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

September 14, 2011

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

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AND FEEDWATER HEATERS OUT-OF-SERVICE (TAC NO. ME4786)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 190 to Renewed Facility Operating License (FOL) No. NPF-57 for the Hope Creek Generating Station (HCGS) in response to your application dated September 22, 2010, as supplemented by letter dated April 28, 2011.

The amendment allows HCGS to operate at a reduced feedwater temperature for purposes of extending the normal fuel cycle. The amendment also allows operation with feedwater heaters out-of-service at any time during the operating cycle. In addition, the amendment revises surveillance requirements related to testing of the Oscillation Power Range Monitor.

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

Sincerely,

RBEnnis

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. 190 to
   Renewed 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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 190
Renewed 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 September 22, 2010, as supplemented by letter dated April 28, 2011, 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 as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 190, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the renewed 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 90 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 the Technical Specifications

Date of Issuance: September 14, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 190

RENEWED FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Renewed Facility Operating License with the revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Page 3	Page 3
Page 5	Page 5

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3/4 3-110	3/4 3-110

reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(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. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

(7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

C. This renewed 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. 190, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(7) Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)

PSEG Nuclear shall obtain NRC approval of the Class B and C solid waste process control program prior to processing Class B and C solid wastes.

(9) Emergency Planning (Section 13.3, SSER No. 5)

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR 50.54(s)(2) will apply.

(10) Initial Startup Test Program (Section 14, SSER No. 5)

Any changes to the Initial Startup Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(11) Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)

The facility shall not be operated with a rated thermal power feedwater temperature less than 329.6°F for the purpose of extending the normal fuel cycle.

(12) Detailed Control Room Design Review (Section 18.1, SSER No. 5)

3/4.3 INSTRUMENTATION

3/4.3.11 OSCILLATION POWER RANGE MONITOR

LIMITING CONDITION FOR OPERATION

3.3.11 Four channels of the OPRM instrumentation shall be OPERABLE*. Each OPRM channel period based algorithm amplitude trip setpoint (Sp) shall be less than or equal to the Allowable Value as 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.

ACTIONS

a. With one or more required channels inoperable:

   1. Place the inoperable channels in trip within 30 days, or
   2. Place associated RPS trip system in trip within 30 days, or
   3. Initiate an alternate method to detect and suppress thermal hydraulic instability oscillations within 30 days.

b. With OPRM trip capability not maintained:

   1. Initiate alternate method to detect and suppress thermal hydraulic instability oscillations within 12 hours, and
   2. Restore OPRM trip capability within 120 days.

c. Otherwise, reduce THERMAL POWER to less than 24% RTP within 4 hours.

SURVEILLANCE REQUIREMENTS

4.3.11.1 Perform CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

4.3.11.2 Calibrate the local power range monitor in accordance with the Surveillance Frequency Control Program in accordance with Note f, Table 4.3.1.1-1 of TS 3/4.3.1.

4.3.11.3 Perform CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. Neutron detectors are excluded.

4.3.11.4 Perform LOGIC SYSTEM FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

4.3.11.5 Verify OPRM is enabled when THERMAL POWER is ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to the percentage of rated core flow as specified in the CORE OPERATING LIMITS REPORT in accordance with the Surveillance Frequency Control Program. The value specified in the CORE OPERATING LIMITS REPORT shall not be less than 60% of rated core flow.

4.3.11.6 Verify the RPS RESPONSE TIME is within limits in accordance with the Surveillance Frequency Control Program. Neutron detectors are excluded.

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

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated September 22, 2010, as supplemented by letter dated April 28, 2011 (References 1 and 2), PSEG Nuclear LLC (PSEG, or the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS) Technical Specifications (TSs) and Facility Operating License (FOL). The proposed amendment would allow HCGS to operate at a reduced feedwater temperature for purposes of extending the normal fuel cycle. The amendment would also allow operation with feedwater heaters out-of-service (FWHOOS) at any time during the operating cycle. In addition, the proposed amendment would revise the surveillance requirements (SRs) related to testing of the Oscillation Power Range Monitor (OPRM).

The supplement dated April 28, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on January 10, 2011 (76 FR 1466).

The attachment to this safety evaluation (SE) contains a list of the acronyms used in the SE.

2.0 REGULATORY EVALUATION

2.1 Background

The licensee provided the following background information regarding the proposed amendment in its application dated September 22, 2010:

> At current end of rated conditions reactor thermal power decreases if cycle operation continues. This condition, commonly identified as a power coastdown, is when core reactivity is decreased below the level which can be compensated for by withdrawal of control rods or the increase of total core flow. FFWTR [final feedwater temperature reduction] offers cycle extension for a given fuel reload by maintaining rated reactor thermal power through the reactivity inserted by

Enclosure

> reducing the feedwater temperature, thus delaying the onset of a power coastdown period. Predictions indicate that the implementation of FFWTR at the end of an operating cycle can result in a cycle extension at rated power of approximately 30 days for a given fuel reload and a 100 °F FWTR [feedwater temperature reduction]. Similar economic benefit could be expected for subsequent operating cycles.
>
> In addition, reduced, or partial feedwater heating during operation is desired to allow for operation with FWHOOS for planned corrective or preventative maintenance activities or to avoid unnecessary reactor power reductions or SCRAMs in response to an unplanned loss of a portion of the feedwater heating capacity.
>
> Operation resulting in partial, or reduced, feedwater heating for cycle extension is currently prohibited at HCGS by License Condition 2.C (11).

In addition to the changes needed to License Condition 2.C.(11) to implement FFWTR, the licensee has also proposed changes to TS 3/4.3.1, "Oscillation Power Range Monitor." Specifically, SR 4.3.11.5 requires verification that the OPRM is enabled when THERMAL POWER is greater than or equal to 26.1% Rated Thermal Power (RTP) and recirculation drive flow is less than or equal to the value corresponding to 60% of rated core flow. The licensee stated in its application dated September 22, 2010, that the value of the rated core flow required to bound the region susceptible to an instability is determined on a cycle-specific basis and will vary depending upon the magnitude of the FWTR implemented for a particular operating cycle. Consequently, PSEG is proposing that the SR be revised to relocate the value of the rated core flow parameter to the Core Operating Limits Report (COLR), which is a licensee-controlled document.

2.2 Proposed FOL and TS Changes

2.2.1 Proposed FOL Changes

Currently License Condition 2.C.(11) reads as follows:

> Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)
>
> The facility shall not be operated with reduced feedwater temperature for the purpose of extending the normal fuel cycle unless analyses supporting such operation are submitted by the licensee and approved by the staff.

