UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 1, 1995

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION (TAC NO. M89426)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 76 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 April 25, 1994, as supplemented July 24, 1995.

This amendment adds a new note (Note 11) to supplement Note 4 of Technical Specification Table 3.6.3-1, "Primary Containment Isolation Valves." With the addition of this new note, the Technical Specifications will be revised to indicate that 10 CFR Part 50, Appendix J, Type C leak testing is not necessary for certain Containment Isolation Valves, which serve lines terminating below the minimum water level in the suppression chamber (i.e., torus).

The amendment eliminates the requirement from the Hope Creek technical specifications to perform Type C leak rate tests, in accordance with 10 CFR Part 50, Appendix J, of identified containment isolation valves that penetrate the primary containment and terminate below the minimum water level in the suppression chamber (torus). The valves would still be subject to testing in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

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

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You are requested to inform the NRC, in writing, when this amendment has been implemented.

This requirement affects fewer than nine respondents and, therefore, it is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

David H. Jaffe, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-354

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

cc w/encls: See next page

DISTRIBUTION

Docket File	JStolz	GHill(2)	JWhite, RGN-1
PUBLIC	MO'Brien	Tech Branch	
PDI-2 Reading	DJaffe	CGrimes	
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*Previously Concurred

OFC	:PDI-2/LA	:PDI-2/PM	:OGC*	:PDI-2/D	:	:
NAME	:MO'Brien	:DJaffe:rb	:	:JStolz	:	:
DATE	:7/31/95	:7/28/95	:07/07/95	:7/31/95	:	:

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Docket No. 50-354

Enclosures: 1. Amendment No. 76 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

Mr. Leon R. Eliason
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Company

Hope Creek Generating Station

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated April 25, 1994, as supplemented July 24, 1995, 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. 76 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G 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

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: August 1, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 6-20	3/4 6-20
3/4 6-21	3/4 6-21
3/4 6-29	3/4 6-29
3/4 6-30	3/4 6-30
3/4 6-31	3/4 6-31
3/4 6-35	3/4 6-35
3/4 6-36	3/4 6-36
3/4 6-42	3/4 6-42

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) MSIV Sealing System Isolation Valves				M-72-1
Outside:				
Line A HV-5834A (KP-V010)	P1A	45	1	
Line B HV-5835A (KP-V009)	P1B	45	1	
Line C HV-5836A (KP-V008)	P1C	45	1	
Line D HV-5837A (KP-V007)	P1D	45	1	
2. Group 2 - Reactor Recirculation Water Sample System				
(a) Reactor Recirculation Water Sample Line Isolation Valves				M-43-1
Inside: BB-SV-4310	P17	15	3	
Outside: BB-SV-4311	P17	15	3	
3. Group 3 - Residual Heat Removal (RHR) System				
(a) RHR Suppression Pool Cooling Water & System Test Isolation Valves				M-51-1
Outside:				
Loop A: HV-F024A (BC-V124)	P212B	180	11	
HV-F010A (BC-V125)	P212B	180	11	
Outside:				
Loop B: HV-F024B (BC-V028)	P212A	180	11	
HV-F010B (BC-V027)	P212A	180	11	
(b) RHR to Suppression Chamber Spray Header Isolation Valves				M-51-1
Outside:				
Loop A: HV-F027A (BC-V112)	P214B	75	3	
Loop B: HV-F027B (BC-V015)	P214A	75	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) RHR Shutdown Cooling Suction Isolation Valves				M-51-1
Inside: HV-F009 (BC-V071)	P3	45	3	
Outside: HV-F008 (BC-V164)	P3	45	3	
(d) RHR Head Spray Isolation Valves				M-51-1
Inside: HV-F022 (BC-V021)	P10	60	3	
Outside: HV-F023 (BC-V020)	P10	60	3	
(e) RHR Shutdown Cooling Return Isolation Valves				M-51-1
Outside:				
Loop A: HV-F015A (BC-V110)	P4B	45	3	
Loop B: HV-F015B (BC-V013)	P4A	45	3	
4. Group 4 - Core Spray System Outside:				
(a) Core Spray Test to Suppression Pool Isolation Valves				M-52-1
Loop A: HV-F015A (BE-V025)	P217B	80	11	
Loop B: HV-F015B (BE-V026)	P217A	80	11	
5. Group 5 - High Pressure Coolant Injection (HPCI) System				
(a) HPCI Turbine Steam Supply Isolation Valves				M-55-1
Inside: HV-F002 (FD-V001)	P7	NA	3	
HV-F100 (FD-V051)	P7	NA	3	
Outside: HV-F003 (FD-V002)	P7	NA	3	
(b) HPCI Pump Suction Isolation Valve				M-55-1
Outside:				
HV-F042 (BJ-V009)	P202	NA	11	

