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

December 8, 2009

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: MODE CHANGE LIMITATIONS (TAC NO. ME0341)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 180 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 5, 2009, as supplemented by letters dated June 9, and September 2, 2009. The amendment modifies TS requirements for mode change limitations in accordance with Revision 9 of Nuclear Regulatory Commission-approved TS Task Force (TSTF) change TSTF-359, "Increase Flexibility in Mode Restraints."

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

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 180 to
   License No. NPF-57
2. Safety Evaluation

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

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. NPF-57

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

   A. The application for amendment filed by PSEG Nuclear LLC dated January 5, 2009, as supplemented by letters dated June 9, and September 2, 2009, 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. 180, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: December 8, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 0-1	3/4 0-1
3/4 0-2	3/4 0-2
3/4 0-3	3/4 0-3
3/4 0-4	3/4 0-4
3/4 1-4	3/4 1-4
3/4 1-6	3/4 1-6
3/4 1-8	3/4 1-8
3/4 1-10	3/4 1-10
3/4 1-11	3/4 1-11
3/4 1-13	3/4 1-13
3/4 2-3	3/4 2-3
3/4 3-1	3/4 3-1
3/4 3-9	3/4 3-9
3/4 3-62	3/4 3-62
3/4 3-74	3/4 3-74
3/4 3-109	3/4 3-109
3/4 4-2	3/4 4-2
3/4 4-18	3/4 4-18
3/4 4-26	3/4 4-26
3/4 4-28	3/4 4-28
3/4 5-2	3/4 5-2
3/4 6-5	3/4 6-5
3/4 6-17	3/4 6-17
3/4 7-11	3/4 7-11
3/4 8-1	3/4 8-1
3/4 8-24	3/4 8-24
3/4 8-41	3/4 8-41

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, and except as provided in LCO 3.0.8.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

3.0.4 When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LIMITING CONDITION FOR OPERATION (Continued)

3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 Not used.

3.0.7 Not used.

3.0.8 Inoperability of Snubbers

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be a failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within its specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50 Sections 50.55a(f) and 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(f)(6)(i) or Section 50.55a(g)(6)(i).

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall conform to the staff positions on schedule, methods, and personnel, and sample expansion included in that generic letter, or as otherwise approved by the NRC.

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.

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

d. With one or more scram discharge volume (SDV) vent or drain lines*** with one valve inoperable, isolate the associated line within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.****

e. With one or more SDV vent or drain lines*** with both valves inoperable, isolate the associated line 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

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* 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.

*** Separate Action entry is allowed for each SDV vent and drain line.

**** An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

   1. Declare the control rod(s) with the slow insertion time inoperable, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

3. With one or more control rod scram accumulators inoperable and reactor pressure < 900 psig,

    a) Immediately upon discovery of charging water header pressure < 940 psig, verify all control rods associated with inoperable accumulators are fully inserted otherwise place the mode switch in the shutdown position**, and

    b) Within one hour insert the associated control rod(s), declare the associated control rod(s) inoperable and disarm the associated control valves either electrically or 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*:

    1. With one or more withdrawn control rods inoperable, upon discovery immediately initiate action to fully insert inoperable withdrawn control rods.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

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

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

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.

*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.

REACTIVITY CONTROL SYSTEMS

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 alternative 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.

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*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.

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the EOC-RPT inoperable limit specified in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 24% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT:

a. Once within 12 hours after THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER and at least once per 24 hours thereafter.

b. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

3/4.3 INSTRUMENTATION
3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. For the Reactor Vessel Steam Dome Pressure - High Functional Unit and the Reactor Vessel Water Level - Low, Level 3 Functional Unit, the sensor is eliminated from response time testing for RPS circuits. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 or 3 from OPERATIONAL CONDITION 1 for the Intermediate Range Monitors.

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for one trip system, either

    1) place the inoperable channel(s) in the tripped condition within

        a) 1 hour for trip functions without an OPERABLE channel,
        b) 12 hours for trip functions common to RPS instrumentation, and
        c) 24 hours for trip functions not common to RPS instrumentation,

    or

    2) take the ACTION required by Table 3.3.2-1.

c. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for both trip systems,

    1) place the inoperable channel(s) in one trip system in the tripped condition within one hour, and

    2) a) place the inoperable channel(s) in the remaining trip system in the tripped condition within

            1) 1 hour for trip functions without an OPERABLE channel,
            2) 12 hours for trip functions common to RPS instrumentation, and
            3) 24 hours for trip functions not common to RPS instrumentation,

        or

        b) take the ACTION required by Table 3.3.2-1.

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 and Table 3.3.7.4-2 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-2, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

4.3.7.4.2 At least one of each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.10 Two channels of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 with mechanical vacuum pump in service and any main steam line not isolated.

ACTION:

a. With one channel of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip inoperable, restore the channel to OPERABLE status within 12 hours. Otherwise, trip the mechanical vacuum pumps, or isolate the main steam lines or be in HOT SHUTDOWN within the next 12 hours.

b. With mechanical vacuum pump trip capability not maintained:

   1. Trip the mechanical vacuum pumps within 12 hours; or

   2. Isolate the main steam lines within 12 hours; or

   3. Be in HOT SHUTDOWN within 12 hours.

c. When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated ACTIONS may be delayed for up to 6 hours provided the mechanical vacuum pump trip capability is maintained.

SURVEILLANCE REQUIREMENTS

4.3.10 Each channel of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL CHECK at least once per 12 hours;

b. Performance of a CHANNEL FUNCTIONAL TEST at least once per 92 days;

c. Performance of a CHANNEL CALIBRATION at least once per 18 months. The Allowable Value shall be ≤ 3.6 x normal background; and

d. Performance of a LOGIC SYSTEM FUNCTIONAL TEST, including mechanical vacuum pump trip breaker actuation, at least once per 18 months.

ACTION (Continued)

reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specification 3.3.6.

4. Within 4 hours, reduce the Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoints and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip setpoints and Allowable Values of the remaining channels to those applicable for single recirculation loop operation per Specification 3.3.6.

5. Deleted

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

b. With no reactor coolant system recirculation loops in operation, initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries per gram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

a. Note: LCO 3.0.4.c is applicable.
In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

2. Greater than $100/\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

   a) Restore the inoperable valve(s) to OPERABLE status, or

   b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,##, with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.**

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system, one recirculation pump, or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation or at least one recirculation pump is in operation.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

**Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

ACTION:

NOTE: LCO 3.0.4.b is not applicable to HPCI.

a. For the Core Spray system:

   1. With one core spray subsystem inoperable, provided that at least two LPCI subsystem are OPERABLE, restore the inoperable core spray subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With both core spray subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

   1. With one LPCI subsystem inoperable, provided that at least one core spray subsystem is OPERABLE, restore the inoperable LPCI subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With two LPCI subsystems inoperable, provided that at least one core spray subsystem is operable, restore at least one LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   3. With three LPCI subsystems inoperable, provided that both core spray subsystems are OPERABLE, restore at least two LPCI subsystems to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   4. With all four LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

c. For the HPCI system, provided the Core Spray System, the LPCI system, the ADS and the RCIC system are OPERABLE:

* Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Primary Containment Leakage Rate Testing Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each primary containment isolation valve and each reactor instrumentation line excess flow check valve shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

   1. The inoperable valve is returned to OPERABLE status, or

   2. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

Note: LCO 3.0.4.b is not applicable to RCIC.

