June 8, 1992

Docket No. 50-354

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear Officer
Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

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Dear Mr. Miltenberger:

SUBJECT: REVISE RADIATION PROTECTION AND RADIOLOGICAL ENVIRONMENTAL MONITORING TECHNICAL SPECIFICATIONS AND VARIOUS ORGANIZATIONAL CHANGES, HOPE CREEK GENERATING STATION (TAC NO. M75880)

The Commission has issued the enclosed Amendment No. 52 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 January 24, 1990 and supplemented on April 27, 1990, December 4, 1990, September 10, 1991 and December 10, 1991. The supplemental letters did not affect the original no significant hazards determination.

This amendment changes the Technical Specifications by revising the radiation protection and radiological environmental monitoring Technical Specifications, and revises the organization and responsibilities of the Nuclear Safety Review department and the Station Operations Review Committee.

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

Sincerely,

/S/ James C. Stone for

Stephen Dembek, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 52 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

*Previous Concurrence

OFFICE	PDI-2/LA	*PDI-2/PM	PDI-2/PM	PDI-2/D	OGC
NAME	MO'BRIEN	CSHIRAKI	SDEMBEK:tlc	CMiller	S. Horn
DATE	3/17/92	06/07/90	3/23/92	6/7/92	4/6/92

OFFICE	*PRPB				
NAME	LCUNNINGHAM				
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Vice President and Chief Nuclear
Officer
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Post Office Box 236
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1. Amendment No. 52 to
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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Steven E. Miltenberger
Public Service Electric & Gas
Company

Hope Creek Generating Station

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
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 January 24, 1990, and supplemented on April 27, 1990, December 4, 1990, September 10, 1991 and December 10, 1991, 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. 52 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, 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: June 8, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 52

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. Overleaf pages provided to maintain document completeness.*

Remove	Insert
xxiii	xxiii*
xxiv	xxiv
xxv	xxv
xxvi	xxvi*
3/4 12-9	3/4 12-9
3/4 12-10	3/4 12-10
6-1	6-1*
6-2	6-2
6-5	6-5
6-6	6-6
6-7	6-7
6-8	6-8
6-9	6-9
6-10	6-10
6-11	6-11
6-12	6-12
6-13	6-13
6-14	6-14
6-15	6-15
6-16	6-16*
6-17	6-17
6-18	6-18
6-19	6-19*
6-20	6-20
6-21	6-21
6-22	6-22
6-23	6-23*
6-24	6-24
6-25	6-25
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ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2*	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is a 40 CFR Part 141 value. If no drinking water pathway exists, a value of 20 pCi/ℓ may be used.

TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1)(2)

LOWER LIMIT OF DETECTION (LLD)(3)

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m³)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1*	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*LLD for drinking water samples. If no drinking water pathway exists, a value of 10 pCi/ℓ may be used.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager - Hope Creek Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Senior Nuclear Shift Supervisor or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President and Chief Nuclear Officer shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 ONSITE AND OFFSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Hope Creek Generating Station Updated Final Safety Analysis Report and updated in accordance with 10 CFR 50.71(e).

b. The General Manager - Hope Creek Operations shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President and Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 The unit organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

UNIT STAFF (Continued)

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator shall be in the control room;

c. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions e.g., licensed Senior Reactor Operators, licensed Reactor Operators, radiation protection technicians, equipment operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal** 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance on major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the appropriate department manager, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the General Manager-Hope Creek Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

**The shift schedule is based upon a 12 hour shift with a work week of either 36 hours or 48 hours.

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SNSS*	1	1
NSS*	1	None
NCO	2	1
EO	2	1
STA	1	None
RPT	1	1

TABLE NOTATION

SNSS - Senior Nuclear Shift Supervisor with a Senior Reactor Operator license on the Unit
NSS - Nuclear Shift Supervisor with a Senior Reactor Operator license on the Unit
NCO - Nuclear Control Operator with a Reactor Operator license on the Unit
EO - Equipment Operator
STA - Shift Technical Advisor
RPT - Radiation Protection Technician

Except for the Senior Nuclear Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Senior Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual with a valid Senior Reactor Operator license shall be designated to assume the control room command function. During any absence of the Senior Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Reactor Operator license or Operator license shall be designated to assume the control room command function.