TABLE 3.6.3-1 (Continued)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(b) DLD-RMS Return Isolation Valves				M-25-1
Outside:				
HU-4957 (SK-V008)	J5A	45	3	
HV-4981 (SK-V009)	J5A	45	3	
B. Remote Manual Isolation Valves				
1. Group 21 - Feedwater System				
(a) Feedwater Isolation Valves				M-41-1
Outside Check Valves				
HV-F032B (AE-V001)	P2A	NA	2	
HV-F032A (AE-V005)	P2B	NA	2	
(b) Reactor Water Cleanup System Return				
Outside:				
HV-F039 (AE-V021)	P2A&B	NA	2	M-44-1
2. Group 22 - High Pressure Coolant Injection (HPCI) System				
(a) Core Spray Discharge Valve				
Outside:				
HV-F006 (BJ-V001)	P5B	NA	3	M-55-1
(b) Turbine Exhaust Valve				
Outside:				
HV-F071 (FD-V006)	P201	NA	4	M-55-1
(c) HPCI Minimum Return Line Valve				
Outside:				
HV-F012 (BJ-V016)	P203	NA	11	M-55-1
(d) Feedwater Line Discharge Valve				
Outside:				
HV-8278 (BJ-V059)	P2B	NA	2	M-55-1
3. Group 23 - Reactor Core Isolation Cooling (RCIC) System				
(a) RCIC Turbine Exhaust Valve				
Outside:				
HV-F059 (FC-V005)	P207	NA	4	M-49-1

