the aerosol removal rate using the Hope Creek specific main steam piping parameters. The portions of the main steam piping that the licensee credited for aerosol removal are classified as either seismic Category 1, or seismically analyzed, and are located in the reactor building and the turbine building steam tunnel. The staff finds that the method that the licensee used to calculate aerosol deposition in the main steam pipe is acceptable.

Gaseous iodine, in elemental form, also deposits on the piping surface by chemical adsorption. The iodine deposited on the pipe surface undergoes both physical and chemical changes and can be resuspended as different iodine chemical species, or permanently fixed to the pipe surface. For elemental iodine deposition and re-suspension, the licensee used the model and methodology developed by Science Applications International Corporation, an NRC technical contractor, for the staff to use in establishing iodine transport and removal models and in estimating radiological doses at selected receptors at nuclear power plants. The models are provided in a contractor's report titled "MSIV Leakage Iodine Transport Analyses," dated August 1990. RG 1.183 states that these models are acceptable and the staff has determined that the licensee applied these models appropriately.

The radiological consequence contribution from this release pathway resulting from the postulated LOCA, as calculated by the licensee, is shown in Table 1. The overall radiological consequences from the combined contributions from all release pathways are evaluated in Section 3.4.4 of this SE.

3.4.4 Resulting Radiological Consequences from the Postulated LOCA

The licensee re-evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The staff has reviewed the licensee's re-evaluation. In performing this review, the staff relied upon information provided by the licensee; staff experience in performing similar reviews; and, where deemed necessary, on confirmatory calculation. The staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183.

To verify the licensee's radiological consequence analyses, the staff performed its confirmatory radiological consequence dose calculation and found the staff's results are also within the dose criteria specified in 10 CFR 50.67. Although the staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence dose calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and the staff are listed in Tables 2 and 4. The radiological consequences calculated by the licensee and by the staff for the EAB and at the LPZ, and in the control room are all within the dose criteria specified in 10 CFR 50.67. The staff, therefore, concludes that the proposed TS changes implementing the AST meet the relevant dose acceptance criteria and are, therefore, acceptable.

3.4.5 Tables for Radiological Consequence Evaluation

TABLE 1
Radiological Consequences
for
Postulated Design Basis LOCA
(rem TEDE)[(1)]

Release Pathway	EAB	LPZ	Control Room
LOCA			
Containment leak	0.44	0.17	1.0
ESF leak	0.02	0.09	1.2
MSIV leak	2.45	0.43	2.0
TOTAL	2.91	0.69	4.2
Dose criteria [(2)]	25	25	5
Fuel Handling Accident	0.52	0.05	3.21
Dose criteria [(3)]	6.3	6.3	5

(1) Rounded to two significant digits
(2) From 10 CFR 50.67
(3) From SRP 15.0.1

TABLE 2
Parameters and Assumptions Used in Radiological Consequence Calculations for a LOCA

Parameter	Value
Reactor power	3,857 MWt
Drywell air volume	1.69E+5 ft^3
Containment air volume	3.06E+5 ft^3
Reactor building air volume	4.0E+6 ft^3
Containment leak rate to environment	
0 - 24 hours	0.5% per day
1 - 30 days	0.25% per day
Reactor building pressure drawdown time	375 seconds
Aerosol deposition rate in drywell	10 percentile in RADTRAD
Reactor building mixing efficiency	50%
FRVS vent exhaust filter efficiencies	
Elemental iodine	90%
Organic iodine	90%
Aerosol (particulate)	99%
FRVS recirculation filter efficiencies	
Elemental iodine	Not credited
Organic iodine	Not credited
Aerosol (particulate)	99%
FRVS recirculation flow rate	1.08E+5 cfm
ECCS leak rate	1 gpm
ECCS iodine partition factor	10%
ECCS leak initiation time	0 minutes
Sump volume	1.18E+5 ft^3
MSIV leak rate	
All four lines	250 scfh
Line with MSIV failed	150 scfh
First intact line	50 scfh
Second intact line	50 scfh
Aerosol settling velocity on main steamlines	8.1E-4 meters/second
Aerosol settling area (well-mixed region volumes)	
MSIV faulted line	1398 ft^3
MSIV intact lines	1476 ft^3

TABLE 2
(continued)
Parameters and Assumptions Used in Radiological Consequence Calculations for a LOCA

Control room volume	8.5E+4 ft^3
CREF system outside air intake flow	1000 cfm
CREF recirculation flow	2600 cfm
Control room isolation time	30 minutes
Unfiltered air in leakage rate into control room	
0 to 30 minutes	500 cfm
30 minutes to 30 days	350 cfm
CREF system filter efficiencies	
Elemental iodine	99%
Organic iodine	99%
Aerosol (particulate)	99%

