October 3, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer & President
PSEG Nuclear LLC-X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: INCREASE IN ALLOWABLE MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE RATE AND ELIMINATION OF MSIV SEALING SYSTEM (TAC NO. MB1970)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 134 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 17, 2001, as supplemented on August 6, August 17, and September 12, 2001.

The amendment revises the TSs to permit an increase in the allowable leak rate for the MSIVs and to delete the MSIV Sealing System. These changes are based on the use of an alternate source term and the guidance provided in Regulatory Guide 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

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

Sincerely,

/RA/

Richard B. Ennis, 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. 134 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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ACCESSION NO. ML012600176 * See previous concurrence

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Hope Creek Generating Station

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PSEG NUCLEAR LLC

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
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 May 17, 2001, as supplemented on August 6, August 17, and September 12, 2001, 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 134, 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 during Refueling Outage 10, currently scheduled to commence in October 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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: October 3, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 134

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
xi	xi
xix	xix
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-7	3/4 6-7
3/4 6-19	3/4 6-19
3/4 6-20	3/4 6-20
3/4 6-42	3/4 6-42
B 3/4 1-5	B 3/4 1-5
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
B 3/4 7-1	B 3/4 7-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR, LLC

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated May 17, 2001, as supplemented on August 6, August 17, and September 12, 2001, the PSEG Nuclear LLC (PSEG or the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS) Technical Specifications (TSs). The requested changes would revise the TSs to permit an increase in the allowable leak rate for the Main Steam Isolation Valves (MSIVs) and to delete the MSIV Sealing System (MSIVSS). These changes are based on the use of an alternate source term and the guidance provided in Regulatory Guide (RG) 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

The specific changes requested by the licensee are as follows:

(1) The allowable leak rate in TS 3.6.1.2 would be changed from 46 standard cubic feet per hour (scfh) combined through all four main steamlines to 150 scfh per main steamline and 250 scfh combined through all four main steamlines;

(2) TS 3/4.6.1.4, its associated Bases, and TS Tables 3.3.2.1 and 3.6.3-1 would be revised to eliminate the requirements for the MSIVSS;

(3) The Bases for TS 3.1.5 would be revised to address the use of the standby liquid control system (SLCS) to raise and maintain the long-term post-accident coolant inventory pH levels at 7 or above;

(4) The Bases for TS 3.6.1.2 would be revised to reflect that any MSIV that exceeds the specified leakage limits will be restored to less than or equal to 25 scfh prior to plant restart;

(5) The Bases for TS 3.7.2 would be revised to reflect the use of the total effective dose equivalent (TEDE) acceptance criteria consistent with Section 50.67 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67); and

(6) The TS index would be revised to reflect the above changes.

The letters dated August 6, August 17, and September 12, 2001, provided clarifying information that did not change the scope of the request nor the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Each of the four main steamlines contains two quick-closing MSIVs, one located inside and one located outside of the primary containment. These valves function to isolate the reactor coolant system in the event of a break in a steamline outside the primary containment, a design-basis loss-of-coolant accident (LOCA), or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. Operating experience at various boiling water reactor (BWR) plants has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs, and the specified low leakage limits are difficult to maintain.

Due to recurring problems with excessive leakage of MSIVs, the Nuclear Regulatory Commission (NRC) staff issued RG 1.96, which recommends the installation of a supplemental leakage control system to ensure that the isolation function of the MSIVs complies with the specified leakage limits. To meet this RG, the licensee installed a safety-related MSIVSS that is designed to eliminate the release of fission products through the closed MSIVs that would otherwise bypass the Filtration, Recirculation, and Ventilation System (FRVS) after a LOCA. This is accomplished by pressurizing the sections of the main steamlines between the inboard and outboard MSIVs, and between the outboard MSIVs and the main steam stop valves, to a pressure above that of the reactor pressure vessel. Sealing gas is supplied from two independent primary containment instrument gas (PCIG) receivers. Leakage past the MSIVs is directed back into primary containment where it can be processed as a filtered release and reduce the potential contribution to offsite and control room doses.