*In cases where an individual has a Senior Reactor Operator's license on the unit, is a qualified STA, and has a Professional Engineers License by virtue of successful completion of the Professional Engineers examination or a bachelor's degree in a scientific, engineering, or engineering technology discipline from an accredited institution, the individual can serve in a dual role capacity as either the SNSS/STA or NSS/STA. (Note: For those individuals with a bachelor's degree in a scientific or engineering technology discipline, course work must have included physical, mathematical, or engineering science.) Otherwise, there shall be a qualified STA as well as a SNSS and NSS on-shift.

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support to the Senior Nuclear Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1981 for comparable positions, except for the individual designated as the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Reactor Operators and Senior Reactor Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

6.3.2 The Operations Manager, Operating Engineer, Senior Nuclear Shift Supervisors, and Nuclear Shift Supervisors, shall hold a senior reactor operator license. The Nuclear Control Operators shall hold a reactor operator license.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager-Nuclear Training, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1981 and 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager - Site Protection and shall meet or exceed the requirements of the SRP (NUREG-0800) Section 13.2.2.II.6, 10 CFR 50 Appendix R and Branch Technical Position CMEB 9.5.1, Section C.3.d.

6.5 REVIEW AND AUDIT

6.5.1 STATION OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.1.1 The SORC shall function to advise the General Manager - Hope Creek Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The SORC shall be composed of the:

Chairman:	General Manager - Hope Creek Operations
Member and Vice Chairman:	Operations Manager
Member and Vice Chairman:	Technical Manager
Member and Vice Chairman:	Maintenance Manager
Member:	Maintenance Engineer
Member:	Technical Engineer
Member:	Radiation Protection/ Chemistry Manager
Member:	Radiation Protection Engineer or Chemistry Engineer
Member:	Onsite Safety Review Engineer
Member:	Operating Engineer

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman.

a. Only designated Vice Chairmen or the General Manager - Hope Creek Operations may act as Chairman of a SORC meeting.

b. No more than two alternates to members shall participate as voting members in SORC activities at any one meeting.

c. Alternate appointees will only represent their respective department.

d. Alternates for members will not make up part of the voting quorum when the member the alternate represents is also present.

MEETING FREQUENCY

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the SORC necessary for the performance of the SORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four members including alternates.

RESPONSIBILITIES

6.5.1.6 The SORC shall be responsible for:

a. Review of: (1) Upper tier administrative procedures within the scope of Regulatory Guide 1.33 (2/78), and changes thereto and (2) Newly created procedures or changes to existing procedures that require a 10 CFR 50.59 safety evaluation as described in Section 6.5.3.2.d.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Review of the safety evaluations that have been completed under the provisions of 10 CFR 50.59.

f. Initiation or review of investigations of all violations of the Technical Specifications including the reports covering evaluations and recommendations to prevent recurrence.

g. Review of all REPORTABLE EVENTS.

h. Review of facility operations to detect potential nuclear safety hazards.

i. Performance of special reviews, investigations or analyses and reports thereon as determined by the SORC.

j. Review of the Facility Security Plan and implementing procedures and changes thereto that require a 10 CFR 50.59 safety evaluation, or involve a potential decrease in the effectiveness of the plan, per 10 CFR 50.54(p).

k. Review of the Facility Emergency Plan and implementing procedures and changes thereto that require a 10 CFR 50.59 safety evaluation, or involve a potential decrease in the effectiveness of the plan, per 10 CFR 50.54(q).

l. Review of the Fire Protection Program and implementing procedures and changes thereto that require a 10 CFR 50.59 safety evaluation.

m. Review of all unplanned on-site releases of radioactivity to the environs including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence.

n. Review of changes to the PROCESS CONTROL MANUAL and the OFF-SITE DOSE CALCULATION MANUAL.

REVIEW PROCESS

6.5.1.7 A technical review and control system utilizing qualified reviewers shall function to perform the periodic or routine review of procedures and changes thereto. Details of this technical review process are provided in Section 6.5.3.

AUTHORITY

6.5.1.8 The SORC shall:

a. Provide recommendations in writing to the General Manager - Hope Creek Operations indicating approval or disapproval of items considered under Specification 6.5.1.6 prior to their implementation.

b. Provide written notification within 24 hours to the Vice President and Chief Nuclear Officer and to the General Manager - Quality Assurance and Nuclear Safety of disagreement between the SORC and the General Manager - Hope Creek Operations; however, the General Manager - Hope Creek Operations shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS AND REPORTS

6.5.1.9 The SORC shall maintain minutes of each SORC meeting, and copies shall be provided to the Vice President and Chief Nuclear Officer, General Manager - Quality Assurance and Nuclear Safety and Manager - Nuclear Safety.