Table 3
Parameters and Assumptions
Used in
Radiological Consequence Calculations
FHA

Parameter	Value
Reactor power	3972 MWt
Radial peaking factor	1.5
Fission product decay period	24 hours
Number of fuel rod damaged	124
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5%
Kr-85	10%
I-131	8%
Alkali metals	12%
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Duration of accident	2 hours
Fission product release point	ground level release from reactor building truck bay door
Control room volume	8.5E+4 ft^3
Control room isolation	Not isolated
Control room normal flow rate	
0 to 720 hours	3000 cfm
Unfiltered air in leakage rate into control room	
0 to 30 minutes	500 cfm
30 minutes to 30 days	350 cfm
CREF system filter efficiencies	
Elemental iodine	Not credited
Organic iodine	Not credited
Aerosol (particulate)	Not credited

TABLE 4
Hope Creek Meteorological Data

EAB

Time	X/Q (sec/m^3)
0 - 2 hrs	1.9 E-04

LPZ

Time	X/Q (sec/m^3)
0 - 2 hrs	1.9 E-05
2 - 4 hrs	1.2 E-05
4 - 8 hrs	8.0 E-06
8 - 24 hrs	4.0 E-06
1 - 4 days	1.7 E-06
4 - 30 days	4.7 E-07

Control Room from Reactor Building Truck Bay Door

Time	X/Q (sec/m^3)
0 - 2 hrs	1.39 E-03
2 - 8 hrs	1.17 E-03
8 - 24 hrs	4.76 E-04
1 - 4 days	3.20 E-04
4 - 30 days	2.60 E-04

3.5 Changes to FRVS Systems

The licensee proposed the following changes to SRs for the FRVS located in TS 3/4.6.5.3 which consists of two sub-systems: TS 3/4.6.5.3.1, FRVS ventilation sub-system; and TS 3/4.6.5.3.2, FRVS recirculation sub-system. As described in the licensee's request, the proposed changes are needed to support removing the FRVS-VS heaters and the FRVS-RS heaters and charcoal adsorbers from the TSs.

3.5.1 Ventilation Sub-System

3.5.1.1 SR 4.6.5.3.1

Current SR 4.6.5.3.1. b requires each of the two ventilation units to be demonstrated OPERABLE "at least once per 31 days, by initiating from the control room flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters."

The proposed SR 4.6.5.3.1.b would change this requirement to " at least once per 31 days, by initiating from the control room flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 15 minutes."

The purpose of the 10-hour SR is to reduce the buildup of moisture on the carbon adsorber and to verify that both the heaters and the system are capable of operating as assumed in the accident analysis. In this case, the licensee is removing the heaters from the TSs and reducing the surveillance time from 10 hours to 15 minutes and, therefore, will no longer take credit for the operation of heaters in the accident analysis. The licensee continues to verify the operability of the charcoal adsorbers by testing them at 95% humidity as opposed to 70% humidity (see SRs below). The staff's position, as outlined in Regulatory Position C.6.1 of RG 1.52, Revision 3, June 2001, is that ESF atmospheric cleanup trains should be operated for 15 minutes each month to justify operability of the system and its components. Because the heaters are disconnected and the operability of the charcoal adsorbers is verified by testing, the 15 minute run time is adequate to verify the operability of the system. This requested change is consistent with the staff's position and, therefore, is acceptable.

3.5.1.2 SR 4.6.5.3.1

Current SR 4.6.5.3.1.c.2 requires each of the two ventilation units to be demonstrated OPERABLE by "verifying within 31 days after removal from the FRVS ventilation units, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30^{o}C and a relative humidity of 70%."

Proposed SR 4.6.5.3.1.c.2 would change this requirement to demonstrating OPERABILITY by "verifying within 31 days after removal from the FRVS ventilation units, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration of less than 5% when tested in accordance with ASTM D3803-1989 at a temperature of 30^{o}C and a relative humidity of 95%."

The licensee is changing the test for charcoal addsorber OPERABILITY from one that shows a penetration of less than 2.5% and is conducted at 70% relative humidity to one that shows a penetration of 5% and is conducted at 95% relative humidity. The higher humidity requirement for testing reflect the removal of the heaters. RG 1.52, Revision 3, and GL 99-02 both state that a test that shows a penetration of 5% and is conducted at 95% relative humidity is acceptable; therefore, the staff finds this change to the TSs acceptable.