The current restrictions specified in License Condition 2.C(11) were put in place as part of the HCGS extended power uprate (EPU) amendment (Amendment No. 174) which was approved by the NRC staff on May 18, 2008 (Agencywide Documents Access and Management System (ADAMS) package Accession No. ML081230540). As discussed in the NRC staff's SE supporting the EPU amendment:

> The HCGS design FW temperature at CPPU [constant pressure power uprate] conditions is 431.6 °F. HCGS has been evaluated for operation with a FW [feedwater] temperature reduction of approximately 23 °F from the design FW temperature (minimum assumed FW temperature of 409 °F).
>
> The analyses performed by the licensee and documented in its September 18, 2006 submittal support operation with reduced FW temperature and allow continued operation during FW system maintenance, if required. For future operating cycles, the reload process will continue to address the effects of reduced FW temperature on the cycle specific safety analyses. HCGS will not operate with reduced FW temperature for the purpose of extending cycle energy capability beyond the normal end-of-cycle condition without prior NRC review and approval.

The proposed amendment would revise License Condition 2.C.(11) to read as follows:

> <u>Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)</u>
>
> The facility shall not be operated with a rated thermal power feedwater temperature less than 329.6 °F for the purpose of extending the normal fuel cycle.

To support the amendment request, analyses and evaluations have been prepared by PSEG and GE - Hitachi Nuclear Energy Americas LLC (GEH). The results of the analyses and evaluations are documented in GEH Report NEDC-33506P, "Hope Creek Generating Station Operation with Final Feedwater Temperature Reduction and Feedwater Heaters Out-Of-Service," dated September 2010 (Reference 3). The analyses and evaluations are based on plant operation with up to a 102 °F reduction in rated feedwater temperature for FFWTR and up to a 60 °F reduction in rated feedwater temperature for FWHOOS. The 102 °F temperature reduction corresponds to a decrease from the normal feedwater temperature of 431.6 °F to 329.6 °F. The 60 °F temperature reduction corresponds to 371.6 °F.

GEH Report NEDC-33506P (included as Attachment 4 to the application dated September 22, 2010) contains proprietary information and is non-publicly available. Attachment 5 to the application (NEDO-33506) is a non-proprietary, public version of NEDC-33506P (Reference 4).

GEH designates operation at the end-of-cycle exposure as EOC. Extension beyond EOC with FW temperature reduction up to 102 °F is designated as FFWTR. Continued operation at anytime during the cycle with less than full FW heating capability with a FW temperature reduction up to 60°F is an operating flexibility designated as FWHOOS. The licensee's application and this SE refer to FWTR, which encompasses both FFWTR and FWHOOS. Evaluations that apply specifically to FFWTR or FWHOOS are explicitly stated.

2.2.2 Proposed TS Changes

Currently SR 4.3.11.5 states:

> Verify OPRM is enabled when THERMAL POWER is ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to 60% of rated core flow in accordance with the Surveillance Frequency Control Program.

PSEG proposes to revise the SR to state:

> Verify OPRM is enabled when THERMAL POWER is ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to the percentage of rated core flow as specified in the CORE OPERATING LIMITS REPORT in accordance with the Surveillance Frequency Control Program. The value specified in the CORE OPERATING LIMITS REPORT shall not be less than 60% of rated core flow.

2.3 Regulatory Requirements and Guidance Documents

The regulatory requirements and guidance documents the NRC staff considered in its review of the proposed amendment included the following:

- 10 CFR 50.36, "Technical specifications," requires that the TSs include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. Paragraph (c)(3) of 10 CFR 50.36 states that SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 1, "Quality standards and records," states, in part, that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- 10 CFR Part 50, Appendix A, Criterion 2, "Design bases for protection against natural phenomena," states, in part, that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

- 10 CFR Part 50, Appendix A, Criterion 4, "Environmental and dynamic effects design bases," states, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

- 10 CFR Part 50, Appendix A, Criterion 10, "Reactor design," states that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- 10 CFR Part 50, Appendix A, Criterion 12, "Suppression of reactor power oscillations," states that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- 10 CFR Part 50, Appendix A, Criterion 14, "Reactor coolant pressure boundary," states that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- 10 CFR Part 50, Appendix A, Criterion 15, "Reactor coolant system design," states that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- 10 CFR Part 50, Appendix A, Criterion 16, "Containment design," states that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- 10 CFR Part 50, Appendix A, Criterion 20, "Protection system functions," states that the protection system be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- 10 CFR Part 50, Appendix A, Criterion 29, "Protection against anticipated operational occurrences," states that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

- 10 CFR Part 50, Appendix A, Criterion 35, "Emergency core cooling," states, in part, that a system be provided to provide abundant emergency core cooling. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

- 10 CFR Part 50, Appendix A, Criterion 50, "Containment design basis," states, in part, that the reactor containment structure, including access openings, penetrations, and the containment heat removal system be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- NUREG-0661, "Safety Evaluation Report - Mark I Containment Long-Term Program," dated July 1980, provides the NRC staff's generic evaluation of hydrodynamic loads in boiling-water reactor (BWR) facilities with the Mark I pressure-suppression containment design.

- Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 3, 1988, provides guidance for the preparation of license amendment requests to relocate cycle-specific TS information to the COLR.

3.0 TECHNICAL EVALUATION

3.1 Mechanical and Civil Engineering Review Considerations

3.1.1 Overview

The NRC staff's review, with respect to mechanical and civil engineering considerations, covered the structural integrity of SSCs important to safety designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section III, Division 1, and General Design Criteria (GDC) 1, 2, 4, 14, and 15. The NRC staff's review focused on the effects of the proposed amendment on the design input parameters and the design-basis loads and load combinations for normal, upset, emergency, and faulted conditions. The NRC staff's review covered: (1) the analyses due to applicable loads (including deadweight, earthquake, flow, temperature and pressure induced loads; and (2) the analytical methodologies and assumptions used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and cumulative fatigue usage factors against the Code-allowable limits.

