August 28, 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: EXCESS FLOW CHECK VALVE TESTING REQUIREMENTS (TAC NO. MB1723)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 132 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 April 11, 2001, as supplemented on June 13, 2001.

The amendment revises the HCGS TSs to relax the frequency for testing of excess flow check valves (EFCVs). Specifically, TS surveillance requirement 4.6.3.4 has been changed to revise required testing of EFCVs from once per 18 months for all valves to a test of a representative sample each 18 months such that all valves are tested once in 10 years. Your application also included an associated request for relief from the requirements of the Inservice Testing Program. That request is being reviewed separately under TAC No. MB1724.

Your letter dated May 17, 2001, which provided supplemental information for the license change request, was not submitted under oath or affirmation in accordance with the requirements in 10 CFR 50.30(b). This was discussed with Mr. John Nagle of your staff on May 22, 2001. The same supplemental information was resubmitted under oath or affirmation by your letter dated June 13, 2001.

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. 132 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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Division of Licensing Project Management
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cc w/encls: See next page

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Hope Creek Generating Station

cc:

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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. 132
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 April 11, 2001, as supplemented June 13, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No.   , 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 RCroteau 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: August 28, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 132

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
3/4 6-18	3/4 6-18
B 3/4 6-5	B 3/4 6-5
B 3/4 6-6	B 3/4 6-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 132 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 April 11, 2001, as supplemented June 13, 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 surveillance test requirements for excess flow check valves (EFCVs). The June 13, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

2.1 Purpose and Function of EFCVs

EFCVs are installed in boiling water reactor (BWR) instrument lines penetrating the primary containment boundary to limit the release of fluid in the event of an instrument line break. Regulatory Guide (RG) 1.11, "Instrument Lines Penetrating Primary Reactor Containment," provides guidance on the implementation of General Design Criteria (GDC) 55 and 56, of Appendix A to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, for instrumentation lines that penetrate primary reactor containment and are part of the reactor coolant pressure boundary. As stated by RG 1.11, EFCVs in combination with flow restricting features (line size or orifice) satisfy the requirements of GDC 55 and 56 for automatic isolation capability, maintain the reliability of the connected instrumentation, and ensure the functional performance of secondary containment in the event of an instrumentation line break. Examples of EFCV installations include reactor pressure vessel (RPV) level and pressure instrumentation, main steamline flow instrumentation, recirculation pump suction pressure, and reactor core isolation cooling steamline flow instrumentation. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post loss-of-coolant accident (LOCA) conditions.

HCGS TS Surveillance Requirement (SR) 4.6.3.4 currently requires that each reactor instrumentation line EFCV be demonstrated to be operable at least once per 18 months by verifying that the valve checks flow. The proposed change revises TS SR 4.6.3.4 to relax the required testing of all EFCVs each 18 months to a test of a "representative sample" of EFCVs each 18 months. As discussed in the licensee's submittal and in the proposed changes to the

TS Bases for TS 3/4.6.3, the "representative sample" would consist of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal).

2.2 Topical Report NEDO-32977-A

The basis for the licensee's request is the high degree of reliability shown by the EFCVs and the low consequences of an EFCV failure. The supporting analysis for the licensee's conclusion is based on General Electric Nuclear Energy (GENE) Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000. The topical report provided: (1) estimate of steam release frequency into the reactor building due to a break in an instrument line concurrent with an EFCV failure to close, and (2) assessment of the radiological consequences of such a release. The topical report concluded that the EFCV test interval could be extended up to 10 years based on the topical report reliability and consequence analysis without significantly affecting plant risk. The topical report suggested a staggered test interval based on actual valve performance with each valve being tested at least once every 10 years. The staff accepted the generic applicability of the topical report by a safety evaluation report (SER) dated March 14, 2000, and agreed that the EFCV test interval could be extended to as much as 10 years. The staff also noted that licensees adopting the topical report must have a failure feedback mechanism and corrective action program to ensure that EFCV performance continues to be bounded by the topical report results. Additionally, each licensee is required to perform a plant-specific radiological dose assessment and EFCV failure rate and release frequency analysis to confirm that their facility is bounded by the generic analysis of the topical report.

2.3 TSTF-334

The proposed change adopts the staff's approved Technical Specification Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valves Testing." TSTF-334 was approved by the staff on October 31, 2000, by letter from W. D. Beckner (NRC) to A. R. Pietrangelo (Nuclear Energy Institute). The TSTF proposed specific changes to the Standard Technical Specifications (STS), providing guidance for licensee's implementing the extended EFCV surveillance test intervals proposed in the topical report. TSTF-334 is applicable only for those plants for which NEDO-32977-A is applicable and are subject to EFCV performance and corrective action criteria to be developed by the licensee.