Current SR 4.6.5.3.1.d requires each of the two ventilation units to be demonstrated OPERABLE "after every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FRVS ventilation units, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30^{o}C and a relative humidity of 70%."

Proposed SR 4.6.5.3.1.d would change this requirement to read "after every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FRVS ventilation units, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration of less than 5% when tested in accordance with ASTM D3803-1989 at a temperature of 30^{o}C and a relative humidity off 95%."

The staff reviewed the proposed changes and determined that the licensee will test charcoal adsorbers in accordance with ASTM, D3803, 1989, at a temperature of 30°C and a relative humidity of 95% to show methyl iodide penetration of less than 5%. This is in accordance with the staff guidance provided in RG 1.52 and in GL 99-02 and is, therefore, acceptable.

Current SR 4.6.5.3.1.e.3 requires each of the two ventilation units to be demonstrated OPERABLE by "verifying that the heaters dissipate 32 ± 3kw for each ventilation unit when tested in accordance with ANSI N510-1980, and verifying humidity is maintained less than or equal to 70% relative humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation."

The licensee's proposed amendment would delete this SR. The licensee is removing the heaters from the TSs and will disconnect them; therefore, removing this SR on the heaters is acceptable.

The staff reviewed the changes to the ventilation sub-system and understands that the changes are based on the removal of the heaters from the ventilation system. The purpose of the heaters is humidity control; without the heaters charcoal adsorber operability must be determined by testing at 30°C and 95% relative humidity. The licensee will be conducting testing at 30°C and 95% relative humidity; therefore, the proposed changes are acceptable.

3.5.2 Recirculation Sub-System

The licensee is proposing to remove the TS requirements for charcoal adsorbers in the FRVS-RS subsystem. The FRVS-RS is designed to filter and clean contaminated air in the reactor building after a DBA or abnormal occurrence that could result in high airborne radiation levels in the reactor building. The FRVS-RS consists of six units. Each unit has, among other things, a pre-HEPA filter, a 2-inch deep charcoal adsorber, and a post-HEPA filter. The licensee requested to delete these charcoal adsorbers in the FRVS-RS (not in the FRVS-VS) from the Hope Creek TSs. The licensee did not request any changes in the operation of the FRVS-RS.

In 1995, the NRC published NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants.” NUREG-1465 utilized the results of the major research efforts on developing alternative source terms started by the NRC and the nuclear industry after the accident at Three Mile Island, Unit 2. Subsequently, the NRC sponsored significant review efforts of the results of the major research efforts by peer reviewers, foreign research partners, industry groups, and general public. The revised source terms in NUREG-1465 are described in terms of radionuclide composition, and magnitude, physical and chemical form, and timing of release.

Where old source terms (described in TID-14844) assume radioiodine to be predominantly (greater than 95%) in elemental or organic form that is amenable to be removed by charcoal adsorbers, the revised alternative source terms assume radioiodine to be predominantly (greater than 95%) in cesium iodide, an aerosol that is amendable to be removed by HEPA filters. Consequently, the role of charcoal filters in mitigating fission products became less significant compared to that of HEPA filters.

In their letter dated February 25, 2003, the licensee provided the following radiological consequence information (all in rem TEDE):

	EAB	LPZ
With charcoal adsorbers in FRVS-RS	2.83	0.55
Without charcoal adsorbers in FRVS-RS	3.07	0.69
Increase in dose	0.24	0.14
Allowable regulatory limits in 10 CFR 50.67	25.0	25.0
Percentage increase in dose to the limits	0.96%	0.55%

The staff confirmed the incremental doses provided by the licensee with its own independent dose calculations. The staff concluded that the incremental increases in the radiological consequence are small and that the radiological consequence at the EAB and LPZ following a postulated LOCA are within the dose criteria specified in 10 CFR 50.67. The staff finds, therefore, that the proposed deletion of charcoal adsorbers in the FRVS-RS is acceptable. A detailed evaluation of the radiological consequences of the removal of the charcoal filters is presented in Section 3.4 of this SE.

3.5.2.1 SR 4.6.5.3.2

Current SR 4.6.5.3.2.b requires each of the six FRVS recirculation units to be demonstrated OPERABLE “at least once per 31 days, by initiating from the control room flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters.”

Proposed SR 4.6.5.3.2.b would change this requirement to “at least once per 31 days, by initiating from the control room flow through the HEPA filters and verifying that the subsystem operates for at least 15 minutes.”