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and
b. Four separate and independent diesel generators, each with:

   1. A separate fuel oil day tank containing a minimum of 360 gallons of fuel,
   2. A separate fuel storage system consisting of two storage tanks containing a minimum of 44,800 gallons of fuel, and
   3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Note: LCO 3.0.4.b is not applicable to DGs.

a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 separately for each diesel generator within 24 hours* unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated. If continued operation is permitted by LCO 3.7.1.3, restore the inoperable diesel generator to OPERABLE status within 72 hours for diesel generators A or B, or within 14 days for diesel generators C or D, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system, and

   1. For 4.16 kV circuit breakers, de-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by disconnecting* the breaker within 72 hours and verify the inoperable breaker(s) to be disconnected at least once per 7 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that each of the medium voltage 4.16 kV circuit breakers are OPERABLE by performing:

      a) A CHANNEL CALIBRATION of the associated protective relays, and

      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

*After being disconnected, these breakers shall be maintained disconnected under administrative control.

CLASS 1E ISOLATION BREAKER OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.5 All Class 1E isolation breaker (tripped by a LOCA signal) overcurrent protective devices shown in Table 3.8.4.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the overcurrent protective devices shown in Table 3.8.4.5-1 inoperable, declare the affected isolation breaker inoperable and remove the inoperable circuit breaker(s) from service within 72 hours and verify the inoperable breaker(s) to be disconnected at least once per 7 days thereafter.

SURVEILLANCE REQUIREMENTS

4.8.4.5 Each of the Class 1E isolation breaker overcurrent protective devices shown in Table 3.8.4.5-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and a value between 150% and 250% of the pickup of the short time delay, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. For circuit breakers equipped with solid state trip devices, the functional testing may be performed with use of portable instruments designed to verify the time-current characteristics and pickup calibration of the trip elements. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated January 5, 2009, as supplemented by letters dated June 9, and September 2, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML090130384, ML091670251, and ML092580380, respectively), PSEG Nuclear LLC (PSEG, or the licensee) requested changes to the Hope Creek Generating Station (HCGS) Technical Specifications (TSs). The supplements dated June 9, and September 2, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards determination as published in the *Federal Register* on February 24, 2009 (74 FR 8286).

The proposed amendment would modify TS requirements for mode (i.e., operational condition) change limitations in accordance with Revision 9 of TS Task Force (TSTF) change TSTF-359, "Increase Flexibility in Mode Restraints."

By letter to the NRC dated July 17, 2002 (ADAMS Accession No. ML022000470), the Nuclear Energy Institute (NEI) Risk Informed TS Task Force (RITSTF) submitted proposed change TSTF-359, Revision 7, to the Standard Technical Specifications (STS) (i.e., NUREG 1430-1434) on behalf of the industry. TSTF-359, Revision 7, proposed to change Limiting Condition for Operation (LCO) 3.0.4 and Surveillance Requirement (SR) 3.0.4 in the STS by risk-informing limitations on entering the mode of applicability of an LCO. The NRC staff published a notice in the *Federal Register* on August 2, 2002 (67 FR 50475) requesting comments on the model safety evaluation (SE) for this TS improvement using the consolidated line item improvement process (CLIIP). In response to the public comments received on that notice, the NRC staff found that TSTF-359, Revision 7, should be revised. The RITSTF submitted TSTF-359, Revision 8, by letter to the NRC dated December 4, 2002 (ADAMS Accession No. ML023430238). The NRC staff prepared a model SE incorporating changes resulting from the public comments and subsequently made minor editorial changes to the SE. TSTF-359, Revision 8, as modified, provided the complete approved change, as discussed the NRC's *Federal Register* notice of availability dated April 4, 2003 (68 FR 16579). The RITSTF subsequently incorporated the modifications to Revision 8 into TSTF-359, Revision 9, which was submitted to the NRC by letter dated April 23, 2003 (ADAMS Accession No. ML031190607).

Enclosure

TSTF-359 is one of the industry's initiatives under the risk-informed TS program. These initiatives are intended to maintain or improve safety while reducing unnecessary burden and to make TS requirements consistent with the Commission's other risk-informed regulatory requirements, in particular the maintenance rule.

The current STS (NUREG 1430 - 1434) specify that a nuclear power plant cannot go to higher modes of operation[1] (i.e., move toward power operation) unless all TS systems, normally required for the higher mode, are operable. This limitation is included (with several exceptions for some plants) in LCO 3.0.4 and SR 3.0.4. LCO 3.0.4 and SR 3.0.4 in the STS currently state, in part, that when an LCO or SR is not met, "entry into a MODE or other specified condition in the applicability shall not be made except when the associated actions to be entered permit continued operation in the MODE or other specified condition in the applicability for an unlimited period of time." The industry believes that this requirement is unnecessarily restrictive and can unduly delay plant startup while considerable resources are being used to resolve startup issues that are risk insignificant or low risk. A maintenance activity that takes longer than planned can delay a mode change and adversely impact a utility's orderly plant startup and return to power operation. The objective of the proposed change is to provide additional operational flexibility without compromising plant safety.

HCGS has not adopted STS (NUREG-1433, Revision 2), which results in some administrative differences between the HCGS TSs and TSTF-359, Revision 9. Specifically,

1. STS SR 3.0.1 is SR 4.0.1 in the HCGS TSs

2. STS SR 3.0.4 is SR 4.0.4 in the HCGS TSs

3. STS LCO 3.4.7 is LCO 3.4.5 in the HCGS TSs

4. STS LCO 3.5.3 is LCO 3.7.4 in the HCGS TSs

5. STS LCO 3.8.1 is 3.8.1.1 in the HCGS TSs

6. HCGS administrative requirements are located under TS 6.0, not TS 5.0.

7. The location of pre-existing LCO 3.0.4 exceptions in the current HCGS TSs differs from those in STS.

8. The HCGS TSs use "OPERATIONAL CONDITION" instead of "MODE" for describing reactor condition states; consequently "OPERATIONAL CONDITION" is used instead of MODE in the markup of the TSs.

The proposed changes to LCO 3.0.4 and SR 4.0.4 would allow, for systems and components, mode changes into a TS condition that has a specific required action and completion time (CT). The licensee will utilize the LCO 3.0.4 and SR 4.0.4 allowances only when they determine that there is a high likelihood that the LCO will be satisfied within the LCO CT, after the operational

1 MODE numbers decrease in transition "up to a higher mode of operation;" power operation is MODE 1.

condition change. In addition, the LCO 3.0.4 and SR 4.0.4 allowances can be applied to values and parameters in specifications when explicitly stated in the TS (non-system/component TS such as: Reactor Coolant System Specific Activity). These changes are in addition to the current operational condition change allowance when a required action has an indefinite completion time. The LCO 3.0.4 and SR 4.0.4 mode change allowances are not permitted for the systems and components (termed "higher risk") listed in Section 3.1.1, "Identification of Risk-Important TS Systems and Components," for the modes specified. Two examples are: (1) Boiling Water Reactor (BWR) plants cannot transition from Mode 3 to Mode 2 or to Mode 1 with the Reactor Isolation Cooling System inoperable; and, (2) BWR plants cannot transition up into any mode with required Diesel Generators (including other Emergency/Shutdown Alternating Current (AC) Power Supplies) inoperable.

