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PSEG
*Nuclear LLC*
10 CFR 50.90

SEP 22 2010

LR-N10-0343
LAR H10-04

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Hope Creek Generating Station
Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: **LICENSE AMENDMENT REQUEST – OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AND FEEDWATER HEATERS OUT-OF-SERVICE**

In accordance with 10CFR50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Facility Operating License No. NPF-57 for Hope Creek Generating Station (HCGS). In accordance with 10 CFR 50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The amendment request proposes changes to paragraph 2.C (11) of the Facility Operating License (FOL). The proposed change would allow HCGS to operate with Final Feedwater Temperature Reduction (FFWTR) for the purposes of extending the normal fuel cycle. In addition, the analysis provided would also allow operation with Feedwater Heaters Out-Of-Service (FWHOOS) at any time during the operating 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 August 2010. The analyses and evaluations support 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.

This amendment request also proposes changes to Technical Specification (TS) Surveillance Requirement (SR) 4.3.11.5, related to testing of the Oscillation Power Range Monitor (OPRM). SR 4.3.11.5 requires, per an 18 month frequency, 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 less than or equal to the value corresponding to 60% of rated core flow. As discussed in the attachments to this LAR, 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 feedwater temperature reduction (FWTR) implemented

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NRR

for a particular operating cycle. The magnitude of the FWTR will be within the bounds stated above. Consequently, for the implementation of FFWTR/FWHOOS at HCGS, 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).

Finally, relative to the stability application, HCGS will be referencing the *ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions,* NEDE-33213P-A, Revision 0, GE Hitachi Nuclear Energy (Proprietary), April 2009 for the implementation of FFWTR/FWHOOS and for reference in subsequent cycle specific reload analyses.

No new regulatory commitments are established by this submittal.

Attachment 1 to this letter provides an evaluation supporting the proposed changes. The marked-up Operating License and TS pages, with the proposed changes indicated, are provided in Attachment 2 to this letter. Attachment 3 provides the marked-up TS Bases pages, for information only. Attachment 4 to this letter provides GEH Report NEDC-33506P, "Hope Creek Generating Station Operation With Final Feedwater Temperature Reduction and Feedwater Heaters Out-Of-Service," dated September 2010, which GEH considers to contain proprietary information. The proprietary information is identified by bracketed text. GEH requests that the proprietary information in Attachment 4 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). A signed affidavit supporting this request is also included in Attachment 4. Attachment 5 to this letter provides a nonproprietary version of the GEH Report (NEDO-33506).

PSEG requests approval of the proposed change by October 1, 2011, with the amendment being implemented within 90 days of issuance.

If you have any questions or require additional information, please do not hesitate to contact Mr. Jeff Keenan at (856) 339-5429.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 9/22/10
(Date)

Sincerely,

John F. Perry
Site Vice President
Hope Creek Generating Station

Attachments (5)

Attachments (5)

M. Dapas, Regional Administrator (Acting) - NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
P. Mulligan, Manager IV, NJBNE
Commitment Coordinator – Hope Creek
PSEG Commitment Coordinator - Corporate

# LICENSE AMENDMENT REQUEST (LAR) H10-04 – OPERATION WITH FINAL FEEDWATER TEMPERATURE REDUCTION AND FEEDWATER HEATERS OUT-OF-SERVICE

## Table of Contents

## 1.0 DESCRIPTION

The license amendment request (LAR H10-04) proposes changes to paragraph 2.C (11) of the Facility Operating License (FOL). The proposed change would allow Hope Creek Generating Station (HCGS) to operate with Final Feedwater Temperature Reduction (FFWTR) for the purposes of extending the normal fuel cycle. In addition, the analysis provided would also allow operation with Feedwater Heaters Out-Of-Service (FWHOOS) at any time during the operating cycle.

To support the amendment request, analyses and evaluations have been prepared by PSEG Nuclear LLC (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 (Attachment 4 of this submittal). The analyses and evaluations support 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 431.6°F to 329.6°F; the 60°F temperature reduction corresponds to 371.6°F.

This amendment request also proposes changes to Technical Specification (TS) Surveillance Requirement (SR) 4.3.11.5, related to testing of the Oscillation Power Range Monitor (OPRM). SR 4.3.11.5 requires, per an 18 month frequency, 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 less than or equal to the value corresponding to 60% of rated core flow. As discussed in Section 4 of this attachment, 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 feedwater temperature reduction (FWTR) implemented for a particular operating cycle. The magnitude of the FWTR will be within the bounds stated above. Consequently, for the implementation of FFWTR/FWHOOS at HCGS, 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).