As discussed in the licensee’s submittal, the current HCGS TS allowable MSIV leak rate is limiting due to the MSIV leakage problems and routinely requires the repair and retesting of the MSIVs. The submittal also stated that the MSIVSS is a maintenance intensive system. The MSIVSS also affects the PCIG system by allowing leaking steam/moisture into the dry gas system. The licensee concluded that the proposed increase of the MSIV leak rate and deletion of the MSIVSS would reduce outage durations and would reduce radiation exposures to maintenance personnel.

The changes proposed by the licensee are based on full implementation of an alternate source term pursuant to 10 CFR 50.67 using the guidance provided in RG 1.183. As described in RG 1.183, full implementation is a modification of the facility design basis that addresses all characteristics of the alternate source term, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the alternate source term in place of the previous source term and establishes the TEDE acceptance criteria in 10 CFR 50.67 in lieu of the whole body and thyroid dose guidelines provided in 10 CFR 100.11.

3.0 EVALUATION

3.1 Licensee's Technical Analysis

To demonstrate the adequacy of the HCGS engineered safety features (ESF) to mitigate the radiological consequences of design-basis accidents (DBAs) with a maximum combined total MSIV leakage rate of 250 scfh from four main steamlines without relying on the MSIVSS to remove fission products, the licensee recalculated the offsite and control room radiological doses from a postulated LOCA at a reactor core power level of 3,458 megawatts thermal (MWt). This power level is 3.56 percent above the current licensed power level of 3,339 MWt (power level recently implemented following a 1.4-percent power uprate). 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 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 exclusion area boundary (EAB), in the low population zone (LPZ), and the control room.

In its submittal, the licensee concluded that the existing HCGS ESF systems, with the increased MSIV leakage rates and without relying on the MSIVSS, 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) MSIV leakage.

These three potential fission product release pathways are evaluated in Sections 3.2, 3.3, and 3.4 of this Safety Evaluation.

Table 1 summarizes the results of the licensee's radiological consequence calculations, while Tables 2 and 3 list the major parameters and assumptions used by the licensee in its radiological consequence calculations and by the staff in its confirmatory dose calculations.

3.2 Containment Leakage Pathway

The licensee evaluated the radiological consequences resulting from containment leakage following a postulated design basis LOCA at a reactor core power level of 3,458 MWt. The licensee used a containment leak rate of 0.5-percent per day based on the allowable HCGS TS limit for the first 24 hours and a 0.25-percent 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 containment mixes instantaneously and homogeneously

throughout the free air volume of the 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 consists of two ESF subsystems: the FRVS recirculation subsystem (FRVS-RS) and FRVS ventilation subsystem (FRVS-VS). Both of these subsystems are located inside the reactor building and are seismic Category 1 design. 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 25-percent capacity trains, each of which has a flow capacity of 30,000 cubic feet per minute (cfm). Of the six trains, four 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 four trains (90 percent of the rated capacity of each train, or 27,000 cfm each).

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 percent. To simulate the 50-percent air mixing in the reactor building, the licensee doubled the FRVS-VS release rates to the environment. The staff finds that the assumptions used for the air mixing in the reactor building and release rates to the environment are consistent with RG 1.183 and, therefore, are acceptable. In HCGS License Amendment No. 30, issued in August 1989, the staff determined that the overall iodine removal efficiency is 99 percent for the FRVS charcoal adsorbers. This determination was based on the test acceptance criteria in the HCGS TS for the charcoal adsorbent from the two beds in series (FRVS-RS and FRVS-VS). These criteria are 7.5 percent and 1 percent, respectively, corresponding to a combined iodine penetration for the two beds in series of less than 0.075 percent. The licensee's evaluation of radiological consequences for the current license amendment request also used a 99-percent iodine removal efficiency for the FRVS.