PSEG proposes a change to the wording of STS LCO 3.0.4, and is inserting the word "or" between paragraph LCO 3.0.4.a and 3.0.4.b to highlight the mutual exclusivity of these elements. This "or" connector was contained in the NRC's notice of opportunity to comment on the model SE (67 FR 50475, August 2, 2002) and the NRC's notice of availability for this TS improvement (68 FR 16579, April 4, 2003). However, it was not included in the traveler for TSTF-359, Revision 9, and was not incorporated into the STS.

In its application dated January 5, 2009, PSEG stated that it had reviewed the model SE for TSTF-359 published in the NRC's *Federal Register* notice of availability dated April 4, 2003 (68 FR 16579). PSEG concluded that the justifications presented in SE are applicable to HCGS and justify the proposed amendment. The SE that follows is based on the model SE with revisions as necessary to address the specific changes proposed for HCGS.

2.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2)(i), the "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications . . ." By convention, the LCOs are contained in Sections 3.1 through 3.10 of the STS, TS Sections 3/4.1 through 3/4.12 for HCGS. HCGS TS Section 3/4.0, "Applicability," provides details or ground rules for complying with the LCOs and SRs. LCO 3.0.4 and SR 4.0.4 address requirements for LCO compliance when transitioning between operational conditions.

TSs have taken advantage of risk technology as experience and capability have increased. Since the mid-1980's, the NRC has been reviewing and granting improvements to the TSs that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the Commission stated that it expects that licensees will utilize any plant-specific PRA or risk survey in preparing their TS-related submittals. In evaluating these submittals, the NRC staff applies the guidance in Regulatory

Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998 and in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998. The NRC staff has appropriately adapted this guidance to assess the acceptability of upward mode changes with equipment inoperable. This review had the following objectives:

- To ensure that the plant risk does not increase unacceptably during the actual implementation of the proposed change (e.g., when the plant enters a higher operational condition while an LCO is not met). This risk increase is referred to as "temporary."

- To compare and assess the risk impact of the proposed change to the acceptance guidelines of the Commission's Safety Goal Policy Statement, as documented in RG 1.174. The risk impact, which is measured by the average yearly risk increase associated with the change, aims at minimizing the "cumulative" risk associated with the proposed change so that the plant's average baseline risk is maintained within a minimal range.

- To assess the licensee's ability to identify risk-significant configurations resulting from maintenance or other operational activities and take appropriate compensatory measures to avoid such configurations.

The NRC staff reviewed licensee reliance on 10 CFR 50.65(a)(4) for the non-higher-risk systems and components, and related guidance to assess and manage the risk of upward operational condition changes. The Commission has found that compliance with the industry guidance for implementation of 10 CFR 50.65(a)(4), as endorsed by RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," and mandated by LCO 3.0.4, SR 4.0.4, and SR 4.0.3, satisfies the configuration risk management objectives of RG 1.177 for TS surveillance interval and completion time extensions. The licensee's reliance on 10 CFR 50.65(a)(4) processes that are consistent with the provisions of the NRC-endorsed industry guidance was also found to be adequate for managing the risk of missed surveillances as described in the *Federal Register* on September 28, 2001 (66 FR 49714).

The NRC staff review also had the objective of ensuring that existing NRC inspection programs have the necessary controls in place to allow the NRC staff to oversee the implementation of the proposed change, reliance on 10 CFR 50.65(a)(4) processes or programs, and the ability to adequately assess the licensee's performance associated with risk assessments. The review encompassed inspection procedures (i.e., NRC Inspection Procedure 62709 dated December 28, 2000, "Configuration Risk Assessment and Risk Management Process," and NRC Inspection Procedure 71111.13 dated January 17, 2002, "Maintenance Risk Assessments and Emergent Work Control"), the significance determination process (SDP) (i.e., "Maintenance Risk Assessment and Risk Management Significance Determination Process"), enforcement guidance (i.e., Enforcement Manual Section 7.11, "Actions Involving the Maintenance Rule"), and the associated reactor oversight process.

2.1 Proposed Change to HCGS LCO 3.0.4 and SR 4.0.4

Currently, HCGS LCO 3.0.4 does not allow entrance into a higher operational condition (or other specified condition) in the applicability when an LCO is not met, except when the associated

Actions to be entered permit continued operation in that operational condition or condition indefinitely, or a specific exception is granted. Similarly, when an LCO's surveillances have not been met within their specified frequency, entry into a higher mode (or other specified condition) is not allowed by SR 4.0.4. The current HCGS TS LCO 3.0.4 reads as follows:

> Entry into an OPERATIONAL CONDITION or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL CONDITION or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

The licensee's proposed revision to LCO 3.0.4 will read:

> When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:
>
> a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or
>
> b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or
>
> c. When an allowance is stated in the individual value, parameter, or other Specification.
>
> This Specification shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The HCGS current TS SR 4.0.4 reads as follows:

> Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillances Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

The licensee's proposed revision to SR 4.0.4 will conform to the proposed changes to LCO 3.0.4 and will read as follows:

> Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.
>
> This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The proposed LCO 3.0.4.a retains the current allowance for when the required actions allow indefinite operation. The proposed LCO 3.0.4.b allows entering operational conditions or other specified conditions in the applicability except when higher-risk systems and components (listed below in SE Section 3.1.1), for the operational condition being entered, are inoperable. When applying LCO 3.0.4.b, the decision for entering a higher operational condition or condition in the Applicability of the LCO will be made by plant management after the required risk assessment has been performed and requisite risk management actions established, through the program established to implement 10 CFR 50.65(a)(4). Entry into the operational conditions or other specified conditions in the Applicability of the TS shall be for no more than the duration of the applicable required actions completion time, or until the LCO is met. The licensee has proposed to remove current notes in individual specifications that prohibit operational condition changes which are now encompassed by LCO 3.0.4.b. Similarly, the licensee has proposed to add notes that prohibit operational condition changes under LCO 3.0.4.b for higher-risk systems and components. The proposed LCO 3.0.4.b allowance can involve multiple components in a single LCO or in multiple LCOs; however, use of the LCO 3.0.4.b provisions are always contingent upon completion of a 10 CFR 50.65(a)(4)-based risk assessment.

The notes limiting the applicability, to Operational Conditions 1, 2 and 3 of the current TS LCO 3.0.4 and SR 4.0.4, are holdovers from Amendment 19 to the HCGS operating license. The notes limiting the applicability of LCO 3.0.4 and SR 4.0.4 are no longer needed and are removed consistent with approved TSTF-359, Revision 9. Consideration was originally given to adding notes to various TSs, as defined by the tables of higher-risk systems, precluding entry into OPERATIONAL CONDITIONS 4 and 5 for BWRs. However, it was determined that the addition of notes in these cases is made unnecessary by action statements that require immediate completion times, which means that entry into the Operational Condition or other specified condition in the Applicability is not allowed and the notes would be superfluous.