Finally, relative to the stability application, HCGS will be referencing for the first time the *ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions,* NEDE-33213P-A, Revision 0, GE Hitachi Nuclear Energy (Proprietary), April 2009, for the implementation of FFWTR/FWHOOS, and for reference in subsequent cycle specific reload analyses.

## 2.0 PROPOSED CHANGE

2.1 Currently FOL Condition 2.C (11) states:

> <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 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.

PSEG proposes to revise the License Condition to state:

> 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.

2.2 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 once per 18 months.

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 once per 18 months. The value specified in the CORE OPERATING LIMITS REPORT shall not be less than 60% of rated core flow.

A copy of the affected HCGS Operating License and TS pages marked-up to show the proposed changes identified above are provided in Attachment 2. The corresponding changes to the TS Bases are provided in Attachment 3. The changes to the affected TS Bases pages will be incorporated in accordance with TS 6.15 "Technical Specifications (TS) Bases Control Program."

## 3.0 BACKGROUND

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 offers cycle extension for a given fuel reload by maintaining rated reactor thermal power through the reactivity inserted by 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. 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). License Condition 2.C (11) was most recently revised as part of the HCGS EPU Amendment, 174, Extended Power Uprate (EPU) [ADAMS ML081230640].

Amendment 174 made changes to License Condition 2.C(11), as discussed below.

*"The following is the current operating license condition 2.C(11):*

*The facility shall not be operated with reduced feedwater temperature for the purpose of extending the normal fuel cycle. After the first operating cycle, the facility shall not be operated with a feedwater heating capacity that would result in a rated power feedwater temperature less than 400°F unless analyses supporting such operation are submitted by the licensee and approved by the staff.*

*The licensee proposed to change the current license condition to the following:*

*The facility shall not be operated with reduced FW 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 HCGS design FW temperature at CPPU conditions is 431.6°F. HCGS has been evaluated for operation with a FW 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 staff finds this proposed change acceptable."*

## 4.0 TECHNICAL ANALYSIS

Final Feedwater Temperature Reduction (FFWTR) is implemented to extend the operating cycle with rated feedwater temperature reduction limited to 102°F. Feedwater Heaters Out-of-Service (FWHOOS) is a plant operating flexibility option allowing continued operation with less than the full feedwater system heating capacity available during the operating cycle with rated feedwater temperature reduction limited to 60°F.

To support the amendment request, analyses and evaluations have been prepared by PSEG and 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 (Attachment 4). The FFWTR evaluation results bound FWHOOS[1]. This submittal refers to feedwater temperature reduction (FWTR), which encompasses both FFWTR and FWHOOS. Evaluations that apply specifically to FFWTR or FWHOOS are explicitly stated in this submittal.

The NEDC-33506P analyses and evaluations determined that the effect of FFWTR/FWHOOS on the following subjects is acceptable: 1) ECCS Performance, 2) Containment System Performance 3) Reactor Asymmetric Loads, 4) Reactor Coolant and Connected Systems, 5) AOO Performance, 6) ATWS Mitigation Capability, 7) Thermal-Hydraulic Stability Performance, 8) High Energy Line Break, 9) Feedwater Nozzle Fatigue, and 10) P-Bypass Setpoint.

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[1] There is one exception to the bounding relationship; as discussed in Section 6.5 of NEDC 33506P, the feedwater nozzle safe end rapid cycling fatigue usage is slightly higher for FWHOOS than FFWTR (0.071 versus 0.070).

A detailed discussion of each of the above ten subjects is provided in Attachment 4. The following topics merit further discussion in support of the amendment request.

## 4.1 GEH SC09-01 Review for FFWTR/FWHOOS

FWTR affects Annulus Pressurization Loads.

GE Hitachi Nuclear Energy (GEH) has issued a 10 CFR Part 21 Safety Information Communication: SC 09-01, Annulus Pressurization Loads Evaluation, dated June 8, 2009. SC 09-01 identifies a potential issue with the methodology that developed Annulus Pressurization (AP) loads. SC 09-01 states that "the AP loads used as input for design adequacy evaluations of NSSS safety related components for "New Loads" plants might have resulted in non-conservative evaluations." SC 09-01 contains the following corrective action:

> "Plants on the affected plant list should review their design and licensing basis in light of the issue presented above and consider reevaluating the AP loads to ensure that they are consistent with the plant's design basis."

As discussed in NEDC-33506P Sections 4 and 5, GEH and PSEG evaluated the effect of FWTR on AP loads. The structural responses and amplified response spectra (ARS) of the RPV, reactor internals, piping, biological shield wall (BSW), and RPV pedestal (drywell inner skirt) were evaluated due to the application of Recirculation Suction Line Break (RSLB) and Feedwater Line Break (FWLB) loads. The results of these evaluations are presented below according to their relevance to the HCGS Design Basis and GEH SC 09-01.