The purpose of the 10-hour SR is to reduce the buildup of moisture on the carbon adsorber and to verify that both the heaters and the system are capable of operating as assumed in the accident analysis. In this case, the licensee is removing the heaters and the adsorbers from the TSs and, therefore, will no longer take credit for their operability in the accident analyses. Because the heaters will be disconnected and the charcoal adsorbers will be removed from the TSs and eventually will be removed from the system, the 15 minute run time is adequate to verify the operability of the system in accordance with RG 1.52.

Current Section 4.6.5.3.2.c requires each of the six FRVS recirculation units to be demonstrated OPERABLE “at least once per 18 months or upon determination ** that the HEPA filters or charcoal adsorber could have been damaged by structural maintenance or adversely affected by any chemicals, fumes or foreign materials (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by: ... ”

The proposed revision to Section 4.6.5.3.2.c would change this requirement to “at least once per 18 months or upon determination ** that the HEPA filters could have been damaged by structural maintenance or adversely affected by any foreign materials (1) after any structural maintenance on the HEPA filter housings by: ... ”

This test is required every 18 months, or following determination that the HEPA filters or charcoal adsorber could have been adversely affected by structural maintenance or other specific events. As stated above, the staff has found that it is acceptable to remove the FRVS-RS charcoal adsorber from these TSs; therefore, any SRs for these adsorbers must be removed from the TSs as well. The requirement to determine OPERABILITY after fire, painting, or chemical release has been removed because these are events that would have affected the OPERABILITY of the charcoal adsorbers, but not the rest of the system.

Current SR 4.6.5.3.2.c.1 requires the licensee to determine recirculation unit OPERABILITY by “verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.6.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 30,000 cfm ± 10% for each FRVS recirculation unit.”

Proposed SR 4.6.5.3.2.c.1 would change this requirement to “verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.6.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 30,000 cfm ± 10% for each FRVS recirculation unit."

This change eliminates the requirement for in-place testing of the charcoal adsorber. As stated above, the staff has found that it is acceptable to remove the FRVS-RS charcoal adsorber from these TSs; therefore, any references to these adsorbers in SRs must be removed from the TSs as well.

Current SR 4.6.5.3.2.c.2 requires the licensee to determine recirculation unit OPERABILITY by "Verifying within 31 days after removal from the FRVS ventilation units, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30^{o}C and a relative humidity of 70%."

The licensee's proposed amendment would delete this SR. This is a SR on the charcoal adsorbers and, as stated above, the staff has determined that it is acceptable to remove these adsorbers from the TSs.

Current SR 4.6.5.3.2.c.3 is renumbered Section 4.6.5.3.2.c.2.

This is an administrative change that has no effect on the operation of the system; therefore, it is acceptable.

Current SR 4.6.5.3.2.d requires each of the six FRVS recirculation units to be demonstrated OPERABLE "after every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FRVS ventilation units, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30^{o}C and a relative humidity of 70%."

The licensee's proposed amendment would delete this SR. This is a SR on the charcoal adsorbers and, as stated above, the staff has determined that it is acceptable to remove these adsorbers from the TSs.

Current Footnote ** to SR 4.6.5.3.2 states: "This determination shall consider the maintenance performed and/or the type, quantity, length of contact time, known effects and previous accumulation history for all contaminants which could reduce the system performance to less than that verified by the acceptance criteria in items c.1 through c.3 below."

Proposed Footnote ** states: "This determination shall consider the maintenance performed and/or the type, quantity, length of contact time, known effects, and previous accumulation history for all contaminants which could reduce the system performance to less than that verified by the acceptance criteria in items c.1 and c.2."

The staff finds the change to this footnote acceptable because it reflects the renumbering necessitated by eliminating current SR 4.6.5.3.2.c.2.

Current SR 4.6.5.3.2.e.1 requires the licensee to demonstrate recirculation unit OPERABILITY by “verifying that the pressure drop across the combined HEPA filter and charcoal adsorber banks is less than 3 inches Water Gauge in the recirculation filter train while operating the filter train at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.”

The proposed SR 4.6.5.3.2.e.1 would change this requirement to “verifying that the pressure drop across the exhaust duct is less than 3 inches Water Gauge in the recirculation filter train while operating the filter train at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.”

This change reflects removal of the charcoal adsorber, and is acceptable because the staff agreed to its removal.

Current SR 4.6.5.3.2.e.3 requires the licensee to demonstrate recirculation unit OPERABILITY by “verifying that the heaters dissipate 32 ± 3kw for each ventilation unit when tested in accordance with ANSI N510-1980, and verifying humidity is maintained less than or equal to 70% relative humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation.”