LCO 3.0.4 allowances related to values and parameters of the TSs are not typically addressed by LCO 3.0.4.b risk assessments, and are therefore addressed by a new LCO 3.0.4.c. LCO 3.0.4.c refers to allowances already in the TSs and annotated in the individual TS. LCO 3.0.4.c also allows for entry into the operational conditions or other specified conditions in the Applicability for TSs for no more than the duration of the applicable required actions completion time or until the LCO is met or the unit is not within the applicability of the TS.

3.0 TECHNICAL EVALUATION

During the development of the current STS, improvements were made to LCO 3.0.4, such as clarifying its applicability with respect to plant shutdowns, cold shutdown mode and refueling mode. In addition, during the STS development, almost all the LCOs with completion times greater than or equal to 30 days, and many LCOs with completion times greater than or equal to 7 days, were given individual LCO 3.0.4 exceptions. During some conversions to the STS, individual plants provided acceptable justifications for other LCO 3.0.4 exceptions. All of these specific LCO 3.0.4 exceptions allow entry into a mode or other specified condition in the TS applicability while relying on the TS required actions and associated completion times. The change proposed for HCGS would provide standardization and consistency to the use and application of LCO 3.0.4, both internal to and between each of the specifications, as well as with the STS. This proposed change will also ensure consistency through the utilization of appropriate levels of risk assessment of plant configurations for application of LCO 3.0.4. However, nothing in this SE should be interpreted as encouraging upward mode transition with inoperable equipment. Good practice should dictate that such transitions should normally be initiated only when all required equipment is operable and that mode transition with inoperable equipment should be the exception rather than the rule.

The current LCO 3.0.4.a allowances are retained in the proposal and do not represent a change in risk from the current situation. The LCO 3.0.4.b allowances apply to systems and components, and require a risk assessment prior to utilization to ensure an acceptable level of safety is maintained. The LCO 3.0.4.c allowances apply to parameters and values which have been previously approved by the NRC in a plant's specific TS. In its application dated January 5, 2009, the licensee provided proposed changes to the HCGS TS Bases that discuss and list each LCO 3.0.4.c specific value and parameter allowance. The proposed TS Bases for LCO 3.0.4 explain the new allowances and their utilization.

In its review of TSTF-359, the NRC staff did a generic qualitative assessment of the risk impact of the proposed change in LCO 3.0.4.b allowances by evaluating how licensee implementation of the proposed risk-informed approach is expected to meet the requirements of the applicable RGs. The NRC staff referred to the guidance provided in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." RG 1.177 provides the NRC staff's recommendations on using risk information to assess the impact of proposed changes to nuclear power plant TSs on the risk associated with plant operation. Although RG 1.177 does not specifically address the type of generic change in this proposal, the NRC staff considered the approach documented in RG 1.177 in evaluating the risk information provided in support the proposed changes in LCO 3.0.4.

The NRC staff's evaluation of how the implementation of the proposed risk-informed approach, used to justify LCO 3.0.4.b allowances, agrees with the objectives of the guidance outlined in RG 1.177 is discussed below in SE Section 3.1. Oversight of the risk-informed approach associated with the LCO 3.0.4.b allowances is discussed in SE Section 3.2. The HCGS plant-specific TS changes are discussed in SE Section 3.3.

3.1 Evaluation of Risk Management

Both the temporary and cumulative risk of the proposed change is adequately limited. The temporary risk is limited by the exclusion of higher-risk systems and components, and completion time limits contained in the TSs (see SE Section 3.1.1). The cumulative risk is limited by the temporary risk limitations and by the expected low frequency of the proposed operational condition changes with inoperable equipment (see SE Section 3.1.2). Adequate NRC oversight of the licensee's ability to use the LCO 3.0.4.b provisions under appropriate circumstances (i.e., to identify risk-significant configurations when entering a higher operational condition or condition in the applicability of an LCO (see SE Section 3.1.3)), is provided by NRC inspection of the licensee's implementation of 10 CFR 50.65(a)(4) as applied to the proposed change.

3.1.1 Temporary Risk Increases

RG 1.177 proposes the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP) as appropriate measures of the increase in probability of core damage and large early release, respectively, during the period of implementation of a proposed TS change. In addition, RG 1.177 stresses the need to preclude potentially high-risk configurations introduced by the proposed change. The ICCDP associated with any specified plant condition, such as the condition introduced by entering a higher operational condition with plant equipment inoperable, is expressed by the following equation:

$$ICCDP = \Delta R\ d = (R_1 - R_0)\ d$$

where:

ΔR = the conditional risk increase, in terms of core damage frequency (CDF), caused by the specified condition

d = the duration of the specified plant condition

R_1 = the plant CDF with the specified condition permanently present

R_0 = the plant CDF without the specified condition

The same expression can be used for ICLERP by substituting the measure of risk (i.e., large early release frequency (LERF) for CDF). The magnitude of the ICCDP and ICLERP values associated with plant conditions applicable to LCO 3.0.4.b allowances can be managed by controlling the conditional risk increase, ΔR (in terms of both CDF and LERF) and the duration, d, of such conditions. The following sections discuss how the key elements of the proposed risk-informed approach, used to justify LCO 3.0.4.b allowances, are expected to limit ΔR and d and, thus, prevent any significant temporary risk increases.

Identification of Risk-Important TS Systems and Components

A major element that limits the risk of the proposed operational condition change flexibility is the exclusion of certain systems and associated LCOs for the operational condition change allowance. The TSs allow operation in Operational Condition 1 (power operation) with specified levels of inoperability for specified times. This provides a benchmark of currently acceptable risk against which to measure any incremental risk inherent in the proposed LCO 3.0.4.b. If a system inoperability accrues risk at a higher rate in one or more of the transition operational conditions than it would in Operational Condition 1, then an upward transition into that operational condition should not be allowed without demonstration of a high degree of experience and sophistication in risk management. However, the risk management process evaluated in SE Section 3.1.3 is adequate if higher-risk systems/components are excluded from the scope of LCO 3.0.4.b.

The importance of most TS systems in mitigating accidents increases as power increases. However, some TS systems are relatively more important during lower power and shutdown operations, because:

- certain events are peculiar to operational conditions of plant operation other than power operation;

- certain events are more probable at operational conditions of plant operation other than power operation; and

- some operational conditions of plant operation have less mitigation system capability than power operation.

