Docket File
50-354

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

September 30, 1997

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. M99059)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 105 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 June 19, 1997, as supplemented by letters dated July 30 and 31, 1997.

This amendment changes TS 4.1.3.1.2, "Control Rod Operability;" TS 3.1.3.6, "Control Rod Drive Coupling;" TS 3.1.3.7, "Control Rod Position Indication;" TS 3.1.4.1, "Rod Worth Minimizer;" TS 3/4.1.4.2, "Rod Sequence Control System;" TS 3/4.10.2, "Special Test Exceptions - Rod Sequence Control System;" the Bases for TS 2.2.1.2, "Average Power Range Monitor;" the Bases for TS 3/4.1.4, "Control Rod Program Controls;" and the Bases for TS 3/4.10.2, "Rod Sequence Control System." The changes eliminate the Rod Sequence Control System (RSCS) Limiting Condition for Operation and Surveillance Requirements from the TSs and reduce the Rod Worth Minimizer low power setpoint to 10% from 20%. Changes to other sections of the TSs delete reference to the RSCS from the TSs and incorporate additional requirements necessary to support the elimination of the RSCS.

DF01

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/

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

cc w/encls: See next page

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Sincerely,

David H. Jaffe, Senior Project Manager
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2. Safety Evaluation

cc w/encls: See next page

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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. 105
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated June 19, 1997, as supplemented by letters dated July 30 and 31, 1997, 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. 105 , 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.

In addition, paragraph 2.C.(14) to Facility Operating License No. NPF-57 is amended as follows:

(14) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 105 , are hereby incorporated into this license. Public Service Electric and Gas Company shall operate the facility in accordance with the Additional Conditions.

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: 1. Page 1 of Appendix C of License* No. NPF-57
2. Changes to the Technical Specifications

Date of Issuance: September 30, 1997

*Page 1 of Appendix C is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

1. Insert Appendix C, Page 1

2. 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
v	v
xv	xv
xxi	xxi
B 2-6	B 2-6
3/4 1-4	3/4 1-4
3/4 1-11	3/4 1-11
3/4 1-13	3/4 1-13
3/4 1-16	3/4 1-16
-	3/4 1-16a
3/4 1-17	3/4 1-17
3/4 10-2	3/4 10-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-5	B 3/4 1-5
B 3/4 10-1	B 3/4 10-1

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. DPR-57

Public Service Electric and Gas Company and Atlantic City Electric Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
97	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 21, 1997.
103	The licensee shall relocate the list of "Motor Operated Valves - Thermal Overload Protected (BYPASSED)" from the Technical Specifications (Table 3.8.4.2-1) to the Updated Final Safety Analysis Report, as described in the licensee's application dated July 7, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from September 16, 1997
105	The licensee shall use the Banked Pattern Withdrawal System or an improved version such as the Reduced Notch Worth Procedure as described in the licensee's application dated June 19, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from September 30, 1997.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves** either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. The provisions of Specification 3.0.4 are not applicable.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d. With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

e. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore the inoperable valve(s) to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 24 hours verifying each valve to be open,* and

b. At least once per 31 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed

*These valves may be closed intermittently for testing under administrative controls.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   1. If permitted by the RWM, insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      a) Observing any indicated response of the nuclear instrumentation, and
      b) Demonstrating that the control rod will not go to the overtravel position.

   2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM, then until permitted by the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:

   1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

   2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

c. The provisions of Specification 3.0.4 are not applicable.

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within 1 hour:

   1. Determine the position of the control rod by using an alternate method, or:

      a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,
      b) Returning the control rod, by single notch movement, to its original position, and
      c) Verifying no control rod drift alarm at least once per 12 hours, or

   2. Move the control rod to a position with an OPERABLE position indicator, or

   3. When THERMAL POWER is:

      a) Within the preset power level of the RWM, declare the control rod inoperable.
      b) Greater than the preset power level of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of Specification 3.0.4 are not applicable.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER, minimum allowable low power setpoint.

ACTION:

a. With the RWM inoperable after the first 12 control rods are fully withdrawn, operation may continue provided that control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

b. With the RWM inoperable before the first twelve (12) control rods are fully withdrawn, one startup per calendar year may be performed provided that the control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

c. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

* Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

See Special Test Exception 3.10.2.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

SURVEILLANCE REQUIREMENTS (CONTINUED)

b. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

The material originally contained in Section 3/4.1.4.2 was deleted with the issuance of Amendment No. . However, to maintain numerical continuity between the succeeding sections and existing station procedural references to those Technical Specification sections, 3/4.1.4.2 has been intentionally left blank.