3.1.2 High Energy Line Breaks

The NRC staff reviewed the licensee's summary of the evaluations performed regarding the effect that the proposed FWTR (due to both FFWTR and FWHOOS) has on the plant's design-basis high energy line break (HELB) events. The design-basis HELB events are discussed in Section 3.6 of the HCGS Updated Final Safety Analysis Report (UFSAR). As discussed in Section 6.4 of Reference 3, the licensee evaluated the following HELBs to address the effects of the proposed FWTR:

- Main Steam Line Break (MSLB) in the Main Steam Tunnel
- Feedwater Line Break (FWLB) in the Main Steam Tunnel
- Reactor Core Isolation Cooling (RCIC) steam line break

- High Pressure Coolant Injection (HPCI) steam line break
- Reactor Water Cleanup (RWCU) line break

The licensee's evaluations determined that the effect of the proposed FWTR on the postulated HELBs is bounded by the current design basis. In addition, the NRC staff found that the assumptions, methodologies and acceptance criteria utilized in the licensee's evaluations were acceptable. Based on these considerations, the NRC staff finds that there is reasonable assurance that the proposed FWTR will have no adverse impact with respect to its effect on postulated HELBs.

3.1.3 Annulus Pressurization Loads

The licensee evaluated the proposed FWTR effect on annulus pressurization (AP) loads. The AP dynamic loads result from a postulated circumferential pipe break at the interface of the reactor pressure vessel (RPV) nozzle safe-end and its connected piping that penetrates the biological shield (bioshield) wall.

GEH issued Safety Communication, SC 09-01, dated June 8, 2009, to address an error in the methodology that developed the generic AP loads, and lists HCGS as one of the affected plants. SC 09-01 identified issues with the determination of the asymmetric pressurization component of the AP loads (not related to the jet reaction, jet impingement and pipe whip components of the AP loads) and indicated that affected plants should consider reevaluating the AP loads to ensure they are consistent with the plant's design basis. In its supplement dated April 28, 2011, the licensee noted that according to UFSAR Appendices 3C and 6B, the HCGS plant design and licensing basis events for AP loads are the recirculation suction line break (RSLB) and the FWLB events only. The licensee stated that both events were evaluated for the proposed FFWTR and FWHOOS conditions consistent with SC 09-01. The NRC staff reviewed the results of the licensee evaluations as discussed below.

As discussed on page 4 of Attachment 1 of Reference 2, the licensee generated amplified response spectra (ARS) from acceleration time-history data at each break (SC 09-01 recommendation). These spectra were then peak-broadened by +/- 15% to account for uncertainty in the analysis consistent with the guidance in Regulatory Guide (RG) 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1, dated February 1978. The peak-broadened spectra were then compared across the various power/flow points and bounding spectra were created. A scaling factor of 1.6 was applied to these bounding spectra. As discussed on page 16 of Attachment 1 of Reference 2, the licensee used a 2% damping ratio for the FFWTR/FWHOOS AP ARS consistent with the damping ratio used for the original design ARS. The 2% damping ratio is consistent with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," dated October 1973, and is conservative with respect to the 4% damping ratio currently found acceptable as discussed in RG 1.61, Revision 1, dated March 2007.

The NRC staff reviewed the licensee's qualitative and quantitative summary evaluations presented in: (1) Section 4.1 of Attachment 1 of Reference 1; (2) Sections 4.0 and 5.0 of Reference 3; and (3) Reference 2. The staff found that the licensee adequately addressed the effect of the proposed FWTR on the licensing basis events for AP loads, consistent with the SC 09-01, applicable regulatory guidance and the plant's design basis. The licensee's summaries of its evaluations show that the structural integrity of the RPV, the RPV internals, piping and pipe supports, biological shield wall (BSW) and RPV pedestal (drywell inner skirt) will remain within applicable design limits.

3.1.4 RPV Internals

As described in Section 5.3 of Reference 3, and in Reference 2, the licensee evaluated the structural integrity of the RPV internals for the loads associated with the proposed FWTR. The licensee stated that all applicable loads such as seismic, AP, dead weight, reactor internal pressure differences (RIPDs), hydraulic/flow, thermal, acoustic and flow-induced due to a postulated RSLB LOCA were considered in the evaluation, as appropriate. The licensee evaluated RPV internals for normal, upset, emergency and faulted conditions. The licensee reported and discussed results of the stress analyses and fatigue assessments for critical RPV internals. The licensee stated that fatigue assessment is performed only for normal/upset (service level A/B) conditions. The NRC staff found the licensee's statement acceptable, as it is in conformance with the ASME Section III requirements. The following critical RPV internals were evaluated.

- Shroud
- Shroud Support
- Core Plate
- Top Guide
- Orificed Fuel Support
- Control Rod Guide Tube
- Control Rod Drive Housing
- Fuel Channel
- Steam Dryer
- Feedwater Sparger (see additional discussion below in SE Section 3.1.6)
- Jet Pump Assembly
- Core Spray Line and Sparger
- Access Hole Cover
- Shroud Head and Steam Separators Assembly
- In-Core Housing and Guide Tube
- Core Differential Pressure and Liquid Control Line
- Low Pressure Coolant Injection Coupling

The NRC staff reviewed the licensee's evaluation summaries and discussions presented in References 2 and 3. The results of the analyses show that critical RPV internals have satisfied the design basis ASME Section III Code-allowable stresses and the fatigue cumulative usage factor (CUF) allowable value of 1.0 for plant life extension (PLEX) to 60 years (PLEX-60). Therefore, the staff finds that the licensee has adequately demonstrated that the structural integrity of RPV critical internals is acceptable for the proposed FWTR.

3.1.5 Feedwater Nozzle

As described in Section 6.5 of Reference 3 and in Reference 2, the licensee evaluated the feedwater nozzle for fatigue usage due to the proposed FWTR. The licensee provided a discussion to justify that the FWTR does not have an effect on system cycling fatigue usage, but could adversely affect the rapid cycling fatigue usage. The NRC staff agrees with the licensee because the transient time duration is much longer for system cycling than for rapid cycling, which takes place at very short time durations, due to rapid temperature fluctuations. The licensee recalculated the rapid cycling fatigue usage factor for the proposed FWTR. In Reference 2, the licensee stated that the number of days used in the rapid cycling fatigue evaluation for days per year operation with FFWTR and FWHOOS is 128 days and 20 days, respectively. For the monitoring of cycling, the licensee stated that rapid cycling fatigue usage is included in the fatigue monitoring program. The licensee added the calculated rapid cycling usage, due to the proposed FWTR, to the current 40-year system cycling CUF and showed that it is less that the allowable value of 1.0. The licensee's discussion of results in Reference 3 stated that the projected fatigue usage will exceed 1.0 prior to reaching 60 years for the feedwater nozzle safe end, when applying environmental-assisted fatigue (EAF) factors (F_{en} factors) consistent with the recently approved HCGS License Renewal application (LRA). In addition, the licensee stated that the projection of exceeding 1.0 is not affected by the FWTR or FWHOOS. In Reference 2, the licensee stated that in recalculating the rapid cycling fatigue usage factor, it utilized the original design analysis methodology. The licensee provided 40-year and 60-year projected CUF values for the feedwater nozzle safe end and nozzle blend radius, which included system cycling and rapid cycling. The NRC staff reviewed the information provided by the licensee in Reference 2 and noted that the 60-year projected CUF values do not exceed the allowable value of 1.0, as mentioned in Reference 3. The staff concluded from its review that these values do not contain the LRA-required EAF effects and that when the F_{en} multipliers shown on LRA Table 4.3.5-1 (ADAMS Accession No. ML092430374) are utilized, the 60-year projected CUF values, as shown in Reference 2, will exceed the allowable value of 1.0.