The risk information submitted in support of the proposed changes to LCO 3.0.4 and SR 4.0.4 in TSTF-359 included qualitative risk assessments performed by each owners group to identify higher risk systems and components at the various operational conditions of operation, including transitions between operational conditions, as the plant moves upward from the refueling mode of operation toward power operation. The owners groups' generic qualitative risk assessments are included as attachments to TSTF-359, Revision 9. Each of the owners groups' generic qualitative risk assessments discuss the technical approach used and the systems/components subsequently determined to be of higher risk significance; the systems/components not to be granted the LCO 3.0.4 allowances for the various operational conditions are listed. The owners groups' generic qualitative risk assessments are:

- BWR owners' group Risk-Informed Technical Specification Committee, "Technical Justification to Support Risk-Informed Improvements to Technical Specification Mode Restraints for BWR Plants," General Electric Company GE-NE A13-00464 (Revision 2)

- "B&W [Babcock and Wilcox] owners group Qualitative Risk Assessment for Increased Flexibility in MODE Restraints," Framatome Technologies BAW-2383

- Combustion Engineering owners group (CEOG) Task 1181, "Qualitative Risk Assessment for Relaxation of Mode Entry Restraints," CE Nuclear Power LLC, CE NPSD-1207 (Revision 0)

- "WOG [Westinghouse Owners Group] Qualitative Risk Assessment Supporting Increased Flexibility in MODE Restraints"

Following interactions with the NRC staff, all owners groups used the same systematic approach in their qualitative risk assessments to identify the higher-risk systems in the STS, consisting of the following steps:

- identification of plant conditions (i.e., plant parameters and availability of key mitigation systems) associated with changes in plant operational conditions while returning to power;

- identification of key activities that have the potential to impact risk and which are in progress during transitions between operational conditions while the plant is returning to power;

- identification of applicable accident initiating events for each operational condition or other specified condition in the applicability; and

- identification of the higher-risk systems and components by combining the information in the first three steps (qualitative risk assessment).

The risk assessments properly used the results and insights from previous deterministic and probabilistic studies to systematically search for plant conditions in which certain key plant components are more important in mitigating accidents than during operation at power (Mode 1). This search was systematic, taking the following factors into account for the various stages of returning the plant to power:

- the status of accident mitigation and normally operating systems;

- the status of key plant parameters such as reactor coolant system pressure;

- the key activities that are in progress during transitions between operational conditions which have the potential to impact risk (e.g., the transfer from auxiliary to main feedwater at some pressurized water reactor plants when Mode 1 is entered);

- the applicable accident initiating events for each mode of plant operation; and

- design and operational differences among plants or groups of plants.

The following systems and components were identified by the BWR owners group (BWROG) as higher-risk systems and components, when the plant is entering a new operational condition.

System	BWR Type	Entering Mode
High Pressure Coolant Injection (HPCI) System	BWR 3 & 4	2, 1
High Pressure Core Spray (HPCS)	BWR 5 & 6	2, 1
Reactor Core Isolation Cooling (RCIC) System	BWR 3, 4, 5 & 6	2, 1
Isolation Condenser	BWR 2	2, 1

Diesel Generators (including other Emergency/Shutdown AC Power Supplies)	All	All
Hardened Wetwell Vent System	BWR 2, 3 & 4 with Mark I Containment	3, 2, 1
Residual Heat Removal System	All	4

If a licensee identifies a higher-risk system for only some of the operational conditions of applicability, the TSs for that system would be modified by a note that reads, for example, "LCO 3.0.4.b is not applicable when entering OPERATIONAL CONDITION 1 from OPERATIONAL CONDITION 2." Systems identified as higher risk for operational condition 4 and 5 for BWRs, are also excluded from transitioning up to the operational condition of higher risk, and as previously discussed; notes for those transitions are superfluous. In addition, operational condition transitions for operational conditions 4 and 5 for BWRs, will be addressed by administrative controls.

In summary, the NRC staff's review of the owners groups qualitative risk assessments finds that they are of adequate quality to support the application (i.e., they identify the higher-risk systems and components) associated with entering higher operational conditions of plant operation with equipment inoperable while returning to power).

<u>Limited Time in TS Required Actions</u>

Any temporary risk increase will be limited by, among other factors, duration constraints imposed by the TS CTs of the inoperable systems. For the systems and components which are not higher risk, any temporary risk increase associated with the proposed allowance will be smaller than what is considered acceptable when the same systems and components are inoperable at power. This is due to the fact that CTs associated with the majority of TS systems and components were developed for power operation and pose a smaller plant risk for action statement entries initiated or occurring at lower modes of operation as compared to power operation.

The LCO 3.0.4.b allowance will be used only when the licensee determines that there is a high likelihood that the LCO will be satisfied following the operational condition change. This will minimize the likelihood of additional temporary risk increases associated with the need to exit a operational condition due to failure to restore the unavailable equipment within the CT. In most cases, licensees will enter into a higher operational condition with the intent to move up to Operational Condition 1 (power operation). As discussed in Section 3.2, the reactor oversight process monitors unplanned power changes as a performance indicator. The reactor oversight process, thus, discourages licensees from entering an operational condition or other specified condition in the applicability of an LCO, and moving up in power, when there is a likelihood that the operational condition would have to be subsequently exited due to failure to restore the unavailable equipment within the CT. Another disincentive for licensees to enter a higher operational condition when an LCO is not met is related to reporting requirements. It clearly states in 10 CFR 50.72 and 50.73 that a report is required when a nuclear plant shutdown or operational condition change is required by TS. The NRC's oversight program will provide the framework for inspectors and other staff to follow the history at a specific plant of entering higher operational conditions while an LCO is not met, and use such information in assessing the licensee's actions and performance.

3.1.2 Cumulative Risk Increases

The cumulative risk impact of the change to allow the plant to enter a higher operational condition of operation with one or more safety-related components unavailable (as proposed here), is measured by the average yearly risk increase associated with the change. In general, this cumulative risk increase is assessed in terms of both CDF and LERF (i.e., ΔCDF and ΔLERF, respectively). The increase in CDF due to the proposed change is expressed by the following equation, which integrates the risk impact from all expected specified conditions (i.e., all expected plant conditions caused by operational condition changes with various TS systems and components unavailable).

$$\Delta CDF = \Sigma(\Delta CDFi) = \Sigma \ ICCDPi \ fi$$

where:

ΔCDFi = the CDF increase due to specified condition i

ICCDPi = the ICCDP associated with specified condition i

fi = the average yearly frequency of occurrence of specified condition i

A similar expression can be used for ΔLERF by substituting the measure of risk (i.e., LERF for CDF). The magnitude of the ΔCDF and ΔLERF values associated with plant conditions applicable to LCO 3.0.4.b allowances can be managed by controlling the temporary risk increases, in terms of both CDF and LERF (i.e., ICCDP and ICLERP), and the frequency (f), of each of such conditions. In addition to the points made in the previous section regarding temporary risk increases, the following points put into perspective how the key elements of the proposed risk-informed approach, used to justify an LCO 3.0.4.b allowance, are expected to prevent significant cumulative risk increases by limiting the frequency of its use:

- The frequency of risk-significant conditions will be limited by not providing the LCO 3.0.4.b allowances to the higher risk systems and components.

- The frequency of risk-significant conditions will be limited by the requirement to assess the likelihood that the LCO will be satisfied following the operational condition change.

- The frequency of risk-significant conditions is limited by the fact that such conditions can occur only when the plant is returning to power following shutdown (i.e., during a small fraction of time per year). Data over the past 5 years indicate that the plants are averaging 2.1 startups per year.

The addition of the proposed LCO 3.0.4.b allowances to the plant maintenance activities is not expected to change the plant's average (cumulative) risk significantly.