6.5.2 NUCLEAR SAFETY REVIEW AND AUDIT

FUNCTION

6.5.2.1 The Nuclear Safety Department shall function to provide the independent safety review program and audit of designated activities.

COMPOSITION

6.5.2.2 The Nuclear Safety Department, under the cognizance of the General Manager - Quality Assurance and Nuclear Safety, shall consist of the Manager - Nuclear Safety, the Offsite Safety Review staff (OSR) and the Onsite Safety Review Group (SRG). The OSR staff and the SRG each consist of at least four dedicated, full-time engineers.

The Manager - Nuclear Safety and the Offsite Safety Review staff shall meet or exceed the qualifications described in Section 4.7 of ANS 3.1 - 1981 and shall be guided by the provisions for independent review described in Section 4.3 of ANSI N18.7 - 1976 (ANS 3.2).

The Offsite Safety Review staff shall generally possess experience and competence in the areas listed in Section 6.5.2.4.1. A system of qualified reviewers from other technical organizations shall be utilized to augment OSR staff expertise in the disciplines of Section 6.5.2.4.1, where appropriate. Such qualified reviewers shall meet the same qualification requirements as the Offsite Safety Review staff, and shall not have been involved with performance of the original work.

The Onsite Safety Review Group staff shall meet or exceed the qualifications described in Section 4.4 of ANS 3.1 - 1981.

CONSULTANTS

6.5.2.3 Consultants or other technical experts shall be utilized by the Nuclear Safety Department as needed.

6.5.2.4 OFFSITE SAFETY REVIEW (OSR)

FUNCTION

6.5.2.4.1 The OSR staff shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical engineering,

h. Electrical engineering

i. Quality assurance

j. Nondestructive testing

k. Emergency preparedness

REVIEW

6.5.2.4.2 The OSR staff shall review:

a. The safety evaluations for changes to procedures, equipment, or systems; and tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems and tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed changes to Technical Specifications or this Operating License;

d. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

e. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety;

f. All REPORTABLE EVENTS.

g. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

h. Reports and meeting minutes of the SORC.

AUDITS

6.5.2.4.3 Audits of facility activities shall be performed under the cognizance of the OSR staff. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire facility staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The Facility Emergency Plan and implementing procedures at least once per 12 months;

f. The Facility Security Plan and implementing procedures at least once per 12 months;

g. Any other area of facility operation considered appropriate by the General Manager - Quality Assurance and Nuclear Safety or the Vice President and Chief Nuclear Officer;

h. The facility Fire Protection Program and the implementing procedures at least once per 24 months;

i. The fire protection and loss prevention program implementation at least once per 12 months utilizing either a qualified off-site licensee fire protection engineer(s) or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least once per 36 months; and

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

l. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months; and,

m. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

The above audits may be conducted by the Quality Assurance Department or an independent consultant. Audit plans and final audit reports shall be reviewed by the OSR staff.

RECORDS AND REPORTS

6.5.2.4.4 Records of OSR activities shall be maintained. Reports of reviews and audits shall be prepared and distributed as indicated below:

a. The results of reviews performed pursuant to Section 6.5.2.4.2 shall be reported to the Vice President and Chief Nuclear Officer at least monthly.

b. Audit reports prepared pursuant to Specification 6.5.2.4.3 shall be forwarded by the auditing organization to the Vice President and Chief Nuclear Officer and to the management positions responsible for the areas audited (1) within 30 days after completion of the audit for those audits conducted by the Quality Assurance Department, and (2) within 60 days after completion of the audit for those audits conducted by an independent consultant.

6.5.2.5 ONSITE SAFETY REVIEW GROUP (SRG)

FUNCTION

6.5.2.5.1 The SRG shall function to provide: the review of plant design and operating experience for potential opportunities to improve plant safety; evaluation of plant operations and maintenance activities; and advice to management on the overall quality and safety of plant operations.

The SRG shall make recommendations for revised procedures, equipment modifications, or other means of improving plant safety to appropriate station/corporate management.

