

November 5, 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, CORRECTION TO SAFETY EVALUATION RELATED TO AMENDMENT NO. 134, INCREASE IN ALLOWABLE MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE RATE AND ELIMINATION OF MSIV SEALING SYSTEM (TAC NO. MB3092)

Dear Mr. Keiser:

On October 3, 2001, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 134 to Facility Operating License (FOL) No. NPF-57 for the Hope Creek Generating Station (HCGS). The amendment revised the Technical Specifications to permit an increase in the allowable leak rate for the MSIVs and to delete the MSIV Sealing System. These changes were 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."

Following receipt of Amendment No. 134, your staff verbally informed the NRC of several errors in the NRC's Safety Evaluation (SE) which was enclosed with the amendment. In order to correctly reflect the current licensing basis for HCGS, the NRC has revised the SE as follows:

- (1) Page 5, 1st paragraph - The SE stated that the licensee estimated the engineered safety features (ESF) leak rate "to be less than 10 gallons per hour (gph)." The 10 gph value was a typographical error. The staff's evaluation was based on a value of 10 gallons per minute (gpm) consistent with the licensee's submittal and as shown in Table 2 of the SE. Page 5 of the SE has been revised to show this value as 10 gpm.
- 2) SE page 5, 4th paragraph - The SE stated that the licensee assumed a double guillotine pipe rupture in one of the four main steamlines upstream of the inboard MSIV. This assumption is not applicable to the Hope Creek analysis. Since the staff's evaluation was based on the TS maximum allowable leakage through the MSIVs (250 standard cubic feet per hour) and not the type of pipe rupture, this error has no impact on the staff's conclusions. The paragraph has been revised accordingly.
- 3) SE page 6, 3rd paragraph - The SE stated that 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 SE has been revised to clarify that this piping is also located in the turbine building and that all the piping is designed to remain functional during and after a safe-shutdown earthquake. This clarification does not affect the staff's conclusion with respect to aerosol removal since the staff's evaluation was based on an aerosol settling area of 63.76 cubic meters consistent with the licensee's submittal and as shown in Table 2 of the SE.

H. Keiser

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A copy of revised pages 5 and 6 of the SE for Amendment No. 134 are enclosed. All changes are indicated by marginal bars. The NRC staff has determined that the corrections to the original SE do not change our previous conclusions regarding the acceptability of the changes approved in Amendment No. 134 to FOL No. NPF-57 (reference ADAMS Accession No. ML012600176).

We apologize for any inconvenience these errors may have caused you.

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

Enclosure: SE Pages 5 and 6

cc w/encl: See next page

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Hope Creek Generating Station

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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 minute (gpm), 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's analysis assumed leakage through three of the four main steamlines. A total of 250 scfh (the proposed maximum allowable leakage limit) is assumed to occur the following ways: 150 scfh through the steamline with the failed MSIV, 50 scfh through a first intact steamline, and the remaining 50 scfh through a second 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 hygroscopicity; 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 designed to remain functional during and following a safe-shutdown earthquake and are located within the reactor building and the turbine 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.