The NRC staff's basis for finding the feedwater nozzle CUF exceedance of the allowable value of 1.0 acceptable is based on the following reasoning. The staff reviewed HCGS's LRA Table 4.3.5-1, "Environmental Fatigue Results for HCGS for NUREG/CR-6260 Components" and Section 4.3.5.1 of NUREG-2102, "Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station" (ADAMS Accession No. ML11200A221), which show that the RPV feedwater nozzle safe end has a PLEX-60 estimated value which exceeds the allowable value of 1.0. LRA Table 4.3.1-2 identifies the feedwater nozzle safe end and nozzle forging as items of the fatigue monitoring locations included in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, which provides for corrective actions to prevent the CUF from exceeding the design code limit of 1.0. The licensee, in Reference 2, stated that the fatigue monitoring program will ensure that the total fatigue usage, system cycling plus rapid cycling, remains less than the allowable limit and that the temperature limits will be monitored and constrained by operating procedures. The NRC staff finds that reasonable assurance exists regarding the structural integrity of the feedwater nozzle on the basis that the fatigue usage will be properly managed by the fatigue monitoring program.

3.1.6 Feedwater Sparger

As discussed on pages 5-4 and 5-5 of Reference 3, the evaluation of the feedwater sparger for fatigue usage found it acceptable for the 40-year plant design life (CUF less than 1.0). The licensee did not evaluate the feedwater sparger for fatigue usage to PLEX-60. The licensee stated that "the feedwater sparger does not meet any of the conditions in 10 CFR 54.4(a), and thus is not subject to the associated PLEX[-60] requirement." In Reference 2, the licensee stated that the feedwater sparger was evaluated against the License Renewal scoping criteria in 10 CFR 54.4(a). The licensee further stated that the feedwater sparger does not perform a safety-related function and its failure will not result in consequential failure of any safety-related equipment. The NRC staff verified the licensee's statements and found them acceptable by reviewing the HCGS LRA (including Table 2.3.1-4, "Reactor Internals Components Subject to Aging Management Review" and Table 3.1.2-2, "Reactor Internals Summary of Aging Management Evaluation") and NUREG-2102. The NRC staff, in NUREG-2102, Section 2.3.1.4, "Reactor Internals," concluded that "the applicant has appropriately identified the reactor internals mechanical components within the scope of license renewal, as required by 10 CFR 54.4(a)." The FW sparger is not part of the reactor internal components subject to aging management review in the HCGS LRA. Based on the discussion above, the NRC staff finds the licensee's evaluation of the feedwater sparger acceptable with respect to fatigue usage.

3.1.7 Mechanical and Civil Engineering Review Conclusion

Based on the discussion in SE Sections 3.1.2 through 3.1.6, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed FWTR on the structural integrity of SSCs important to safety. As such, the staff further concludes that there is reasonable assurance that the structural integrity of SSCs important to safety will continue to be maintained, consistent with the requirements of 10 CFR 50.55a and GDCs 1, 2, 4, 14, and 15, following implementation of the proposed amendment.

3.2 Containment Review Considerations

3.2.1 Overview

The HCGS primary containment is a Mark I design consisting of: (1) a drywell, which is a steel pressure vessel (in the shape of an inverted light bulb) that encloses the reactor vessel; (2) a pressure-suppression chamber (also called the wetwell or suppression pool), which is a torus-shaped steel pressure vessel that is partially filled with a large volume of water and is located below and encircling the drywell; and (3) a vent system connecting the drywell atmosphere to the wetwell. The NRC staff's review, with respect to containment considerations, focused on the effect of the proposed FWTR on the containment response to a design-basis accident (DBA) LOCA to ensure that the requirements in GDCs 16 and 50 would continue to be met.

As described in Section 3.0 of Reference 3, the licensee evaluated the short-term DBA-LOCA containment response to determine the effect of operation with up to a 102 °F FWTR. The licensee stated that the containment responses of concern for this analysis relate to: (1) peak drywell pressure; (2) peak drywell-to-wetwell pressure difference; and (3) hydrodynamic loads. The NRC staff reviewed the summary of the licensee's evaluation as discussed below.

3.2.2 Technical Evaluation

As described in Section 3.2 of Reference 3, the licensee's analysis used the GEH LAMB computer code to obtain the reactor pressure vessel blowdown flow rate and enthalpy and the GEH M3CPT computer code to obtain the containment response for the peak drywell pressure and the drywell-to-wetwell pressure difference. These codes also provide input to the determination of hydrodynamic loads. The use of both codes for this type of analysis has previously been found acceptable to the NRC staff as shown in Table 1-3 of Reference 3.

The power/flow map used for these analyses is the same as that used for the HCGS EPU. As noted above, the EPU amendment (HCGS Amendment No. 174) was approved by the NRC staff on May 18, 2008 (ADAMS package Accession No. ML081230540). The power/flow points used in the FWTR analyses represent the boundary points of the power/flow map. The licensee increased the value of power used in these analyses by 2% for conservatism. At these power/flow points, the licensee performed calculations with FWTR and compared the results at normal feedwater temperature (NFWT).

The results of the licensee's evaluation of the peak drywell pressure and the peak drywell-to-wetwell pressure difference are discussed in Section 3.3.1 of Reference 3 and on page 9 of Attachment 1 of Reference 1. The analysis demonstrated that the proposed FWTR operation at HCGS would continue to meet design limits for DBA-LOCA. The NRC staff finds these results to be acceptable since they were obtained using acceptable analytical methods and assumptions and are less than the respective design limits.