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside:				
(b) RCIC Pump Suction Isolation Valve HV-F031 (BD-V003)	P208	NA	11	M-49-1
Outside:				
(c) RCIC Minimum Return Line Isolation Valve SV-F019	P209	NA	11	M-49-1
Outside:				
(d) RCIC Vacuum Pump Discharge HV-F060 (FC-V011)	P210	NA	4	M-49-1
(e) Feedwater Line Discharge Valve				
Outside:				
HV-F013 (BD-V005)	P2A	NA	2	M-49-1
4. Group 25 - Core Spray System				
(a) Core Spray injection Valves				M-52-1
Outside:				
Loop A&C HV-F005A (BE-V007)	P5B	NA	3	
Loop B&D HV-F005B (BE-V003)	P5A	NA	3	
(b) Core Spray Suppression Pool Suction Valves				M-52-1
Outside:				
Loop A HV-F001A (BE-V017)	P216D	NA	11	
Loop B HV-F001B (BE-V019)	P216A	NA	11	
Loop C HV-F001C (BE-V018)	P216C	NA	11	
Loop D HV-F001D (BE-V020)	P216B	NA	11	
(c) Core Spray Minimum Flow Valves				M-52-1
Outside:				
Loop A&C HV-F031A (BE-V035)	P217B	NA	11	
Loop B&D HV-F031B (BE-V036)	P217A	NA	11	
(d) Core Spray Injection Line Bypass Valves				M-52-1
Inside:				
HV-F039A (BE-V071)	P5B	NA	3	
HV-F039B (BE-V072)	P5A	NA	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
5. Group 26 - Residual Heat Removal System				
(a) Low Pressure Coolant Injection Valves				M-51-1
Outside:				
Loop A: HV-F017A (BC-V113)	P6C	NA	3	
Loop B: HV-F017B (BC-V016)	P6B	NA	3	
Loop C: HV-F017C (BC-V101)	P6D	NA	3	
Loop D: HV-F017D (BC-V004)	P6A	NA	3	
(b) RHR Containment Spray				M-51-1
Outside:				
Loop A: HV-F021A (BC-V116)	P24B	NA	3	
HV-F016A (BC-V115)	P24B	NA	3	
Loop B: HV-F021B (BC-V019)	P24A	NA	3	
HV-F016B (BC-V018)	P24A	NA	3	
(c) RHR Suppression Pool Suction				M-51-1
Outside:				
Loop A: HV-F004A (BC-V103)	P211C	NA	11	
Loop B: HV-F004B (BC-V006)	P211B	NA	11	
Loop C: HV-F004C (BC-V098)	P211D	NA	11	
Loop D: HV-F004D (BC-V001)	P211A	NA	11	
(d) RHR Minimum Flow Isolation Valves				M-51-1
Outside:				
Loop A: HV-F007A (BC-V128)	P212B	NA	11	
Loop B: HV-F007B (BC-V031)	P212A	NA	11	
Loop C: HV-F007C (BC-V131)	P212B	NA	11	
Loop D: HV-F007D (BC-V034)	P212A	NA	11	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
5. Group 35 - Breathing Air System				M-15-1
Inside KG-V016	P31	NA	3	
Outside KG-V034	P31	NA	3	
6. Group 36 - TIP Purge System				
Inside:				
Check Valve: SE-V006	P34F	NA	3	M-59-1
7. Group 37 - HPCI System				
Outside:				
HPCI Turbine Exhaust: FD-V004	P201	NA	4	M-55-1
8. Group 38 - RCIC System				
Outside:				
RCIC Turbine Exhaust: FC-V003	P207	NA	4	M-49-1
Vacuum Pump Discharge: FC-V010	P210	NA	4	M-49-1
9. Group 39 - RHR System				
(a) Thermal Relief Valves				M-51-1
Outside:				
Loop A: BC-PSV-F025A	P212B	NA	5	
Loop B: BC-PSV-F025B	P212A	NA	5	
Loop C: BC-PSV-F025C	P212B	NA	5	
Loop D: BC-PSV-F025D	P212A	NA	5	
(b) Jockey Pump Discharge Check Valves				M-51-1
Outside:				
Loops A & C: (BC-V206)	P212B	NA	11	
Loops B & D: (BC-V260)	P212A	NA	11	
(c) RHR Heat Exchanger Thermal Relief Valves				M-51-1
Outside:				
BC-PSV-4431A	P213B	NA	5	
BC-PSV-4431B	P213A	NA	5	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(d) RHR Shutdown Cooling Suction Thermal Relief Valve				M-51-1
Inside:				
BC-PSV-4425	P3	NA	3	
(e) LPCI Injection Line Check Valves				M-51-1
Inside:				
HV-F041A (BC-V114)	P6C	NA	3	
HV-F041B (BC-V017)	P6B	NA	3	
HV-F041C (BC-V102)	P6D	NA	3	
HV-F041D (BC-V005)	P6A	NA	3	
(f) Shutdown Cooling Return Line Check Valves				M-51-1
Inside:				
HV-F050A (BC-V111)	P4B	NA	3	
HV-F050B (BC-V014)	P4A	NA	3	
(g) RHR Suppression Pool Return Valves				M-51-1
Outside:				
HV-F011A (BC-V126)	P212B	NA	11	
HV-F011B (BC-V026)	P212A	NA	11	
10. Group 40 - Core Spray System				
(a) Thermal Relief Valves				M-52-1
Outside:				
Loop A&C: BE-PSV-F012A	P217B	NA	5	
Loop B&D: BE-PSV-F012B	P217A	NA	5	
(b) Core Spray Injection Line Check Valves Inside:				M-52-1
HV-F006A (BE-V006)	P5B	NA	3	
HV-F006B (BE-V002)	P5A	NA	3	
11. Group 41 - Drywell Pressure Instrumentation				M-42-1
Outside:				
BB-V563	J6A	NA	6	
BB-V564	J8D	NA	6	
BB-V565	J7A	NA	6	
BB-V566	J10D	NA	6	