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 3.8 of this Safety Evaluation.

3.3 <u>Post-LOCA ESF System Leakage Pathway</u>

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 HCGS Updated Final Analysis Report (UFSAR), the licensee estimated this leakage to be less than 10 gallons per hour (gph), and used that value for the entire duration of the accident (i.e., 30 days).

The licensee assumed that 30 percent 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 percent of the iodine in the liquid leakage becomes airborne, and the airborne iodine is immediately released to the environment. In addition, consistent with RG 1.183, the licensee assumed that radioiodine that is postulated to be available for release to the environment is 97 percent in elemental iodine form and 3 percent in organic iodine form.

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 3.8 of this Safety Evaluation.

3.4 MSIV Leakage Pathway

As previously discussed, HCGS has four main steamlines, 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 steamline outside the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leaktight barrier, the staff recognizes that some leakage occurs through these valves. The current HCGS TS limit for MSIV leakage is 46 scfh combined through all four main steamlines. The licensee assumed a double guillotine pipe rupture in one of the four main steamlines upstream of the inboard MSIV. A total of 250 scfh (the proposed maximum allowable leakage limit) is assumed to occur the following ways: 150 scfh through the broken steamline, 50 scfh through a first intact steamline, the remaining 50 scfh through a second intact steamline, and no leakage from a third intact steamline.

During the postulated LOCA, the main steam leakage flow pattern in the main steamlines could be plug flow, well-mixed flow, or some combination of the two. If temperature gradients exist along the length of the main steamline, then some degree of mixing would occur. For the same leakage rate into the main steamline, 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 steamline. Plug flow effectively results in a longer fission product transport time in the steamline, with more aerosol deposition in the steamlines.

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. 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 steamline. The current RADTRAD code is not capable of calculating aerosol deposition rates under well-mixed flow conditions. In AEB-98-03, the staff performed a Monte Carlo analysis to determine the distribution of aerosol settling velocities in the main steamlines. For the uncertainty analyses, the staff used the ranges and distributions provided in NUREG/CR-6189, “A Simplified Model of 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 40th percentile uncertainty distribution (as the staff justified in AEB-98-03) to calculate the aerosol removal rate using the HCGS specific main steam piping parameters. The portions of the main steam piping that the licensee credited for aerosol removal are classified as seismic Category 1 and are located within the reactor building. The staff finds that the method that the licensee used to calculate aerosol deposition in the main steam pipe is consistent with the method in AEB-98-03 and, therefore, 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. Regulatory Guide 1.183 cites that these models are 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 3.8 of this Safety Evaluation.

3.5 Post-Accident Suppression Pool Water Chemistry Management

In NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the staff concluded that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent cesium iodide (CsI), with no more than 5 percent of iodine (I) and hydriodic acid (HI). Once in the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I^-) in solution. The radiation-induced conversion of iodide in water into elemental iodine (I_2) strongly depends on the pH. The staff stated in NUREG-1465 that, without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be re-evolved into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained above 7, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine.

The licensee performed most of its pH calculation using a proprietary computer code. The licensee provided enough information to permit the staff to perform its independent verification. The licensee calculated pH in the suppression pool after the postulated LOCA by balancing hydroxyl ions (OH^-) and hydrogen ions (H^+) existing in the suppression pool. Since OH^- ions contribute to alkalinity and H^+ ions contribute to acidity of the suppression pool water, their relative amounts determine the value of pH. The licensee assumed that all cesium released to the suppression pool is a form of cesium borate ($CsBO_2$) and is a source of OH^- ions. The amount of cesium borate is proportional to the core cesium inventory and to its fraction released to the suppression pool.