The licensee re-evaluated four LOCA-generated hydrodynamic loads as discussed in Section 3.3.2 of Reference 3. These are: (1) pool swell loads; (2) vent thrust loads; (3) condensation oscillation loads; and (4) chugging loads. These loads are described and explained in NUREG-0661, "Safety Evaluation Report - Mark I Containment Long-Term Program," dated July 1980, and HCGS UFSAR Appendix 3B, "Mark I Long-Term Program Plant Unique Analysis." The licensee stated in Reference 3 that it evaluated the effect of the FWTR on these loads by comparing the containment pressure and temperature responses calculated for the peak drywell pressure and the peak difference between the drywell and wetwell pressures with those used in the hydrodynamic load definition for HCGS to ensure that the loads with a FWTR remain within acceptable limits. The licensee's analysis concluded that the FWTR operation has no adverse effect on DBA-LOCA containment pressure and temperature response and that the current hydrodynamic loads definition for HCGS is not affected by FWTR operation. This method of evaluating hydrodynamic loads is acceptable to the NRC staff since the drywell and wetwell temperatures and pressures (along with other variables that are not affected by a change in feedwater temperature (such as the suppression pool volume)) determine the hydrodynamic loads.

The licensee also considered the effect of FWTR on the drywell head subcompartment pressurization. As discussed in Section 4.1.4 of Reference 3, this analysis assumes a RPV head vent line break. Since the break flow for a line break in this region is a steam break controlled by the RPV pressure, there is no effect due to FWTR because the RPV pressure (and therefore the steam break flow rate) is not changed by FWTR operation. Therefore, the current drywell head subcompartment pressurization analysis remains acceptable.

3.2.3 Containment Review Conclusion

Based on the discussion in SE Section 3.2.2, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed FWTR on the containment response to a DBA-LOCA. As such, the staff further concludes that there is reasonable assurance that the containment response will continue to be consistent with the requirements in GDCs 16 and 50 following implementation of the proposed amendment.

3.3 Reactor Systems Considerations

3.3.1 Overview

The NRC staff review, with respect to reactor systems considerations, covered a number of different technical areas including: the computer codes and methodologies used in the licensee's evaluations supporting the proposed amendment; ECCS performance; thermal-hydraulic stability; anticipated operational occurrence (AOO) performance; and anticipated transients without scram (ATWS) mitigation capability. The review focused on the effect of the proposed FWTR on the above technical areas to ensure the requirements of 10 CFR 50.46 and GDCs 10, 12, 20, 29, and 35 would continued to be met.

3.3.2 Computer Codes and Methodologies Used in Licensee's FWTR Evaluations

The primary computer codes used for the licensee's HCGS FWTR evaluations are listed in the Table 1-3 in Reference 3. The following is a discussion of the computer codes and methodologies used pertinent to the NRC staff's review of reactor systems considerations.

The GESTR-LOCA model (Reference 5) provides the parameters to initialize the fuel-stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA for input to SAFER. GESTR-LOCA also establishes the transient pellet-cladding gap conductance for input to both SAFER and TASC.

The SCAT/TASC computer model performs the transient short-term thermal-hydraulic calculation for large recirculation line breaks. Developed for GE11 and later fuel designs with part-length rods (PLRs), an improved SCAT model (designated "TASC") is used to predict the time and location of boiling transition and dryout. The time and location of boiling transition is predicted during the period of recirculation pump coastdown. When the core inlet flow is low, TASC also predicts the resulting bundle dryout time and location. The calculated fuel dryout time is an input to the long-term thermal-hydraulic transient model, SAFER.

The SAFER model calculates the long-term system response of the reactor over a complete spectrum of hypothetical break sizes and locations. SAFER models fuel rod gap conductance, fission gas release, and the code calculates the core and vessel water levels, system pressure response, ECCS performance and other primary thermal-hydraulic phenomena during an accident as a function of time. SAFER models all regimes of heat transfer that occur inside the core and calculates the appropriate heat transfer coefficients and the resulting peak cladding temperature (PCT) as a function of time. The SAFER model treats part-length rods as full-length rods that conservatively overestimate the hot bundle power.

The LAMB code analyzes the short-term blowdown phenomena for postulated large pipe breaks in which nucleate boiling is lost before the water level drops sufficiently to uncover the active fuel. The LAMB output (primarily core flow as a function of time) is used in the SCAT model for calculating blowdown heat transfer and fuel dryout time.

PANAC is the three-dimensional core physics code used for design, licensing, and core monitoring of the BWR cores. PANAC correctly handles varying axial geometry in nuclear and thermal-hydraulic modeling through the use of its lattice-dependent geometry, nodal thermal-hydraulic properties, and axial-meshing routines. This flexibility allows PANAC to handle multiple PLRs, varying rod diameter and other axially varying features.

ISCOR is a thermal-hydraulic core analysis program where different fuel types can be designated to represent various types of bundles in a core. The introduction of various PLR rod heights can be readily handled by ISCOR since parameters can be varied axially to account for changes in the number of rods, water rod diameters, etc., in the lattice at different axial locations.

ODYSY is a best-estimate, engineering computer program which incorporates a linearized, small perturbation, frequency domain model of the reactor core and associated coolant circulation system. It is based on the ODYN transient analysis code. It will predict both core-wide mode coupled thermal-hydraulic and reactor kinetic instabilities, and single channel thermal-hydraulic instabilities. ODYSY has been applied for licensing calculations for Option I-D and Option II stability long-term solutions (LTS). ODYSY has also been approved for backup stability protection evaluations in the Option III and detect and suppress - confirmation density LTS solutions.

ODYN is a transient reactor analysis code based on one-dimensional neutronics model and a void quality correction fluid model. The application of the ODYN methodology and ODYSY for expanded operating domains was approved by the NRC staff. The ODYN and kinetics models in ODYSY form the basis for determining the forward and feedback transfer functions used in the stability analysis.

The NRC staff finds that the licensee has used all appropriate computer codes and methodologies. The application of these codes for analyses has been found complying with the limitations, restrictions and conditions specified in the respective NRC SE for the codes and methodologies used.