3/4.10.2 ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod worth minimizer (RWM) per Specification 3.1.4.1 may be suspended for the following tests provided that control rod movement prescribed for this testing is verified by a second licensed operator or other technically qualified member of the unit technical staff present at the reactor console:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

With the requirements of the above specification not satisfied, verify that the RWM is OPERABLE per Specifications 3.1.4.1.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RWM are bypassed, verify:

a. That movement of the control rods from 75% ROD DENSITY to the RWM low power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests.

b. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

c. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3. Additional pertinent analysis is also contained in Amendment 17 to the Reference 4 topical report.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

rate, solution concentration or boron equivalent to meet the ATWS Rule must not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron control equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution (natural boron enrichment)."

The described minimum system parameters (82.4 gpm, 13.6 percent concentration and natural boron equivalent) will ensure an equivalent injection capability that exceeds the ATWS Rule requirement. The stated minimum allowable pumping rate of 82.4 gallons per minute is met through the simultaneous operation of both pumps.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's", G. E. Topical Report NEDO-10527, March 1972
2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972
3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores", Supplement 2 to NEDO-10527, January 1973
4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD WORTH MINIMIZER

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations during open vessel testing requires additional restrictions in order to ensure that criticality is properly monitored and controlled. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

The material originally contained in this Technical Specification was deleted with the issuance of Amendment No. 35. However, to maintain the historical reference to this specification, this section has been intentionally left blank.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

The material originally contained in Bases Section 3/4.10.7 was deleted with the issuance of Amendment No. 14. However, to maintain the historical reference to this section, Bases Section 3/4.10.7 is intentionally left blank.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 105 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 June 19, 1997, as supplemented by letters dated July 30 and 31, 1997, (Reference 1), the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Hope Creek Generating Station, Technical Specifications (TSs). The requested changes would be to TS 4.1.3.1.2, "Control Rod Operability;" TS 3.1.3.6, "Control Rod Drive Coupling;" TS 3.1.3.7, "Control Rod Position Indication;" TS 3.1.4.1, "Rod Worth Minimizer;" TS 3/4.1.4.2, "Rod Sequence Control System;" TS 3/4.10.2, "Special Test Exceptions - Rod Sequence Control System;" the Bases for TS 2.2.1.2, "Average Power Range Monitor;" the Bases for TS 3/4.1.4, "Control Rod Program Controls;" and the Bases for TS 3/4.10.2, "Rod Sequence Control System." The changes eliminate the Rod Sequence Control System (RSCS) Limiting Condition for Operation and Surveillance Requirements from the TSs and reduce the Rod Worth Minimizer (RWM) low power setpoint to 10% from 20%. Changes to other sections of the TSs delete references to the RSCS from the TSs and incorporate additional requirements necessary to support the elimination of the RSCS.

2.0 EVALUATION

The RSCS restricts rod movement to minimize the individual worth of control rods to lessen the consequences of a Rod Drop Accident (RDA). Control rod movement is restricted through the use of rod select, insert, and withdrawal blocks. The RSCS is a hardwired (as opposed to a computer controlled), redundant backup to the RWM. It is independent of the RWM in terms of inputs and outputs but the two systems are compatible. The RSCS is designed to monitor and block, when necessary, operator control rod selection, withdrawal and insertion actions, and thus assist in preventing significant control rod pattern errors which could lead to a control rod with a high reactivity worth (if dropped). A significant pattern error is one of several abnormal events all of which must occur to have an RDA which might exceed fuel energy density limit criteria for the event. The RSCS was designed only for possible mitigation of the RDA and is active only during low power operation (currently, generally less than 20 percent power) when an RDA might be significant. The RSCS provides rod blocks on detection of a significant pattern error and does not prevent an RDA. A similar pattern control function is also performed by the RWM, a computer controlled system. All reactors having a RSCS also have an RWM.