The main source of H^+ ions are nitric and hydrochloric acids (HCl). There is also some contribution from the H^+ ions which exist prior to an accident in the suppression pool water because of its pH value of 5.8, and which come from the HI released from the damaged core. Nitric acid is generated in the radiation field existing in the containment after the accident. The rate of its generation is proportional to the amount of radioactivity released from the damaged core. Hydrochloric acid is generated by a decomposition of cable insulation. The licensee determined that there are 9000 pounds of cable insulation in the HCGS which consists of Hypalon and ethylene-propylene rubber (EPR).

Although EPR normally contains a small amount of chlorine, the licensee conservatively assumed that it will also contribute to the generation of HCl. All the cables in the containment are exposed to the radiation fields existing after an accident. However, some of them are located in conduits and cable trays and are, therefore, shielded from less penetrating beta radiation. The licensee followed the methodologies described in NUREG-1081, "Post-Accident Gas Generation from Radiolysis of Organic Materials," dated September 1984 and NUREG/CR-5950, "Iodine Evolution and pH Control," dated December 1992 in calculating the amount of hydrochloric acid generated from the cables at different times after an accident.

The licensee calculated the pH in the suppression pool for two cases assuming 50 percent and 100 percent of cesium released as cesium borate to the suppression pool. The calculations have indicated that for the first case the pH will stay above 8 for approximately 5 hours and for the second it will remain above 7 for 21 hours after a LOCA. The staff verified these licensee's predictions and find them to be acceptable. These calculations also indicated that the value of pH at 30 days after a LOCA was below 7.

However, in the LOCA analyses it is assumed that sodium pentaborate injected into the reactor vessel for reactivity control will get into the suppression pool and will exercise a buffering effect. The licensee calculated this buffering action of sodium pentaborate. This solution, when introduced into the suppression pool, will produce an initial value for the buffered pH of 8.4. The licensee assumed that, when sodium pentaborate is injected into the reactor vessel, it will also be introduced into the suppression pool and will mix with the suppression pool water in a relatively short time. If the path between the reactor vessel and the suppression pool is more restricted, mixing may take longer, but in all cases the solution will eventually reach the suppression pool water and will exercise its buffering action.

Based on engineering judgment, the staff believes that, for the first 24 hours into the postulated LOCA, the fission product source term behavior, its transport, and release to the environment, will be entirely dominated by thermal hydraulic conditions in the drywell and in the containment (drywell leakage, steam production and condensation, and mixing), and by aerosol removal mechanisms (containment spray and aerosol deposition) independent of suppression pool water pH and iodine reevolution from the suppression pool to the containment atmosphere. Consequently, any postulated radiological consequences at any point on the boundary of the exclusion area for a 24-hour period will not be affected by iodine reevolution and pH control.

The licensee made a reasonable assumption that the buffering action of sodium pentaborate in the HCGS will occur at about 5 hours after the beginning of a LOCA. For the buffered suppression pool water, the licensee calculated pH at different time intervals, conservatively neglecting the effects of hydroxide ions coming from cesium borate and assuming only the presence of hydrogen ions coming from the acids. The calculations have indicated that pH will stay at a value above 8 for the whole 30-day period after a LOCA.

The licensee stated that, in the event of an unmitigated LOCA such as the staff postulated to occur, the HCGS severe accident procedures direct the plant operators to inject the SLCS solution into the reactor vessel in the early stages of the accident (within several hours) for both vessel inventory and re-criticality protection when the core is re-flooded. The SLCS is a safety-related system and designed to seismic Category 1 standards. It is designed as a reactivity control system and provides backup capability so as to be able to shut down the reactor if the normal control becomes inoperable. The HCGS TSs require the system to be maintained in an operable status whenever the reactor is critical. The system is manually initiated from the main control room to add a boron neutron absorber solution (sodium pentaborate) to the reactor vessel. The SLCS contains at least 5,700 pounds of sodium pentaborate. Sodium pentaborate dissolves in water, producing boric acid and sodium borate:

$$Na_2B_{10}O_{16} + 15\ H_2O \rightarrow 2\ Na^+ + 2\ H_2BO_3^- + 8\ H_3BO_3$$

Since boric acid is a relatively weak acid and sodium hydroxide is a strong base, their solution has a buffering effect and will maintain the pH of the suppression pool water at pH values higher than 7. The sodium pentaborate solution will be well mixed with the suppression pool water by the end of 24-hour period as a result of reflooding the reactor vessel. The licensee stated that sodium pentaborate from the SLCS is capable of controlling and maintaining long-term suppression pool water pH levels at 8 or above through the entire 30-day period of the accident. The staff verified the licensee's calculation for buffering action of sodium

pentaborate and found that it was based on sound assumptions. Based on review of the licensee's submittal and the NRC's independent verification, the staff concludes that the HCGS suppression pool water could be maintained at a pH above 8 for over a period of 30 days following the postulated LOCA. Therefore, there will be very little (less than 1 percent) conversion of the dissolved iodine to elemental iodine which will re-evolve into the containment atmosphere.

3.6 Control Room Habitability

The licensee normally maintains the HCGS control room at a slightly positive pressure to prevent the introduction of air into the control room from sources other than the 1,000 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 would be recirculated through the control room emergency filtration (CREF) system at 3,600 cfm with a 1,000 cfm of makeup air. The licensee also assumed 900 cfm of unfiltered air inleakage to the control room beginning 30 minutes into the accident and continuing throughout the 30-day accident period.

The CREF system is an ESF system designed to maintain the control room at a 0.125-inch water gauge positive pressure relative to adjacent areas. The CREF system is a redundant system composed of two subsystems, each of which has a design flow capacity of 4,000 cfm +/- 10 percent. Among other components, each subsystem consists of a high-efficiency particulate air (HEPA) filter, charcoal adsorbers, and a post HEPA filter. Consistent with the current HCGS licensing basis, the licensee's analysis and the staff's evaluation used a conservative recirculating flow rate of 3,600 cfm (i.e., 4000 cfm - 10 percent), and a charcoal filter removal efficiency of 99 percent for iodine in elemental, organic, and particulate forms.

In July 2001, the licensee performed tracer gas testing to verify the unfiltered air inleakage rate into the control room following a design-basis accident. The testing was performed in accordance with ASTM E741-93 with the control room emergency ventilation system in the pressurization and recirculation configurations. The preliminary test results indicated that the unfiltered inleakage flow rates were 196 $\pm$ 10 scfm for the "A" train and 138 $\pm$ 15 scfm for the "B" train. To conservatively bound the measured unfiltered inleakage rates into the control room, the licensee used an 900 cfm unfiltered air inleakage rate to the control room in the revised dose calculation. The results of the licensee's control room radiological consequence calculations are given 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 and 3. The radiological consequences to the control room operator calculated by the licensee and the staff are within the dose criterion specified in 10 CFR 50.67 and, therefore, is acceptable.

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

The licensee used a 7-year period of onsite meteorological data collected at 10 and 45.7 meters during calendar years 1988 through 1994 to calculate relative concentration (X/Q) values. Joint wind speed, wind direction, and atmospheric stability data recovery during this period was very good every year, well above the 90-percent minimum goal cited in RG 1.23,

"Onsite Meteorological Programs." The licensee confirmed that the onsite meteorological measurements program meets the recommendations of RG 1.23 without deviation. The licensee also stated that systems are calibrated and preventative maintenance performed on a quarterly basis according to approved plant procedures. Data are collected and validated according to the procedures using knowledge of an experienced meteorologist. The NRC staff performed a review of the data and found reasonable year to year consistency among the data, and in a comparison with historical data. In particular, wind direction data appeared to be well correlated year to year and between the two measurement heights.

The licensee recalculated X/Q values for the EAB and LPZ using site-specific inputs and the methodology described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Confirmatory estimates performed by the NRC staff using the 1988 through 1994 meteorological data and information provided in the HCGS UFSAR agree reasonably well with the X/Q values calculated by the licensee. Calculations were made for an EAB distance of 901 meters and an LPZ distance of 5 miles.