3.0 EVALUATION

The staff reviewed the licensee's submittal for conformance to the March 14, 2000, staff SER for Topical Report NEDO-32977-A, and the guidance of approved TSTF-334, Revision 2. As detailed below, the staff's evaluation focused on the following areas: (1) EFCV failure rate and release frequency, (2) failure feedback mechanism and corrective action program, (3) operational impact, (4) radiological consequences, and (5) conformance of the proposed TS to generic TSTF guidance.

3.1 <u>EFCV Failure Rate and Release Frequency</u>

In the topical report, EFCV reliability was evaluated based on testing experience provided by 12 different BWR plants. The composite data indicated that EFCVs are very reliable. The data represented 12,418.5 valve years of operation with a total of 11 failures noted. The EFCV composite failure rate was 1.67E-07/hour and was referenced as the "upper limit" failure rate in the topical report.

The staff noted in its review that the topical report assumed the EFCV failure rate was constant over time and did not account for potential age-related degradation. Additionally, the staff questioned the use of an instrument line break frequency based on NRC report WASH-1400, "Reactor Safety Study: Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," published in 1974, and not on more current data. To address these concerns, the Boiling Water Reactor Owners' Group (BWROG) response included an updated instrument line failure frequency of 3.52E-05 failures/year based on Electric Power Research Institute (EPRI) Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants," dated July 1992. This value is 6.6 times greater than the value calculated in the GENE topical report using WASH-1400 data. The BWROG response also assumed the observed EFCV failures were five times the actual observed number listed in the topical report (i.e., 55 versus 11). The additional impact of an increase in instrument line failure frequency and a fivefold increase in EFCV failures assumed by the BWROG response demonstrated that release frequencies remained low with limited impact on release frequency even with significantly different assumptions on break frequency and valve failure rates.

To estimate the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency downstream of the EFCV, and (2) the probability of the EFCV failing to close. The HCGS data was found to be consistent both in time sampled and EFCV reliability (1 EFCV failure, 106 valves, and 1,590 valve years operating time) when compared to the topical report data. Employing the updated EPRI instrument line failure rate to HCGS plant specific data, the 18-month and 10-year plant release frequency is estimated at 8.3E-06 release/year and 5.6E-05 release/year, respectively. The 10-year release frequency shows an increase of 4.7E-05 release/year over the 18-month value. This represents the increase in the total plant release frequency for a random break of any of the 106 instrument lines with a concurrent failure of the EFCV to isolate the break. The increase for HCGS is consistent with the staff's SER on NEDO-32977-A, which concluded that an increase in release frequency of 7.3E-05 release/year was not significant. The HCGS plant-specific EFCV failure data and release rates are also comparable with industry data and the results given in the topical report. Based on the above, the staff does not consider the HCGS increase in estimated release frequency for a 10-year surveillance interval to be significant.

3.2 <u>Failure Feedback Mechanism and Corrective Action Program</u>

The staff's SER noted that the GENE topical report does not provide a specific failure feedback mechanism, but does state that a plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. During review of the topical report, the BWROG responded to the staff's question concerning failure feedback by stating that each licensee who adopts the relaxed surveillance intervals recommended by the topical report should ensure that an appropriate feedback mechanism responsive to EFCV failure trends is in place.

The licensee's submittal dated June 13, 2001, stated that the HCGS Maintenance Rule (10 CFR 50.65) Program will be revised to provide a means to track the performance of the EFCVs. Any future failure will be evaluated per the HCGS Corrective Action Program. To ensure EFCV performance remains consistent with the extended test interval, minimum performance criteria has been established by the licensee. Performance criteria for reactor instrument line EFCVs has been established by HCGS such that additional testing of EFCVs will be required in the case of one failure of an EFCV to check flow. This will ensure that EFCV performance remains consistent with the extended surveillance interval assumptions and that adverse trends in EFCV performance are identified. The staff considers the licensee's program to account for potential changes in EFCV failure rates to be acceptable and satisfies TSTF-334 performance and corrective action criteria.