RESPONSIBILITIES

6.5.2.5.2 The SRG shall be responsible for:

a. Review of selected plant operating characteristics, NRC issuances, industry advisories, and other appropriate sources of plant design and operating experience information which may indicate areas for improving plant safety.

b. Review of selected facility features, equipment, and systems.

c. Review of selected procedures and plant activities including maintenance, modification, operational problems, and operational analysis.

d. Surveillance of selected plant operations and maintenance activities to provide independent verification* that they are performed correctly and that human errors are reduced to as low as reasonably achievable.

AUTHORITY

6.5.2.6 The Nuclear Safety Department shall report to and advise the Vice President and Chief Nuclear Officer on those areas of responsibility specified in Sections 6.5.2.4 and 6.5.2.5.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 All programs and procedures required by Technical Specification 6.8 and changes thereto, and any other proposed procedures or changes thereto, which affect plant nuclear safety as determined by the General Manager - Hope Creek Operations, other than editorial or typographical changes, shall be reviewed as follows:

PROCEDURE RELATED DOCUMENTS

6.5.3.2 Procedures, Programs and changes thereto shall be reviewed as follows:

a. Each newly created procedure, program or change thereto shall be independently reviewed by an individual knowledgeable in the area affected other than the individual who prepared the procedure, program or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Station Administrative procedures will be approved by the appropriate station Department Manager or by the General Manager - Hope Creek Operations. Each station Department Manager shall be responsible for a pre-designated class of procedures. The General Manager - Hope Creek Operations shall approve Station Administrative Procedures, Security Plan implementing procedures and Emergency Plan implementing procedures.

b. On-the-spot changes to procedures which clearly do not change the intent of the approved procedures shall be approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. Revisions to procedures which may involve a change in intent of the approved procedures, shall be reviewed in accordance with Section 6.5.3.2.a above.

c. Individuals responsible for reviews performed in accordance with item 6.5.3.2.a above shall be approved by the SORC Chairman and designated as a Station Qualified Reviewer. A system of Station Qualified Reviewers shall be maintained by the SORC Chairman. Each review shall include a written determination of whether or not additional

*Not responsible for sign-off function.

cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel. The Station Qualified Reviewers shall meet or exceed the qualifications described in Section 4.1 and 4.7 of ANS 3.1, 1981.

d. If the Department Manager determines that the documents involved require a 10 CFR 50.59 safety evaluation, the documents shall be forwarded for SORC review and also to the OSR staff for an independent review to determine whether or not an unreviewed safety question is involved. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions or requiring Technical Specification changes shall be obtained prior to implementation.

NON-PROCEDURE RELATED DOCUMENTS

6.5.3.3 Tests or experiments, and changes to equipment or systems shall be forwarded for SORC review and also to the OSR staff for an independent review to determine whether or not an unreviewed safety question is involved. The results of the OSR staff reviews will be provided to SORC. Recommendations for approval are made by SORC to the General Manager - Hope Creek Operations. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions or requiring Technical Specification changes shall be obtained prior to implementation.

RECORDS AND REPORTS

6.5.3.4 Written records of reviews performed in accordance with item 6.5.3.2a above, including recommendations for approval or disapproval, shall be maintained. Copies shall be provided to the General Manager - Hope Creek Operations, SORC, the OSR staff, and/or NRC as necessary when their reviews are required.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50 and a report submittal pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC, and the results of this review shall be submitted to the OSR staff and the Vice President and Chief Nuclear Officer.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President and Chief Nuclear Officer and the General Manager - Quality Assurance and Nuclear Safety shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the General Manager - Quality Assurance and Nuclear Safety and the Vice President and Chief Nuclear Officer within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. PROCESS CONTROL PROGRAM implementation.

i OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environment monitoring.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved in accordance with specification 6.5.1.6 or 6.5.3, as appropriate, prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 On-the-Spot changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

PROCEDURES AND PROGRAMS (Continued)

c. The change is documented and receives the same level of review and approval as the original procedure under Specification 6.5.3.2a within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the HPCI, CS, RHR, RCIC, Containment Hydrogen Recombiner, H_2/O_2 analyzer, Post-Accident Sampling, Control Rod Drive Hydraulic (Scram Discharge portion) systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and
2. A service pressure leak test for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the USNRC Administrator, Region 1, unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance,

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or self reading dosimeter measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions; and

b. Documentation of all challenges to main steamline safety/relief valves.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine radiological environmental operating reporting covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. Deviations from the sampling program identified in Technical Specification 3.12.1 shall be reported. In the event that some results are not available for inclusion with the reports, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps, one covering sampling locations near the SITE BOUNDARY, and a second covering the more distant locations, all keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, as required by Specification 3.12.3.