3.1.3 Risk Assessment and Risk Management of Operational Condition Changes

With all safety systems and components operable, a plant can transition up in operational condition to power operation. With one or more system(s) or component(s) inoperable, this change permits a plant to transition up in operational condition to power operation if the inoperable system(s) or component(s) are not in the pre-analyzed higher risk category, a 10 CFR 50.65(a)(4) based risk assessment is performed prior to the operational condition transition, and the requisite risk management actions are taken.

As shown in Attachment 4 of the licensee's application dated January 5, 2009, the proposed TS Bases for LCO 3.0.4 state, in part, that:

> LCO 3.0.4.b allows entry into an OPERATIONAL CONDITION or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.
>
> The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
>
> The results of the risk assessment shall be considered in determining the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and any corresponding risk management actions.

A risk assessment and establishment of risk management actions, as appropriate, are required for determination of acceptable risk for entering the OPERATIONAL CONDITION or other specified conditions in the Applicability when an LCO is not met. Elements of acceptable risk assessment and risk management actions are included in Section 11 of NUMARC 93-01, "Assessment of Risk Resulting from Performance of Maintenance Activities," as endorsed by RG 1.182, which addresses general guidance for conduct of the risk assessment, gives quantitative and qualitative guidelines for establishing risk management actions, and provides example risk management actions. These risk management actions include actions to plan and conduct other activities in a manner that controls overall risk, actions to increase risk awareness by shift and management personnel, actions to reduce the duration of the conditions, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed OPERATIONAL CONDITION change is acceptable.

The guidance references state that a licensee's risk assessment process should be sufficiently robust and comprehensive to assess risk associated with maintenance activities during power operation, low power, and shutdown conditions (all operational conditions of operation), including changes in plant conditions. NUMARC 93-01 states that the risk assessment should include consideration of: the degree of redundancy available for performance of the safety function(s) served by the out-of-service equipment; the duration of the out-of-service condition; component and system dependencies that are affected; the risk impact of performing the maintenance during shutdown versus at power; and, the impact of operational condition transition risk. For power operation, key plant safety functions are those that ensure the integrity of the reactor coolant pressure boundary, ensure the capability to shutdown and maintain the reactor in safe shutdown condition, and ensure the capability to prevent or mitigate the consequences of accidents that could result in potentially significant offsite exposures.

While the inoperabilities permitted by the CTs of TS-required actions take into consideration the safety significance and redundancy of the system or components within the scope of an LCO, the CTs generally do not address or consider concurrent system or component inoperabilities in multiple LCOs. Therefore, the performance of the 10 CFR 50.65(a)(4) risk assessment which looks at the entire plant configuration is essential (and required) prior to changing the operational condition. The 10 CFR 50.65(a)(4)-based risk assessment will be used to confirm (or reject) the appropriateness of transitioning up in operational condition given the actual status of plant safety equipment.

The risk impact on the plant condition of invoking an LCO 3.0.4.b allowance will be assessed and managed through the program established to implement 10 CFR 50.65(a)(4). This program is consistent with RG 1.177 and RG 1.174 in its approach. The implementation guidance for paragraph (a)(4) of the Maintenance Rule addresses controlling temporary risk increases resulting from maintenance activities. This guidance, consistent with guidance in RG 1.177, establishes action thresholds based on qualitative and quantitative considerations and risk management actions. Significant temporary risk increases following an LCO 3.0.4.b allowance are unlikely to occur unless:

- high-risk configurations are allowed (e.g., certain combinations of multiple component outages), or

- risk management of plant operation activities is inadequate.

The requirements associated with the proposed change are established to ensure that such conditions will not occur.

The thresholds of the cumulative (aggregate) risk impacts, assessed pursuant to 10 CFR 50.65(a)(4) and the associated implementation guidance, are based on the permanent change guidelines in NRC RG 1.174. Therefore, licensees will manage the risk exercising LCO 3.0.4 in conjunction with the risk from other concurrent plant activities to ensure that any increase, in terms of CDF and LERF will be small and consistent with the Commission's Safety Goal Policy Statement.

3.2 Oversight

The reactor oversight process (ROP) provides a means for assessing the licensee's performance in the application of the proposed operational condition change flexibility. The adequacy of the licensee's assessment and management of maintenance-related risk is addressed by existing inspection programs and guidance for 10 CFR 50.65(a)(4). Although the current versions of that guidance do not specifically address application of the licensee's (a)(4) program to support risk-informed TSs, it is expected that, in most cases, risk assessment and management associated with risk-informed TSs would be required by (a)(4) anyway because maintenance activities will be involved.

Adoption of the proposed change will make failure to assess and manage the risk of an upward operational condition change with inoperable equipment covered by TS, prior to commencing such an operational condition change, a violation of the TSs. Further, as explained above in general, under most foreseeable circumstances, such a change in configuration would also require a risk assessment under 10 CFR 50.65(a)(4). Inoperable systems or components will necessitate maintenance to restore them to operability, and hence a 10 CFR 50.65(a)(4) risk assessment would be performed prior to the performance of those maintenance actions (except for immediate plant stabilization and restoration actions if necessary). Further, before altering the plant's configuration, including plant configuration changes associated with operational condition changes, the licensee must update the existing (a)(4) risk assessment to reflect those changes.

The *Federal Register* Notice issuing a revision to the Maintenance Rule, 10 CFR 50.65 (64 FR 38551, July 19, 1999), along with NRC Inspection Procedure 71111.13, and Section 11, dated February 22, 2000, "Assessment of Risk Resulting from Performance of Maintenance Activities," of NUMARC 93-01, all indicate that to determine the safety impact of a change in plant conditions during maintenance, a risk assessment must be performed before changing plant conditions. The Bases for the proposed TS change state that the risk assessment and management of upward operational condition changes will be conducted under the licensee's program and process for meeting 10 CFR 50.65(a)(4). Oversight of licensee performance in assessing and managing the risk of plant maintenance activities is conducted principally by inspection in accordance with Reactor Oversight Program Baseline Inspection Procedure (IP) 71111.13, "Maintenance Risk Assessment and Emergent Work Control." Supplemental IP 62709, "Configuration Risk Assessment and Risk Management Process," is utilized to evaluate the licensee's process, when necessary.

The ROP is described in overview in NUREG-1649, Rev. 3, "Reactor Oversight Process," and in detail in the NRC Inspection Manual. IP 71111.13 requires verification of performance of risk assessments when they are required by 10 CFR 50.65(a)(4) and in accordance with licensee procedures. The procedure also requires verification of the adequacy of those risk assessments and verification of effective implementation of licensee-prescribed risk management actions. The rule itself requires such assessment and management of risk prior to maintenance activities, including preventive maintenance, surveillance, and testing (and promptly for emergent work) during all operational conditions of plant operation. The guidance documents for both industry implementation of (a)(4) and NRC oversight of that implementation indicate that changes in plant configuration (which would include mode changes) in support of maintenance activities must be taken into account in the risk assessment and management process.

Revisions to NRC inspection guidance and licensee implementation procedures will be needed to address oversight of risk assessment and management required by the TSs in support of operational condition changes that are not already required under the circumstances by (a)(4). This consideration provides performance-based regulatory oversight of the use of the proposed flexibility, and a disincentive to use the flexibility without the requisite care in planning.