In August 1986 (Reference 2), the BWR Owners' Group (BWROG) in cooperation with General Electric proposed an Amendment 17 to GESTAR 11 (Reference 3) which would eliminate the requirement for the RSCS and retain the RWM but lower the setpoint for turn off (during startup) or turn on (during shutdown) to 10% from 20%. The NRC staff review concluded that the proposed changes were acceptable, and approved Amendment 17, but imposed several additional requirements which would be necessary to implement the changes. The staff's safety analysis and additional requirements were presented and discussed in an attachment to Reference 4. (This review and approval is also available in Reference 3, page US.C-379.)

The additional requirements were:

1) The TSs should require provisions for minimizing operations without the RWM system operable.

2) The occasional, necessary, use of a second operator replacement should be strengthened by a licensee's review of relevant procedures, related forms and quality control to assure that the second operator provides an effective and truly independent monitoring process. A discussion of this review should accompany the request for RSCS removal.

3) Rod patterns used should be at least equivalent to Banked Position Withdrawal Sequence (BPWS) patterns.

3.0 EVALUATION

The licensee has proposed changes to the TSs, and associated Bases, and otherwise addressed the concerns of the NRC staff related to removal of the RSCS as follows:

1. Elimination of the RSCS requirements. The licensee has proposed the deletion of TS 3/4.1.4.2, which contains the Limiting Conditions for Operation and Surveillance Requirements associated with the RSCS. A note of explanation would be added which would document the removal of these requirements.

2. Reduction of the RWM setpoint to 10 percent. The requirements of TS 3.1.4.1 would be modified to decrease the RWM setpoint to 10% from 20%.

3. Increased administrative control of RWM operability (intended to result in decreased use of the second operator as a substitute for the RWM), and implementation of Banked Pattern Withdrawal System (BPWS). The licensee's submittal dated June 19, 1997 addresses the procedures for second operator actions, when required, to ensure independent monitoring of the control rod patterns. In addition, TS 3.1.4.1 would be changed to address inoperability of the RWN before the first twelve (12) rods have been withdrawn. Under these conditions, one startup per calendar year is permitted, provided that control rod withdrawal is supervised by an operator or other technically qualified individual. The June 19, 1997, submittal commits to the BPWS, or equivalent, or an improved version.

4. Changes to other sections of the TSs are also proposed as necessary to delete reference to the RSCS from the TSs and to incorporate additional requirements necessary to support the elimination of the RSCS.

The NRC staff's review and basis for approval of the removal of the RSCS and lowering of the setpoint for the RWM, as proposed by the licensee in sections of the submittal relating to topics 1 and 2, above, are provided in References 2 and 3. The proposed changes, in the June 19, 1997, submittal (as supplemented) fall within the scope of the staff's review and approval in Reference 1.

The licensee has increased the administrative control of the RWM, as required in the staff's review of RSCS removal. The proposed revised TSs require the RWM to be operable at the beginning of each startup (until twelve rods are withdrawn) with only one exception per year. This follows the pattern of previously approved RWM TSs (discussed in Reference 4) which have been found to provide the desired improvement in reliability for the system. The TSs and procedures for the use of a second operator, or technically qualified individuals (when the RWM is inoperable), have been reviewed and found to provide a suitable independent check on the rod patterns. Based upon the above, and the incorporation into a license condition the licensee's commitment to the BPWS or equivalent, or an improved version, the NRC staff's review of the proposed TSs that implement the removal of the RSCS indicates that the proposed TSs are appropriate, clearly stated, and are acceptable.

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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 45462). 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 STATE CONSULTATION

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

6.0 CONCLUSION

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

Principal Contributor: D. H. Jaffe

Date: September 30, 1997

7.0 REFERENCES

1. Letter and enclosures from L. Storz, Public Service Electric and Gas Company (PSE&G), to the Nuclear Regulatory Commission (NRC), dated June 19, 1997, "Request for Change to Technical Specifications RSCS Elimination/RWM Low Power Setpoint Reduction." Letters dated July 30, 1997 from D. Powell, PSE&G, and July 31, 1997, from L. Storz, PSE&G, provided supplements to the June 19, 1997 letter.

2. Letter and enclosures from T. A. Pickens, BWR Owners' Group, to G. Lainas, NRC, dated August 15, 1986, "Amendment 17 to GE Licensing Topical Report NEDE-24011-P-A.11."

3. NEDE-24011-P-A-9, September 1988, "General Electric Standard Application for Reactor Fuel," (GESTAR II).

4. Letter from A. Thadani, NRC, to J. Charnley, General Electric, dated December 27, 1987, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Revision 8, Amendment 17.11."