3.3.3 ECCS Performance Analysis

The effect of FWTR of up to 102 °F on the ECCS performance at HCGS was evaluated by the licensee using the NRC-approved SAFER/GESTR-LOCA methodology (Reference 5). The application methodology consists of: (1) nominal calculations performed over the break spectrum and for various break locations; and (2) conformance calculations for the limiting break per Appendix K of 10 CFR Part 50. The PCT determined in (2) represents the margin for licensing evaluations.

The limiting break and failure combination for HCGS was evaluated with the maximum FWTR for Appendix K and nominal assumptions using an approved set of ECCS parameters at the various power and flow conditions described in Section 2.0 of Reference 3. Results of PCT from the analysis determined that the current HCGS licensing basis PCT of 1380 °F will continue to be applicable and bounding for FWTR of up to 102 °F. Therefore, the 10 CFR 50.46 acceptance criteria continue to be met for the proposed amendment.

The NRC staff has evaluated the results of the ECCS performance analysis and determined that the criteria stipulated in 10 CFR 50.46 are met for FFWTR with GE14 fuel at HCGS.

3.3.4 Thermal-Hydraulic Stability

HCGS is Option III stability plant (Reference 6) which effectively detects either core-wide or regional modes of reactor instability by combining closely spaced local power range monitors into cells which are called oscillation power range monitor (OPRM) cells. The period-based detection algorithm associated with the Option III licensing basis provides an instrument setpoint that is designed to trip the reactor before an oscillation can grow to the point where the safety limit minimum critical power ratio (SLMCPR) is exceeded. The Option III stability-based operating limit minimum critical power ratio (OLMCPR) calculation is based on the delta critical power ratio over initial minimum critical power ratio versus the oscillation magnitude (DIVOM) curves in Reference 7. Appendix B of Reference 7 concludes that significant variations in feedwater temperature have very little effect on the slope of the DIVOM curve. This means that the stability-based OLMCPR values, including penalties, will not change due to reduced feedwater temperature operation. However, significant reduction in feedwater temperature will result in higher decay ratios that increase the size of the area susceptible to instability. In order to assure stability, the OPRM trip-enabled region should encompass the backup stability protection (BSP)-controlled entry region as discussed in Section 6.3.2 of Reference 3.

If the OPRM system is declared inoperable, the Option III stability solution uses BSP. The BSP solution uses the NRC-approved PANACEA/ODYSY methodology (Reference 8) that creates two BSP regions: Region I (scram region); and Region II (controlled entry region). BSP regions are determined for both normal feedwater temperature (NFWT) and reduced feedwater temperature on a reload-specific basis.

BSP regions with respect to a feedwater temperature of 329.6 °F (FFWTR) and for a feedwater temperature of 371.6 °F (FWHOOS) are demonstrated in Figures 6-1, and 6-2 of Reference 3, respectively. The BSP analysis for FWHOOS operation allows the Option III OPRM trip-enabled region power and flow boundaries to remain at their current values of 26.1% of rated core power and 60.0% of rated core flow. For the proposed FFWTR (329.6 °F) operation, the Option III OPRM trip-enabled region power boundary remains at its current value of 26.1% of rated core power. However, the flow boundary would need to be increased from 60% to 70% of rated core flow. This change would encompass the regions susceptible to instability for reduced feedwater temperature. The Option III trip enabled region is assessed for each reload cycle.

The NRC staff has reviewed the licensee's evaluation with respect to thermal-hydraulic stability. The staff finds that the cycle-specific calculations, performed with approved methodologies, provide reasonable assurance that thermal-hydraulic stability will be maintained following implementation of the proposed amendment.

3.3.5 AOO Performance

GDC 10 states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

Operating limits are established to ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The SLMCPR ensures that 99.9% of the fuel rods are protected from boiling transition during steady-state operation. The OLMCPR assures that the SLMCPR will not be exceeded as result of an AOO.

In Section 6.1 of Reference 3, the licensee provided a summary of the results of its evaluations for the following AOOs in justifying FWTR operation for HCGS:

- Feedwater Controller Failure - increasing flow
- Rod Withdrawal Error
- Fuel Loading Error
- Loss of Feedwater Heating
- Main Steam Isolation Valve Closure with Flux Scram
- Load Rejection with No Bypass
- Turbine Trip with No Bypass

Each of these AOOs is discussed below.

Feedwater Controller Failure - Increasing Flow (FWCF)

The licensee stated that the only AOO that requires consideration in assessing the effect of FWTR on operating limits is FWCF. This is based on the licensee's finding that the other AOOs are less sensitive to a reduction in FW temperature. The licensee further indicated that, at normal FW temperature conditions, the end-of-cycle (EOC) Load Rejection with No Bypass and Turbine Trip with No Bypass AOOs are more limiting than the EOC FWCF with normal FW temperature and with FWTR. As such, the licensee concluded that no penalty on the rated power OLMCPR is required while operating at FWTR. As also discussed in Section 6.1 of Reference 3, the licensee will evaluate the FWTR effect on the change in critical power ratio (ΔCPR) on a cycle-specific basis.

The NRC staff reviewed the licensee's Supplemental Reload Licensing Report (SRLR) for the current operating cycle at HCGS, Cycle 17 (Reference 9). Page 5 of the SRLR states the basis for the report is the "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-16 dated October 2007. This topical report (also known as GESTAR II) is referenced in HCGS TS 6.9.1.8, "Core Operating Limits Report," as a methodology approved by the NRC as being applicable for determination of the HCGS core operating limits. Section 8 of the SRLR indicates that the analysis assumed a 60 °F FW temperature reduction during the cycle and a FFWTR of 102 °F (i.e., consistent with proposed amendment). Section 9 of the SRLR provides the core-wide AOO analysis results. Section 11 of the SRLR provides a summary of the cycle-specific OLMCPR values. The NRC staff confirmed that the licensee's statements in Section 6.1 of Reference 3 (discussed above) were consistent with the results of the analysis for the current operating cycle. The staff finds that the cycle-specific evaluations, performed with approved methodologies, provide reasonable assurance that the OLMCPR values will be determined consistent with the intent of GDC 10.

As noted above, the licensee stated that the only AOO that requires consideration in assessing the effect of FWTR on operating limits is FWCF. The NRC staff evaluated the licensee's basis for not evaluating the other AOOs (with respect to FWTR) as discussed below.

Rod Withdrawal Error (RWE)

The licensee stated that "the most important parameters affecting the RWE transient response are the initial control rod pattern and the error rod position, both of which are not affected by the RFWT [reduced FW temperature] operating condition."