The licensee’s proposed amendment would delete this requirement. This is a calibration test to verify that the heaters are capable of maintaining the humidity through the carbon adsorber at 70% or less. This change is acceptable because the staff agreed to the removal of the charcoal adsorber, thereby eliminating the need for both the calibration test and the heaters.

Current SR 4.6.5.3.2.g requires the licensee to demonstrate recirculation unit OPERABILITY “after each complete or partial replacement of a charcoal adsorber bank by verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.6.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rates of 30,000 cfm ± 10% for each FRVS recirculation unit.”

The licensee’s proposed amendment would delete this requirement.

This is an in-place penetration test requirement for the charcoal adsorber bank; the staff agreed to the removal of the charcoal adsorber and, therefore, finds the deletion of the test requirement acceptable.

The staff has reviewed the licensee’s regulatory and technical analyses in support of its proposed license amendment, which are described in the licensee’s submittal with reference to changes in the FRVS system. On the basis of the above regulatory and technical evaluations of the licensee’s justifications for TS changes, the staff concludes that the licensee’s proposed TS changes are acceptable.

3.6 Evaluation for Compliance with TSTF-51

Following reactor shutdown, rapid decay of the short-lived fission products quickly reduces the fission product inventory present in irradiated fuel. The proposed TS changes are based on a specific minimum decay period which takes advantage of the reduced radionuclide inventory available for release in the event of an FHA. For Hope Creek, this specific decay period is calculated to be 24 hours. Beyond 24 hours, containment isolation is no longer required to mitigate the consequences of the FHA. The FHA is the bounding accident during fuel handling and core alterations.

TSTF-51 uses the concept of recently irradiated fuel. Fuel that is not sufficiently decayed to allow relaxation of the containment OPERABILITY requirement is referred to as recently irradiated fuel. During movement of recently irradiated fuel containment OPERABILITY is required to ensure that the offsite doses remain within acceptable limits in the event of an FHA. Hope Creek's analysis demonstrates that at least a 24-hour decay time will sufficiently reduce the inventory of short-lived radionuclides. For Hope Creek, therefore, recently irradiated fuel is defined as fuel that has decayed less than 24 hours. When using 24 hours for the decay time in the design-basis FHA, the radiological consequences remain within the acceptance criteria of 10 CFR 50.67 and GDC 19.

When implementing TSTF-51, licensees commit to following the guidelines in Revision 3 of NUMARC 93-01, Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities." This guidance states in part that, when licensees are conducting maintenance that involves the need for an open containment, they should evaluate their ability to close the containment in time to mitigate potential fission product releases. The guidance goes on to state that licensees should develop a method to close containment penetrations promptly in order to enable ventilation systems to draw any release from an FHA in such a way that it could be treated and monitored.

The proposed Hope Creek amendment uses the concept of recently irradiated fuel. The proposed amendment is consistent with the TSTF-51 revision to the Standard Technical Specifications. In their January 18, 2003, submittal, PSEG stated that they would follow the guidelines in Section 11 of NUMARC 93-01, Revision 3, at Hope Creek during refueling inside containment. The staff, therefore, concludes that the proposed amendment is consistent with TSTF-51.

4.0 SUMMARY

The staff has evaluated the licensee's proposed changes to the TSs and has determined that they will not result in doses in excess of a small percentage of the limits of 10 CFR 50.67. The staff has also concluded that the changes to the TSs will not result in doses that will exceed the guidance of GDC 19. With respect to the changes to the TS governing the FRVS system, the staff has concluded that the removal of the charcoal filters is acceptable and that the changes to the SRs are either necessary to support the removal of these filters or are otherwise in conformance with staff positions. The staff has also determined that the proposed changes to the TSs are consistent with the TSTF-51 revisions to the Standard Technical Specifications. These proposed changes are, therefore, acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. By letter dated April 4, 2003, the State of New Jersey submitted comments on the proposed amendment. The comments stated that the State of New Jersey: 1) opposed the removal of the FRVS-RS charcoal filters because they could lead to increased radioiodine release off site in the event of an accident, and 2) opposed the relaxation of containment operability requirements during fuel handling because of the increased possibility of an unfiltered release to the environment. The staff has found that the licensee has met the applicable regulations and that their proposed changes provide an adequate level of safety. The staff also finds that the State of New Jersey has not raised any technical issues that were not considered in the review of the licensee's request for amendment.

6.0 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 (68 FR 7818). 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.

7.0 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.

Principal Contributors: J. Lee
L. Brown
E. Forrest

Date: April 15, 2003