The report shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 Routine radioactive release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The historical annual average meteorology or the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM). The Semiannual Radioactive Effluent Release Report shall identify those radiological environmental sample parameters and locations where it is not possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In addition, the cause of the unavailability of samples for the pathway and the new location(s) for obtaining replacement samples should be identified. The report should also include a revised figure(s) and table(s) for the ODCM reflecting the new location(s).

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive effluent release report to be submitted within 60 days after January 1, of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each class of solid waste (as defined by 10 CFR 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclide (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compact dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to the UNRESTRICTED AREA of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP), OFFSITE DOSE CALCULATION MANUAL (ODCM) or radioactive waste systems made during the reporting period.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the USNRC Administrator, Region 1, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the PSE&G generated CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following Technical Specifications:

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE
3/4.2.3 MINIMUM CRITICAL POWER RATIO
3/4.2.4 LINEAR HEAT GENERATION RATE

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A (the latest approved revision), General Electric Standard Application for Reactor Fuel (GESTAR II).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

RECORD RETENTION (Continued)

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Quality Assurance Program.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of SORC meetings and OSR activities.

l. Records of the snubber service life monitoring pursuant to Technical Specification 4.7.5.

m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity

HIGH RADIATION AREA (Continued)

of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem* shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Senior Nuclear Shift Supervisor on duty and/or the radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

*Measurement made at 18 inches from source of radioactivity.

**Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

PROCESS CONTROL PROGRAM (PCP) (Continued)

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determination; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the UFSAR for the period in which the evaluation was reviewed by SORC. The discussion of each change shall contain:

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS (Continued)

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 52 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

By letter dated January 24, 1990 and supplemented on April 27, 1990, December 4, 1990, September 10, 1991 and December 10, 1991, Public Service Electric & Gas Company (PSE&G) requested an amendment to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. The supplemental letters did not affect the original no significant hazards determination. The proposed amendment would revise the Radiation Protection and Radiological Environmental Monitoring Technical Specifications and make various organizational changes.

2.0 EVALUATION

The changes proposed in this amendment may be categorized and justified as follows:

a. Changes related to PSE&G's organizational configuration, which reflect titular changes, and changes in management responsibility are administrative in nature and do not adversely impact management attentiveness to safe operation of the Hope Creek Generating Station. The recent reorganization of PSE&G's Nuclear Department is intended to increase overall management effectiveness, in some cases by consolidating oversight of related activities. For example, quality assurance and nuclear safety review and audit activities will both be under the direction of one general manager. The Manager - Nuclear Safety will assume management responsibility of the Onsite Safety Review staff, thereby allowing the Onsite Safety Review Engineer to dedicate his efforts to review activities.

   This category includes the proposed change to require SORC review for procedure changes and changes to Security Plan, Emergency Plan and Fire Protection Program Plan only if a 10 CFR 50.59 safety evaluation is involved. This approach will consolidate the screening and review process for procedure changes by doing away with the significant safety issue determination currently in use at the Hope Creek Generating Station. Screening for safety significance will be performed by determining 10 CFR 50.59 applicability (10 CFR 50.54(p) for Security Plan changes and 10 CFR 50.54(q) for Emergency Plan changes), which is consistent with NRC regulations regarding procedure changes (10 CFR 50.59 and 10 CFR 50.54), and is acceptable.