In addition, the NRC staff developed the significance determination process (SDP) guidance for use in assessing inspection findings related to 10 CFR 50.65(a)(4). This guidance was issued in draft for comment and became final during August 2008. The ROP considers inspection findings and performance indicators in evaluating licensee ability to operate safely. The SDP is used to determine the significance of inspection findings related to licensee assessment and management of the risk associated with performing maintenance activities under all plant operating or shutdown conditions. Unplanned reactor scrams and unplanned power changes are two of the Reactor Safety Performance Indicators that the ROP utilizes to assess licensee performance and inform the public. The ROP will provide a disincentive to entering into power operation (Operational Condition 1) when there is a significant likelihood that the mode would have to be subsequently exited due to failure to restore the unavailable equipment within the completion time.

3.3 HCGS Plant-Specific TS Changes

The licensee is adopting the wording in TSTF-359 for LCO 3.0.4 and SR 4.0.4. LCO 3.0.4.c has been referenced appropriately for parameters and values in LCO 3.4.5, "Specific Activity."

HCGS is a BWR/4 reactor type with a Mark I containment. The licensee has, consistent with the higher-risk systems and components identified by the BWROG (shown above in the table in SE Section 3.1.1), proposed that notes to the appropriate TSs be added to state that the revised LCO 3.0.4.b is not applicable to HPCI (LCO 3.5.1), RCIC (LCO 3.7.4), and the emergency diesel generators (LCO 3.8.1.1).

HCGS has a hardened torus (wetwell) vent system. As shown above in the table in SE Section 3.1.1, this system has been identified by the BWROG as a higher-risk system for entering Operational Conditions 1, 2, and 3. There are no TSs associated with this system. As discussed in its letter dated June 9, 2009, PSEG is proposing to address this system by using the requirements of 10 CFR 50.65(a)(4) and guidance of NUMARC 93-01, "Industry Guidance for monitoring the Effectiveness of Maintenance at Nuclear Power Plants" to provide the appropriate considerations for determining the acceptability of, and risk management actions necessary to support, the use of LCO 3.0.4.b with the Hardened Torus Vent System out of service. The NRC staff finds that the requirements of 10 CFR 50.65(a)(4) and guidance of NUMARC 93-01 are appropriate for addressing the risk impact of the inoperability of this system on planned entry into Operational Conditions 1, 2, and 3.

As shown above in the table in SE Section 3.1.1, the residual heat removal system has been identified by the BWROG as a higher risk system for entering Operational Condition 4. The licensee proposes to address the operational condition transition from Operational Condition 4 to Operational Condition 5 by administrative controls. In its letter dated June 9, 2009, PSEG stated in response to a request for additional information that:

> Entry into Operational Condition 4 from Operational Condition 5 is controlled by the Shutdown Safety Management Program (SSMP). The SSMP (PSEG Procedure OU-AA-103) is designed to meet the applicable requirements of 10 CFR 50.65(a)(4) and guidance of NUMARC 93-01, 'Industry Guidance for monitoring the Effectiveness of Maintenance at Nuclear Power Plants'."

The NRC staff finds that the requirements of 10 CFR 50.65(a)(4) and guidance of NUMARC 93-01, as implemented through the SSMP, are appropriate for addressing the risk impact of the inoperability of this system on planned entry into Operational Condition 4.

The proposed amendment would remove pre-existing LCO 3.0.4 exceptions (i.e., statements that the provisions of LCO 3.0.4 are not applicable) from the Action Requirements in the following HCGS LCOs:

LCO	Description
3.1.3.1	Control Rod Operability
3.1.3.2	Control Rod Maximum Scram Insertion Times
3.1.3.4	Four Control Rod Group Scram Insertion Times
3.1.3.5	Control Rod Scram Accumulators
3.1.3.6	Control Rod Drive Coupling
3.1.3.7	Control Rod Position Indication
3.2.3	Minimum Critical Power Ratio
3.3.1	Reactor Protection System Instrumentation
3.3.2	Isolation Actuation Instrumentation
3.3.7.1	Radiation Monitoring Instrumentation
3.3.7.4	Remote Shutdown Instrumentation and Control
3.3.10	Mechanical Vacuum Pump Trip Instrumentation
3.4.1.1	Recirculation System
3.4.7	Main Steam Line Isolation Valves
3.4.9.1	Residual Heat Removal
3.6.1.3	Primary Containment Air Locks
3.6.3	Primary Containment Isolation Valves
3.8.4.1	Primary Containment Penetration Conductor Overcurrent Protective Devices
3.8.4.5	Class 1E Isolation Breaker Overcurrent Protective Devices

In its letter dated September 2, 2009, the licensee provided its justification for the proposed removal of the pre-existing LCO 3.0.4 exceptions. PSEG stated, in part, that:

> There are two categories of 'Action Requirements' in the existing TS that have '3.0.4 not applicable' statements: (i) The first specifies the remedial actions that permit continued operation of the facility not restricted by the time limits of Action Requirements. In this case, conformance to the Action Requirements provides an acceptable level of safety for continued operation of the facility, and operation may proceed indefinitely as long as the remedial Action Requirements are met. (ii) The second type of Action Requirement specifies a time limit in which the LCO must be met. This time limit is the time allowed to restore an inoperable system or component to operable status or to restore parameters within specified limits. If these actions are not completed within the allowable outage time limits, action

> must be taken to shut down the facility by placing it in a mode or condition of operation in which the LCO does not apply.

With respect to Action Requirements that permit continued operation for an unlimited period of time, the NRC staff's position is stated in Generic Letter 87-09, "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements." The Staff Position section included under the Problem #1 section in Enclosure 1 to Generic Letter 87-09 states:

> Specification 3.0.4 unduly restricts facility operation when conformance to the Action Requirements provides an acceptable level of safety for continued operation. For an LCO that has Action Requirements permitting continued operation for an unlimited period of time, entry into an operational mode or other specified condition of operation should be permitted in accordance with those Action Requirements. This is consistent with NRC's regulatory requirements for an LCO. The restriction on a change in operational modes or other specified conditions should apply only where the Action Requirements establish a specified time interval in which the LCO must be met or a shutdown of the facility would be required. However, nothing in this staff position should be interpreted as endorsing or encouraging a plant startup with inoperable equipment. The staff believes that good practice should dictate that the plant startup should normally be initiated only when all required equipment is operable and that startup with inoperable equipment must be the exception rather than the rule.

The NRC staff finds that PSEG justification of the removal of pre-existing LCO 3.0.4 exceptions, from the HCGS TSs for which the Action Requirements permit continued operation for an unlimited period of time, is consistent with the NRC staff's position in Generic Letter 87-09 and, therefore, is acceptable.

With respect to Action Requirements which specify a time limit in which the LCO must be met, the licensee's letter dated September 2, 2009, also stated that:

> Currently HCGS TS permit entry into an OPERATIONAL CONDITION or other specified condition when the LCOs in this category are not met. Upon removal of the individual TS 3.0.4 exceptions, entry into an OPERATIONAL CONDITION or other specified condition under the same circumstances will also require performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate. Removal of the individual TS 3.0.4 exceptions and application of the revised LCO 3.0.4 requirements to the above listed LCOs represents a more restrictive change to the HCGS TS.