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

NOTES

NOTATION

1. Main Steam Isolation Valves are sealed with a seal system that maintains a positive pressure of 5 psig above reactor pressure. Leakage is inleakage and is not added to 0.60 La allowable leakage.*

2. Containment Isolation Valves are sealed with a water seal from the HPCI and/or RCIC system to form the long-term seal boundary of the feedwater lines. The valves are tested with water at 1.10 Pa, 52.9 psig, to ensure the seal boundary will prevent by-pass leakage. Seal boundary liquid leakage will be limited to 10 gpm.

3. Containment Isolation Valve, Type C gas test at Pa, 48.1 psig. Leakage added to entire system leakage. Allowable leakage for entire system limited to 0.60La.

4. Containment Isolation Valve, Type C water test at 1.10 Pa, 52.9 psig delta P. Leakage added to entire system leakage. Allowable leakage for entire system limited to 10 gpm.

5. Containment boundary is discharge nozzle of relief valve, leakage tested during Type A test.*

6. Drywell and suppression chamber pressure and level instrument root valves and excess flow check valves, leakage tested during Type A.*

7. Explosive shear valves (SE-V021 through SE-V025) not Type C tested.*

8. Surveillances to be performed per Specification 3.6.1.8.

9. All valve I.D. numbers are preceded by a numeral 1 which represents an Unit 1 valve.

10. The reactor vessel head seal leak detection line (penetration J5C) excess flow check valve (BB-XV-3649) is not subject to OPERABILITY testing. This valve will not be exposed to primary system pressure except under the unlikely conditions of a seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source; therefore, this valve need not be OPERABILITY tested.

11. Containment Isolation Valve(s) are not Type C tested. Containment by-pass leakage is prevented since the line terminates below the minimum water level in the suppression chamber and the system is a closed system outside Primary Containment. Refer to Specification 4.0.5.

*Exemption to Appendix J of 10 CFR Part 50.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

In a submittal dated April 25, 1994, as supplemented July 24, 1995, the licensee for Hope Creek Generating Station requested an amendment to Technical Specifications (TSs) Table 3.6.3-1, "Primary Containment Isolation Valves." The licensee requested elimination of 10 CFR Part 50 Appendix J, Type C leak rate testing for certain containment isolation valves (CIVs), which are located in lines that penetrate the primary containment and terminate below the minimum water level in the suppression pool, by deleting Note 4 and adding a new Note 11 for these valves. Note 4 requires Type C water test at 1.10 Pa, and leakage added to 10 gpm allowable leakage. Note 11 clarifies that Type C testing is not required for certain CIVs because containment bypass leakage is prevented since the line terminates below the minimum water level in the suppression pool and the system is a closed system outside primary containment. Note 11 also refers to existing ASME Section XI requirements in the TS (4.0.5) for performance of inservice testing.

The licensee indicated that this amendment request is not an exemption to 10 CFR Part 50, Appendix J requirements. Appendix J, Type C testing is not appropriate for certain CIVs on lines which penetrate the suppression pool (i.e., torus) and terminate below the minimum water level. Inservice testing will be performed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code, Section XI - Division 1, Article IWV-3000.

The licensee has proposed specific changes for the following valves:

- RHR Suppression Pool Cooling Water & Test Isolation Valves BC-V124, V125, V028 and V027, Penetrations P212B and P212A.

- Core Spray Test to Suppression Pool Isolation Valves BE-V025 and V026, Penetrations P217B and P217A.

- HPCI Pump Suction Isolation Valve BJ-V009, Penetration P202.

- HPCI Minimum Return Line Isolation Valve BJ-V016, Penetration P203.

- RCIC Pump Suction Isolation Valve BD-V003, Penetration P208.