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b. PSE&G's current practice of preparing and submitting the Annual Radiological Environmental Operating Report includes the two sampling location maps explicitly described by the proposed change. This amendment clarifies the current practice, which is acceptable to the NRC, in order to remove any ambiguity from the Technical Specifications, and is acceptable.

c. Changes to reporting levels of radioactivity concentrations and lower limits of detection (LLD) for the radiological environmental monitoring program are consistent with NUREG-0473, "Standard Radiological Effluent Technical Specifications for Boiling Water Reactors," Revision 3, draft. These changes propose allowing a higher level of I-131 if a drinking water pathway is not potentially affected by the effluent being monitored. The provisions of 40 CFR 141 will still be complied with where applicable. 10 CFR 20.106 requires that effluents released to unrestricted areas are maintained within the limits of Appendix B, Table II of 10 CFR Part 20. Table II specifies a limit of 300 pCi/1 for soluble I-131 and 60,000 pCi/1 for the insoluble form. Using 300 pCi/1 for comparison purposes, the proposed reporting level and LLD for I-131 are respectively 6.7% and 3.3% of the Table II limits.

   Therefore, the proposed changes will not affect compliance with 10 CFR 20.106 and will not allow for an increase in radiation dose to any member of the public, and is acceptable.

d. Changes to the description of Nuclear Safety Review (NSR) responsibilities include consolidating the management of the Offsite and Onsite Safety Review Groups and revising the description of NSR activities to increase specificity and eliminate redundancy, and is acceptable.

   Changes in this category will not lessen the scope of NSR activities and will increase the effectiveness of the Onsite Safety Review Group by allowing the Onsite Safety Review Engineer to dedicate his time to review activities, since the Manager - Nuclear Safety will have management responsibility. None of the changes in this category will reduce the effectiveness of NSR review and audit functions, and they are acceptable.

e. Changes deleting references to outdated requirements or documents are justified on the basis that they are largely editorial and provide clarification without reducing any commitments, and are acceptable.

3.0 SUPPLEMENTAL CORRESPONDENCE

The original amendment request dated January 24, 1990 proposed incorporating the provisions of 10 CFR 20.203(c)(4), which allows a high radiation area established for a period of 30 days or less to be controlled using direct surveillance, into Technical Specification 6.12.1. During discussions between the staff and the licensee on March 13, 1990, it was determined that the existing Technical Specifications do not conflict with 10 CFR 20.203(c)(4). Therefore, the amendment request was modified in supplemental correspondence dated April 27, 1990 to delete the changes that had been requested to paragraph 6.12.1.

In addition, the original amendment request dated January 24, 1990 proposed changes to the distribution requirements for SORC meeting minutes per Technical Specification 6.5.1.9. "Vice President - Nuclear Operations" was added to the required distribution in lieu of the "Vice President and Chief Nuclear Officer." The original amendment request was modified in supplemental correspondence dated December 4, 1990 which withdrew the aforementioned proposed change and requested that the Vice President and Chief Nuclear Officer remain on the required distribution for SORC meeting minutes. This request is consistent with the Administrative Control Technical Specifications proposed for the Salem Generating Station.

The September 10, 1991, supplemental correspondence modified the SORC membership and quorum requirements and revised the description of the administrative procedures requiring SORC review. The changes to SORC composition requirements will allow one Technical Engineer and one Maintenance Engineer to serve as SORC member at any given time. In order to assure continuity among SORC meetings, membership in such positions will be controlled by administrative procedure. In its December 10, 1991, letter PSE&G made a commitment to have the designated member serve for a minimum period of 90 days. The PSE&G proposal would not change the requirement that a minimum quorum consists of the Chairman plus four members including alternates. Therefore, at least 50% of the SORC members must be present in order to have a quorum. Based on the staff's review of PSE&G's submittals, the staff finds that PSE&G's proposal would adequately maintain the SORC's ability to continue its function to advise the General Manager - Hope Creek Operations on all matters related to nuclear safety. Therefore, PSE&G's proposal is acceptable.

Finally, the September 10, 1991, supplemental correspondence revised TS 6.5.1.6a to more accurately describe the administrative procedures requiring SORC review. The proposed change would specify that upper tier administrative procedures and changes that are within the scope of Regulatory Guide 1.33, Revision 2 (February 1978) (RG 1.33), require SORC review. PSE&G's proposal limits the scope of the Station Administrative Procedures and changes thereto that would require SORC review. Based on the staff's review of RG 1.33 and the Hope Creek TS, the staff finds that the licensees' proposal would continue to provide for review of all administrative procedures related to nuclear safety and is, therefore, acceptable.

The supplemental correspondence dated April 27, 1990, December 4, 1990, September 10, 1991 and December 10, 1991 did not change the staff's original no significant hazards determination.

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 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 (55 FR 6116). 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.

The amendment also involves changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment also meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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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