3.3 Operational Impact

The operational impact of an EFCV failing to close, after the break in an instrument line connected to the RPV boundary, is based on the environmental effects of a steam release in the vicinity of the instrument racks in the reactor building. The topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of the reactor building ventilation systems and that the integrity and functional performance of secondary containment and the Standby Gas Treatment System following an instrument line break would continue to be met. The licensee confirmed that if an EFCV should fail, the restricting orifice or line restriction limits the steam release and the integrity and functional performance of secondary containment and the Filtration, Recirculation, and Ventilation System (FRVS) will be maintained. Section 15.6.2.2 of the HCGS Updated Safety Analysis Report (UFSAR) identifies the operator actions in response to an instrument line break including the manual isolation of the affected instrument line, initiation of the FRVS, and plant shutdown to terminate the event. The separation of divisional instrument lines and equipment in the reactor building is expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. The staff finds that the operational impact of an EFCV failing to close after an instrument line break has acceptably been addressed by the existing HCGS plant design and operation.

3.4 Radiological Consequences

The radiological consequences for an instrument line break have been previously evaluated by the licensee in HCGS UFSAR Section 15.6.2.5. The analysis does not credit the EFCVs for isolating the break and assumes a discharge of reactor water through an instrument line with a 1/4-inch restricting orifice (or line restriction) for the duration of the event. The resulting offsite exposures are a small fraction of the 10 CFR Part 100 limits. As a result, a failure of an EFCV to close is bounded by the licensee's previous analysis. The existing radiation dose consequences for an instrument line break are, therefore, not impacted by the proposed change.

3.5 Conformance of the Proposed TS to Generic TSTF Guidance

The licensee proposed to revise SR 4.6.3.4 to read, "At least once per 18 months, verify that a representative sample of reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 actuates to the isolation position on a simulated instrument line break signal." The term "representative sample," as proposed by the topical report and TSTF-334 is not defined in the TS itself. However, in response to the staff's question on this issue, the BWROG stated that the term "representative sample" with an accompanying explanation in the TS Bases, is identical to the current usage in the STS, NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term "representative" in SR 3.8.6.3 in reference to battery cell testing, and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. The criterion for "representative sample" and the basis for the normal 10-year testing interval are provided in the licensee's submittal, which are similar to Insert 1 and Insert 2 stated in the staff's approved TSTF-334, Revision 2. Therefore, the application of a "representative sample" for the EFCV testing SR, with an accompanying explanation in the TS Bases, is consistent with TSTF-334, Revision 2, to the STS usage and, is therefore, acceptable to the staff.

The licensee included in its submittal, for information, a revised Bases for SR 4.6.3.4 that included a discussion of the EFCV test frequency and the term "representative sample." The Bases for SR 4.6.3.4 included the following insert:

> Surveillance 4.6.3.4 requires demonstration that a representative sample of reactor instrumentation line excess flow check valves are tested to demonstrate that the valve actuates to check flow on a simulated instrument line break. This surveillance requirement provides assurance that the instrument line EFCV's will perform so that the predicted radiological consequences will not be exceeded during a postulated instrument line break event as evaluated in the UFSAR. The 18-month frequency is based on the need to perform this surveillance under the conditions that apply immediately prior to and during the plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. The representative sample consists of an approximately equal number of EFCV's, such that each EFCV is tested at least once every ten years (nominal). In addition, the EFCV's in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potential common problem with a specific type or application of EFCV is detected at the earliest possible time. The nominal 10 year interval is based on performance testing as discussed in NEDO 32977-A, "Excess Check Valve Testing Relaxation." Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

The staff finds that the licensee's proposed revisions to SR 4.6.3.4 and the associated Bases are consistent with TSTF-334, Revision 2.

3.6 Evaluation Summary

As described in the preceding evaluation, the staff finds that: (1) The HCGS plant-specific EFCV failure data and release rates are comparable with industry data and the results given in Topical Report NEDO-32977-A; (2) the increase in estimated release frequency for HCGS for a 10-year surveillance interval is not considered by the staff to be significant; (3) the proposed amendment is consistent with TSTF-334, Revision 2, and Topical Report NEDO-32977-A; (4) the licensee has an acceptable program to account for potential changes in EFCV failure rates; (5) the operational impact of an EFCV failing to close after an instrument line break has acceptably been addressed by the existing HCGS plant design and operation; and (6) the radiological consequences of an EFCV failing to close after an instrument line break is bounded by the licensee's previous analysis. Based on these findings, the staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 29361). 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.

Principal Contributors: C. Doutt
N. Le

Date: August 28, 2001