The NRC staff agrees with the licensee's basis for not evaluating this AOO under FWTR conditions. The staff also notes that this justification is consistent with analysis performed for several other plants that are authorized for FWTR operation.

Fuel Loading Error (FLE)

The licensee stated that "[t]he fuel loading error (FLE) consequences are the result of a power mismatch between the correctly and incorrectly loaded fuel. This power mismatch is independent of operating conditions."

The FLE is predominantly affected by the R-factor uncertainty change, and the R-factor change due to the mis-oriented fuel bundle. The R-factor is a number which characterizes the local peaking pattern relative to any given rod. The increase in inlet subcooling due to the reduced FW temperature causes the core power to increase and moves the boiling boundary higher in the core. The result is that the axial power will shift lower in the core. This will have a slight affect on the local peaking distribution at the new boiling boundary. However, the effect on R-factor is minor because the bundle R-factor is the axial averaged rod R-factor and this small change in the peak power represents only a tiny fraction of the total fuel length. FWTR does not change the limiting FLE. As such, the NRC staff finds that the licensee's justification for not evaluating this AOO under FWTR conditions is acceptable.

Loss of Feedwater Heating (LFWH)

The licensee stated that "[t]he loss of feedwater heating (LFWH) event is a core wide transient that is driven by the magnitude of the decrease in feedwater temperature. Initializing from a RFWT reduces the feedwater perturbation and the severity of the LFWH event."

With an initial temperature of 329.6 °F, the additional subcooling of 100 °F from a LFWH event will not have a significant impact. As such, the NRC staff finds that the licensee's justification for not evaluating this AOO under FWTR conditions is acceptable. The staff also notes that this justification is consistent with analysis performed for several other plants that are authorized for FWTR operation.

Main Steam Isolation Valve Closure with Flux Scram (MSIVF)

The licensee stated that "[t]he limiting event for vessel overpressure considerations, main steam isolation valve closure with flux scram (MSIVF), is bounded by the same event analyzed at NFWT [normal FW temperature]. This conclusion is based on the reduced steam generation rate associated with the FWTR condition that results in a milder vessel pressurization transient during the MSIVF event, as compared to that for NFWT."

The NRC staff agrees with the licensee's basis for not evaluating this AOO under FWTR conditions. The staff also notes that this justification is consistent with analysis performed for several other plants that are authorized for FWTR operation.

Load Rejection with No Bypass (LRNBP) And Turbine Trip With No Bypass (TTNBP)

The licensee stated that "[s]imiliar to the MSIVF event, the load rejection with no bypass (LRNBP) and turbine trip with no bypass (TTNBP) events are bounded by the same event at NFWT due to the reduced steam generation rate associated with the FWTR condition. The reduction in steam flow reduces the pressurization rate, which results in a lower peak power from a milder void collapse and neutron flux increase."

The NRC staff agrees with the licensee's evaluation that both the LRNBP and TTNBP events at FWTR conditions will be less severe than the same events at NFWT. FWTR does not change the limiting LRNBP or TTNBP; therefore, the NRC staff finds that the licensee's justification for not evaluating these AOOs under FWTR conditions is acceptable.

AOO Performance Conclusion

Based on its review as discussed above, the NRC staff agrees with the licensee's assessment that the limiting AOO, with respect to the effect of FWTR on operating limits, is FWCF. The staff further finds that the cycle-specific evaluations, performed with approved methodologies, provide reasonable assurance that the OLMCPR values will be determined consistent with the intent of GDC 10. As such, the NRC staff concludes that the proposed amendment is acceptable with respect to AOO performance.

3.3.6 ATWS Mitigation Capability

As discussed in Section 6.2 of Reference 3, the licensee stated that the effect of FWTR operation on anticipated transient without scram (ATWS) performance has previously been evaluated on a generic basis. The licensee stated that these evaluations have shown that peak values for fuel surface heat flux, vessel bottom pressure, and suppression pool temperature were all reduced when the FW temperature was reduced.

The licensee also stated that, as a result of FWTR, the steam generation rate and core void fraction are reduced. The lower steam generation rate increases the ratio of steam flow rate through the relief valves to steam generation rate, and therefore, the peak vessel pressure is lower. There is also less steam released to the suppression pool so the pool heats up less.

The licensee concluded and the NRC staff agrees that an ATWS at normal FW temperature bounds the results of ATWS under FWTR. The staff also notes that this conclusion is consistent with analysis performed for several other plants that are authorized for FWTR operation.

3.3.7 Reactor Systems Review Conclusion

Based on the discussion in SE Sections 3.3.2 through 3.3.6, the NRC concludes that the licensee has adequately addressed the effects of the proposed FWTR with respect to the reactor systems topics discussed above. As such, the staff further concludes that there is reasonable assurance that the requirements of 10 CFR 50.46 and GDCs 10, 12, 20, 29, and 35 would continued to be met following implementation of the proposed amendment.

3.4 FOL and TS Changes

FOL Changes

As discussed in SE Section 2.2.1, the proposed amendment would revise License Condition 2.C(11) to allow HCGS to operate with a FW temperature as low as 329.6 °F. Based on the discussion in SE Sections 3.1 through 3.3, the NRC staff concludes that the licensee's analysis and evaluations support HCGS operation under this reduced FW temperature. Therefore, the NRC staff further concludes this proposed change to the FOL is acceptable.

TS Changes

As discussed in SE Section 2.2.2, the proposed amendment would revise SR 4.3.11.5 associated with testing of the OPRM. This SR ensures that trips initiated from the OPRM system are not inadvertently bypassed when the capability of the OPRM system to initiate a reactor protection system trip is required. Currently, this SR requires that the licensee verify that the OPRM is enabled when THERMAL POWER is greater than or equal to 26.1% RTP and recirculation drive flow is less than or equal to the value corresponding to 60% of rated core flow. The frequency of this verification is performed in accordance with the HCGS Surveillance Frequency Control Program. PSEG is proposing that the SR be revised to relocate the value of the rated core flow parameter to the COLR. The SR would be revised to reference the COLR as the source of the rated core flow value. In addition, the SR would be revised to state that the value specified in the COLR shall not be less than 60% of rated core flow.

The licensee stated in its application dated September 22, 2010, that the value of the rated core flow required to bound the region susceptible to instability is determined on a cycle-specific basis and will vary depending upon the magnitude of the FWTR implemented for a particular operating cycle. The licensee's application stated that the rated core flow value associated with SR 4.3.11.5 would be determined in accordance with NRC-approved methods.