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- RCIC Minimum Return Line Isolation Valve SV-F019, Penetration P209.
- Core Spray Suppression Pool Suction Valves BE-V017, V019, V018 and V020, Penetrations P216D, P216A, P216C and P216B.
- Core Spray Minimum Flow Valves BE-V035 and BE-V036, Penetrations P217B and P217A.
- RHR Suppression Pool Suction Valves BC-V103, V006, V098, and V001, Penetrations P211C, P211B, P211D and P211A.
- RHR Minimum Flow Isolation Valves BC-V128, V031, V131, V034, Penetrations P212B and P212A.
- RHR System Jockey Pump Discharge Check Valves BC-V206 and V260, Penetrations P212B and P212A.
- RHR Suppression Pool Return Valves BC-V126 and V026, Penetrations P212B and P212A.

2.0 EVALUATION

The containment isolation valves for which Appendix J, Type C leak rate testing will not be performed are all on lines which penetrate the torus and terminate below the torus minimum water level. Since the torus is designed and operated to be filled with water during and following any postulated design basis accident (DBA), the CIVs will remain water sealed during these conditions. This prevents the primary reactor containment atmosphere from impinging on the CIVs and precludes its leakage out of containment post-accident.

The licensee indicated that the subject CIVs are located in systems for which the system and the associated piping are protected against missiles and pipe whip, are designed to seismic Category I requirements, and are classified as Quality Group B per Regulatory Guide 1.26 (RG 1.26). The systems and piping will not be adversely affected by single active failures.

The torus will remain filled with water for at least 30 days following the onset of an accident. The subject CIVs, therefore, do not constitute potential containment atmosphere leak paths, and as such are not required by Paragraph III.A.1.(d) of Appendix J to be Type C tested. Additionally, in accordance with Sections III.C.2 and III.C.3 of Appendix J, the CIVs need not be tested with air. Further, it is not necessary to test them with water, as the purpose of the water leak rate test is to assure a supply of sealing water for 30 days following an onset of an accident. As the torus is postulated to always remain filled with water, no leak rate test is necessary to satisfy Appendix J requirements. The CIVs will, however, continue to be tested pursuant to the applicable requirements of Section XI of the ASME B&PV Code.

For the above reasons, the staff finds the proposed testing of the CIVs in the above penetrations to be in compliance with the requirements of Appendix J.

3.0 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 June 28, 1994, the New Jersey State Official provided the following comments.

> PSE&G stated in their letter that " This request is based upon an exemption issued by the NRC on October 30, 1986 for Georgia Power Company's Hatch Nuclear Plant." However, the note on page 1 of Attachment 1 (to PSE&G's Amendment Application dated April 25 1994) says that this amendment request is not an exemption to 10CFR50 Appendix J, Type C Testing. They are proposing the substitution of ASME Section XI testing for the subject CIV's. Does it mean an exemption from Type C testing was issued to Hatch and no substitution with an appropriate ASME test was required? Also, shouldn't the details (CIVs and systems involved) of the exemption, issued to Hatch, be provided in the Justification Section of this amendment? The BNE (Bureau of Nuclear Engineering of the State of New Jersey) believes that an appropriate ASME code case could have been considered for Hatch.

The following answers were given to the state official by the staff in a telephone conference on June 21, 1995:

The above safety evaluation for this amendment states the valves listed in PSE&G's amendment application dated April 25, 1994, are in compliance with 10 CFR Part 50 Appendix J and that no leak rate test is required to satisfy Appendix J requirements; and further, the safety evaluation states that the CIVs will, however, continue to be tested pursuant to the applicable requirements of Section XI of the ASME B&PV Code. The staff has agreed with the licensee that no exemption is required. Concerning the Hatch amendment, which Hope Creek used as an example in preparing their amendment application, the staff stated that although Hatch's application was for exemptions to CIVs and closed systems inside containment, the staff found some of the valves that were on lines which penetrated the torus and terminated below the torus minimum level were in compliance with Appendix J, needing no Type C test nor exemption.

The state official had no further comments.

4.0 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. 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 (59 FR 29632). 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.

5.0 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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Date: August 1, 1995