The NRC staff agrees that the removal of the pre-existing LCO 3.0.4 exceptions form the HCGS TSs for which the Action Requirements specify a time limit in which the LCO must be met, represents a more restrictive change to the HCGS TS and, therefore, is acceptable.

HCGS TS 3.6.3, "Primary Containment Isolation Valves," contains a plant-specific non-standard LCO 3.0.4 exception. The plant-specific non-standard exception states:

> The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with ACTION a.2, or a.3, above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

In its letter dated June 9, 2009, PSEG stated in response to a request for additional information:

> Consequently, deleting the "non-standard" exception in TS 3.6.3 is no different that deleting the "standard" exceptions elsewhere in TS. If the 3.0.4.b allowance is used for TS 3.6.3, compliance with Action a.2 or a.3 is still required (these Actions are not changed by the proposed amendment). This would also require meeting the Action requirements for any TS systems that are declared inoperable as a result of actions taken to comply with Action a.2 or a.3.

The NRC staff agrees with PSEG's response to the request for additional information and, therefore, finds the removal of the plant-specific non-standard LCO 3.0.4 exception in TS 3.6.3 acceptable.

3.4 Summary

The licensee submitted a proposed TS change to allow entry into a higher operational condition of operation, or other specified condition in the TS Applicability, while relying on the TS conditions, and associated required actions and completion times, provided a risk assessment is performed to confirm the acceptability of that action. The proposal revises HCGS LCO 3.0.4 and SR 4.0.4, and their application to the TSs. New paragraphs a, b, and c are proposed for LCO 3.0.4.

The proposed LCO 3.0.4.a retains the current allowance, permitting the operational condition change when the TS required actions allow indefinite operation.

Proposed LCO 3.0.4.b is the change to allow entry into a higher operational condition of operation, or other specified condition in the TS Applicability, while relying on the TS conditions and associated required actions and completion times, provided a risk assessment is performed to confirm the acceptability of that action for the existing plant configuration. The NRC staff review finds that the process proposed by the licensee for assessing and managing risk during the implementation of the proposed LCO 3.0.4.b allowances meets Commission guidance and, therefore, is acceptable. Key elements of this process are listed below.

- A risk assessment shall be performed before any LCO 3.0.4.b allowance is invoked.

- The risk impact on the plant condition when invoking an LCO 3.0.4.b allowance will be assessed and managed through the program established to implement 10 CFR 50.65(a)(4) and the associated guidance in RG 1.182. Allowing entry into a higher operational condition or condition in the Applicability of an LCO after a 10 CFR 50.65(a)(4) based risk assessment and appropriate risk management actions are taken for the existing plant configuration will ensure that plant safety is maintained.

- The LCO 3.0.4.b allowance will be used only when the licensee determines that there is a high likelihood that the LCO will be satisfied within the required action's completion time.

- TS systems and components which may be of higher risk during operational condition changes have been identified generically by each owners' group for each plant operational mode or condition. Licensees will identify such plant-specific systems and components in the individual plant TSs. The proposed LCO 3.0.4.b allowance does not apply to these systems and components for the operational condition or condition in the applicability of an LCO at which they are of higher risk.

- In adopting LCO 3.0.4.b, the licensee will ensure that plant procedures in place to implement 10 CFR 50.65(a)(4) address the situation where entering an operational condition or other specified condition in the applicability is contemplated with plant equipment inoperable. Such plant procedures will follow the guidance endorsed by NRC RG 1.182.

The NRC's reactor oversight process provides the framework for inspectors and other NRC staff to oversee the implementation of 10 CFR 50.65(a)(4) requirements at a specific plant and assess the licensee's actions and performance.

The LCO 3.0.4.b allowance does not apply to values and parameters of the TSs that have their own respective LCO (e.g., Reactor Coolant System Specific Activity), but instead those values and parameters are addressed by LCO 3.0.4.c. The TS values and parameters for which operational condition transition allowances apply will have a note that states LCO 3.0.4.c is applicable.

The objective of the proposed change is to provide additional operational flexibility without compromising plant safety.

The licensee has a bases control program in HCGS TS 6.15 which is consistent with the bases control program described in the STS for General Electric BWR/4 plants, NUREG-1433, Revision 3. In its application dated January 5, 2009, the licensee committed to implement the TS Bases change for LCO 3.0.4 and SR 4.0.4 concurrent with the license amendment. The NRC staff agrees that the TS Bases Control Program is the appropriate process for updating the affected TS Bases pages.

Based on the above evaluation, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. In a letter dated April 14, 2009 (ADAMS Accession No. ML091110322), the New Jersey Department of Environmental Protection's Bureau of Nuclear Engineering (NJ BNE) provided the following comment:

> The proposed changes to the Bases pages (Attachment 4 [of the application dated January 5, 2009]) state on the third page that "The LCO 3.0.4.b risk assessments do not have to be documented." The position of the NJ BNE is that documentation of these risk assessments is necessary in order to provide a method for verifying that these risk assessments address all risk concerns, are performed and reviewed correctly, are approved by the proper level of management, and will become a permanent record. Without documentation, any future inquiries as to a specific operational condition change that required a risk assessment would be severely hampered. Therefore, we suggest that the above quoted sentence be deleted from the proposed Bases pages.

In its letter dated June 9, 2009, PSEG provided the following response to the NJ BNE comment:

> The TSTF-359 proposed Bases correctly state that there is no regulatory requirement for documentation of each individual use of the risk assessment and risk management actions for use of LCO 3.0.4.b.
>
> A licensee adopting this CLIIP change is required to commit in the Bases to the Technical Specifications to follow Regulatory Guide 1.182. Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," endorses NUMARC 93-01 Section 11. NUMARC 93-01, Section 11 requires that the risk assessment process be proceduralized. Section 11.3.9 states that the normal work control process suffices as a record that the assessment was performed and that it is not necessary to document the basis of each assessment. Normal PSEG practice is to document 3.0.4.b risk assessments, but adoption of the CLIIP does not create a new regulatory requirement for documentation of risk assessments performed to comply with the requirements of 10 CFR 50.65(a)(4).
>
> However, PSEG will remove the quoted sentence from the HCGS Bases (under the TS Bases Program, TS 6.15).

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. 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 (74 FR 8286). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: K. Bucholtz
R. Ennis

Date: December 8, 2009

December 8, 2009

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: MODE CHANGE LIMITATIONS (TAC NO. ME0341)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 180 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 5, 2009, as supplemented by letters dated June 9, and September 2, 2009. The amendment modifies TS requirements for mode change limitations in accordance with Revision 9 of Nuclear Regulatory Commission-approved TS Task Force (TSTF) change TSTF-359, "Increase Flexibility in Mode Restraints."

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

Sincerely,
/ra/
Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 180 to License No. NPF-57
2. Safety Evaluation

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GHill, OIS
KBucholtz

ADAMS Accession No: ML093270023 *OGC concurrence not required per ML072980209

OFFICE	LPL1-2/PM	LPL1-2/LA	ITSB/BC	OGC*	LPL1-2/BC
NAME	REnnis	ABaxter	RElliott	N/A	HChernoff
DATE	12/2/09	12/2/09	12/7/09	N/A	12/8/09