Control room X/Q values were calculated by the licensee using the ARCON96 methodology (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake.") Calculations were made for several postulated scenarios and the most limiting X/Q values were selected for use in the dose assessments. The X/Q values for containment and ESF leakage from the FRVS were calculated assuming a ground level point release and X/Q values for the MSIV leakage from the turbine building louvers were calculated as a ground level area source from the louver panel nearest the control room intake. Confirmatory estimates performed by the staff agree reasonably well with the X/Q values calculated by the licensee.

The staff has determined that the inputs, assumptions, and methods used by the licensee to calculate the X/Q values for the LOCA dose assessment (shown in Table 3), are acceptable for the reasons discussed above.

3.8 Evaluation Conclusion

The staff reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequences resulting from the postulated LOCA. The doses calculated by the licensee are listed in Table 1. The major parameters and assumptions used by the licensee in its dose calculations and by the staff in its confirmatory dose calculations are listed in Tables 2 and 3. The staff's analysis confirmed the licensee's conclusion that the radiological consequences would not exceed the dose criteria specified in 10 CFR 50.67 for the EAB, the LPZ and the control room. The staff's acceptance is based on our evaluation of the licensee's analyses. The staff confirmed through its own independent evaluation that the results of the licensee's dose calculations are reasonable.

Therefore, the staff concludes that the license amendment requested by the licensee to increase the allowable MSIV leakage rate and to delete the TS requirements for the MSIVSS is acceptable. The bases for the staff's acceptance are that the radiological consequences calculated for the postulated LOCA by both the licensee and the staff are within the dose acceptance criteria specified in 10 CFR 50.67, and the methods and the major parameters and assumptions used in the licensee's dose calculations are consistent with the guidelines provided in RG 1.183 and the staff's technical positions.

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

Release Pathway	EAB	LPZ	Control Room
Containment leak	0.34	0.11	0.43
ESF leak	0.04	0.01	0.26
MSIV leak	1.92	0.37	3.40
Control room filter shine	N/A	N/A	0.003
TOTAL	2.30	0.49	4.1
Dose criteria [2]	25	25	5

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

TABLE 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
for a Loss-of-Coolant Accident

Parameter	Value
Reactor power	3,458 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	0.1 per hour
Reactor building mixing efficiency	50%
FRVS vent exhaust filter efficiencies	
Elemental iodine	99%
Organic iodine	99%
Aerosol (particulate)	99%
FRVS recirculation flow rate	1.08E+5 cfm
ECCS leak rate	10 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-3 meters/second
Aerosol settling area	63.76 m^3
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 inleakage rate into control room	
0 to 30 minutes	500 cfm
30 minutes to 30 days	900 cfm
CREF system filter efficiencies	
Elemental iodine	99%
Organic iodine	99%
Aerosol (particulate)	99%

TABLE 3
Meteorological Data

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-2	1.9E-4

Low Population Zone Distance

Time (hr)	X/Q (sec/m^3)
0-2	1.9E-5
2-4	1.2E-5
4-8	8.0E-6
8-24	4.0E-6
24-96	1.7E-6
96-720	4.7E-7

Control Room
(Containment and ECCS leakage releases via FRVS vent ground level release)

Time (hr)	X/Q (sec/m^3)
0-2	1.25E-3
2-8	8.09E-4
8-24	3.04E-4
24-96	2.10E-4
96-720	1.59E-4

Control Room
(MSIV leakage release via turbine building center louver ground level release)

Time (hr)	X/Q (sec/m^3)
0-2	6.17E-4
2-8	4.00E-4
8-24	1.44E-4
24-96	1.00E-4
96-720	7.49E-5

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. In a letter dated September 14, 2001, the State Official provided the following comments:

> If implemented the resulting change will create an unmonitored, unfiltered containment leakage pathway. By utilizing the alternate source term, recently approved by the NRC for use on a case by case basis, PSEG is able to show that the post accident release through this pathway and other pathways are within NRC limits.
>
> We are concerned that an unmonitored, unfiltered release pathway is being created by this change. We recognize that the resulting analysis show the post accident dose is within limits, however, we question the wisdom of NRC allowing elimination of design features that were intended to limit the release of fission products following an accident, while at the same time, encouraging states to consider implementing of potassium iodide as a supplemental protective action from the release of fission products.
>
> In addition, enhancements to radiation monitoring systems have not been addressed. Finally, models used to calculate projected dose for decision making during emergency preparedness have not been altered to incorporate this unmonitored unfiltered ground level release pathway.

As discussed in a telephone call on September 18, 2001, between Mr. R. Pinney, staff member for the State of New Jersey - Bureau of Nuclear Engineering, and Mr. R. Ennis of the NRC staff, the comment regarding the unmonitored, unfiltered release pathway pertains to statements on page 5 of Attachment 2 of the licensee's submittal dated May 17, 2001. The submittal stated:

> The proposed elimination of the MSIVSS involves additional release paths that were not considered in the current LOCA analysis. Post-LOCA MSIV leakage activity can potentially be released to the environment through the south plant vent (SPV) or the turbine building louver (TBL).

The comments from the State of New Jersey pertain specifically to the TBL release pathway. The licensee's analysis assumes that during post-LOCA conditions (with a concurrent loss of offsite power), MSIV leakage travels through the main steamlines, enters the turbine building, and then releases through openings (louvers) in the turbine building. During normal plant operation, all potentially radioactive air in the turbine building is assumed to be exhausted through the monitored SPV.

The methods used by the licensee in evaluating the post-LOCA radiological consequences due to leakage through the MSIVs is consistent with the model developed and used by the staff in its approval of a similar license amendment request for the Perry Nuclear Power Plant. For HCGS, the staff concludes that the proposed increase in allowable MSIV leak rate and

deletion of the MSIVSS is acceptable because the radiological consequences calculated for the postulated LOCA (total for all pathways) are within the dose acceptance criteria specified in 10 CFR 50.67, and the methods and the major parameters and assumptions used in the licensee's dose calculations are consistent with the guidelines provided in RG 1.183.

In a final rule published in the *Federal Register* on January 19, 2001, the NRC amended the emergency planning regulations specified in 10 CFR 50.47. Section 50.47 establishes requirements for nuclear power plant emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Section 50.47(b) contains 16 planning standards; and, in particular, Section 50.47(b)(10) requires that emergency plans include "a range of protective actions" for the plume exposure pathway emergency planning zone for emergency workers and the public. The final rule amended 10 CFR 50.47(b)(10) to add the following sentence:

> In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate.

The use of KI is intended to supplement, not to replace, other protective measures, such as evacuation, which the Commission continues to view as the primary and most desirable measure in the event of a radiological emergency. The Commission recognizes the supplemental value of KI and the prerogative of the State to decide on the use of KI by its citizens. The staff does not believe that the proposed deletion of the MSIVSS at HCGS is contrary in any respect to the staff's positions regarding the consideration of the use of KI by the State.

With respect to the State comment that enhancements to radiation monitoring systems have not been addressed, the staff did not identify any regulatory requirements or changes to the licensing basis that would require any changes to the radiation monitoring systems.

With respect to the State comment that the models used to calculate projected dose for decision making during emergency preparedness have not been altered to incorporate this unmonitored unfiltered ground level release pathway, the licensee has indicated that the associated changes to the emergency plan do not require NRC review and approval. Pursuant to 10 CFR 50.54q, licensees may make changes to emergency plans if the changes do not decrease the effectiveness of the plans.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 34288). Accordingly, the amendment meets the

eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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