The requirements for establishing and documenting the core operating limits are specified in HCGS TS 6.9.1.9, "Core Operating Limits Report." This TS lists TS 3/4.3.11, "Oscillation Power Range Monitor (OPRM)," as one of the TSs pertaining to the COLR. TS 6.9.1.9 requires that the analytical methods used to determine the core operating limits be those previously reviewed and approved by the NRC and lists 3 specific methodologies including GESTAR II and the OPRM Option III methodology (NEDO-32465-A). The NRC staff finds that the requirements in TS 6.9.1.9 provide reasonable assurance that the value of the rated core flow associated with SR 4.3.11.5 will be determined such that facility operation will continue to protect public health and safety. The NRC further finds that that relocation of the core flow value from the SR to the COLR is consistent with the requirements in 10 CFR 50.36(c)(3) and the guidance in GL 88-16. Based on the above considerations, the NRC staff concludes that the proposed changes to SR 4.3.11.5 are acceptable.

PSEG's application dated September 22, 2010, provided proposed changes to the TS Bases to be implemented with the associated TS changes. The TS Bases pages were provided for information only and will be revised in accordance with the HCGS TS Bases Control Program.

3.5 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 through 3.4, 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 amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes 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 (76 FR 1466). 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.

7.0 REFERENCES

1. PSEG letter (LR-N10-0343) to NRC dated September 22, 2010, "License Amendment Request - Operation with Final Feedwater Temperature Reduction and Feedwater Heaters Out-of-Service" (ADAMS Package Accession No. ML102790111).

2. PSEG letter (LR-N11-0122) to NRC dated April 28, 2011, "Response to Request for Additional Information, License Amendment Request - Operation with Final Feedwater Temperature Reduction and Feedwater Heaters Out-of-Service" (ADAMS Accession No. ML11130A050).

3. GEH Report NEDC-33506P, "Hope Creek Generating Station Operation with Final Feedwater Temperature Reduction and Feedwater Heaters Out-Of-Service," dated September 2010 (Attachment 4 to Reference 1, ADAMS Accession No. ML102790107, non-publicly available).

4. GEH Report NEDO-33506, "Hope Creek Generating Station Operation with Final Feedwater Temperature Reduction and Feedwater Heaters Out-Of-Service," dated September 2010 (Attachment 5 to Reference 1, ADAMS Accession No. ML102790106).

5. General Electric (GE) Report NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," dated October 1984 (ADAMS Accession No. ML102230240, non-publicly available).

6. GE Nuclear Energy Report NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," dated November 1995 (ADAMS Legacy Library Accession No. 9603130121).

7. GE Nuclear Energy Report NEDO-32645-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated August 1996 (ADAMS Legacy Library Accession No. 9609230148).

8. GEH Report NEDE-33213P-A, "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," dated April 2009 (ADAMS Accession No. ML091100203, non-publicly available).

9. PSEG letter (LR-N10-0290) to NRC dated August 3, 2010, "Reload 16 Cycle 17 Supplemental Reload Licensing Report," (ADAMS Package Accession No. ML102240335).

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Date: September 14, 2011

Attachment: List of Acronyms

ATTACHMENT
LIST OF ACRONYMS

ACRONYM	DEFINITION
ADAMS	Agencywide Documents Access and Management System
AOO	anticipated operational occurrence
AP	annulus pressurization
ARS	amplified response spectra
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BSP	backup stability protection
BSW	biological shield wall
BWR	boiling-water reactor
COLR	Core Operating Limits Report
CPPU	constant pressure power uprate
CPR	critical power ratio
CUF	cumulative usage factor
DBA	design-basis accident
DIVOM	delta CPR over initial minimum CPR versus the oscillation magnitude
EAF	environmentally-assisted fatigue
ECCS	emergency core cooling system
EOC	end-of-cycle
EPU	extended power uprate
FFWTR	final feedwater temperature reduction
FLE	fuel loading error
FOL	Facility Operating License
FR	Federal Register
FW	feedwater
FWCF	feedwater controller failure - increasing flow
FWHOOS	feedwater heaters out-of-service
FWLB	feedwater line break
FWTR	feedwater temperature reduction
GDC	General Design Criteria
GE	General Electric
GEH	GE - Hitachi Nuclear Energy Americas LLC
GL	Generic Letter
HCGS	Hope Creek Generating Station
HELB	high energy line break
HPCI	high pressure coolant injection
LCO	Limiting Condition for Operation
LFWH	loss of feedwater heating
LOCA	loss-of-coolant accident
LRA	License Renewal application
LRNBP	load rejection with no bypass
LTS	long-term solutions
MSIVF	main steam isolation valve closure with flux scram

ACRONYM	DEFINITION
MSLB	main steam line break
NFWT	normal feedwater temperature
NRC	Nuclear Regulatory Commission or the Commission
OLMCPR	operating limit minimum critical power ratio
OPRM	Oscillation Power Range Monitor
PCT	peak cladding temperature
PLEX	plant life extension
PLR	part-length rod
PSEG	PSEG Nuclear LLC
RCIC	reactor core isolation cooling
RFWT	reduced feedwater temperature
RG	Regulatory Guide
RIPD	reactor internal pressure difference
RPV	reactor pressure vessel
RSLB	recirculation line break
RTP	Rated Thermal Power
RWCU	reactor water cleanup
RWE	rod withdrawal error
SE	safety evaluation
SLMCPR	safety limit minimum critical power ratio
SR	surveillance requirement
SRLR	Supplemental Reload Licensing Report
SSC	structure, system and component
TS	Technical Specification
TTNBP	turbine trip with no load bypass
UFSAR	Updated Final Safety Analysis Report

September 14, 2011

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

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AND FEEDWATER HEATERS OUT-OF-SERVICE (TAC NO. ME4786)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 190 to Renewed Facility Operating License (FOL) No. NPF-57 for the Hope Creek Generating Station (HCGS) in response to your application dated September 22, 2010, as supplemented by letter dated April 28, 2011.

The amendment allows HCGS to operate at a reduced feedwater temperature for purposes of extending the normal fuel cycle. The amendment also allows operation with feedwater heaters out-of-service at any time during the operating cycle. In addition, the amendment revises surveillance requirements related to testing of the Oscillation Power Range Monitor.

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. 190 to Renewed License No. NPF-57
2. Safety Evaluation

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JGall, SRXB

ADAMS Accession No: ML112380618 *via e-mail

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