### 4.1.1 RSLB

For the existing HCGS design, the RSLB mass and energy releases were calculated using the GEH LAMB code. AP pressure time histories were generated by COMPARE code and used for structural evaluations. Bounding mass and energy conditions were determined to be MELLLA minimum recirculation pump speed assuming a rated feedwater temperature of approximately 400°F.

For FWTR, the RSLB mass and energy releases were calculated using the GEH LAMB code. AP pressure time histories were generated by COMPARE code and used for structural evaluations. Mass and energy releases were calculated for various conditions along the power to flow map, including minimum recirculation pump speed, MELLLA and increased core flow (ICF). The analysis for AP Loads for FWTR did not make the implicit assumption that the maximum overall mass/energy release will result in maximizing the structural response. This approach was used in response to SC 09-01.

The results of the FWTR RSLB analysis show that the FWTR analysis structural loads on the RPV and reactor internals are bounded by the design basis loads.

Composite ARS spectra for FWTR operating conditions were generated for the RSLB break locations in response to SC 09-01. The results of the FWTR RSLB analysis show that the FWTR ARS responses are not bounded by the design basis at certain locations on the BSW and RPV pedestal. However, these responses remain lower than the maximum values at peak accelerations for the governing load case due to RPV rupture. Further detailed evaluation determined that all downstream components are within design limits.

4.1.2 FWLB

For the existing HCGS design, the FWLB is considered to be bounded by the RSLB and therefore a specific analysis of FWLB is not performed.

For FWTR, the FWLB mass and energy releases were calculated using the GEH LAMB code. AP pressure time histories were generated by COMPARE code and used for structural evaluations. Mass and energy releases were calculated for various conditions along the power to flow map, including minimum recirculation pump speed, MELLLA and increased core flow (ICF). This approach was used in response to SC 09-01.

The results of the FWTR FWLB analysis show that the FWTR analysis structural loads on the RPV and reactor internals are bounded by the design basis loads.

Composite ARS spectra for FWTR operating conditions were generated for the FWLB break locations in response to SC 09-01. The results of the FWTR FWLB analysis show that the FWTR ARS responses are bounded by the design basis at representative locations on the RPV, internals, BSW and RPV pedestal.

4.1.3 Other line breaks

Other line breaks are currently outside of the existing HCGS design basis.

For FWTR, other break locations were assessed considering the RSLB and FWLB results and recent EPU/MELLLA analyses. Based on the more rigorous and updated EPU/MELLLA analyses methods, and the location and size of the Recirculation and Feedwater lines, it is expected that any structural impact due to other break locations would be minor with no adverse design impact. Additional actions related to SC 09-01 are being addressed by the BWROG.

4.2 <u>Proposed Revision to SR 4.3.11.5</u>

This amendment request also proposes changes to TS SR 4.3.11.5, related to testing of the Oscillation Power Range Monitor (OPRM). SR 4.3.11.5 requires, per an 18 month frequency, 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 less than or equal to the value corresponding to 60% of rated core flow.

As discussed in Attachment 4, NEDC-33506P Section 6.3, 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 (the magnitude of the FWTR will be within the bounds stated in Section 4.0 above). NEDC-33506P Section 6.3 also provides demonstration analyses performed to illustrate the effect of FWTR on the adequacy of the OPRM trip enabled region. Depending on the magnitude of the cycle specific FWTR, the rated core flow boundary may need to be expanded greater than the current TS SR 4.3.11.5 value of 60% rated core flow. Consequently, for the implementation of FFWTR/FWHOOS at HCGS, PSEG is proposing that the SR be revised to relocate the value of the rated core flow parameter to the COLR. Transferring the value of the requirement is consistent with Generic Letter 88-16. GL 88-16 permitted the relocation of fuel cycle specific parameter limits to the COLR. The COLR requirements are

established in TS 6.9.1.9, which lists TS 3/4.3.11, OPRM, as one of the TS to be included in the COLR. The SR value of the cycle specific core flow parameter for the OPRM trip enabled region will be determined in accordance with NRC approved methods, ensuring that the Safety Limit Minimum Critical Power Ratio (SLMCPR) will be protected if a thermal hydraulic stability event were to occur. SR 4.3.11.5 will maintain the current "not less than 60%" restriction on the rated core flow parameter.

The HCGS administrative control for developing the COLR will require that the cycle specific value of the OPRM trip enable region core flow boundary cannot have a FWTR basis that exceeds 102°F for FFWTR or 60°F for FWHOOS. HCGS operating procedures that control operation relative to maintaining feedwater temperature within the design basis will be updated, as required.

4.3 Low Power Scram Bypass Setpoint

As discussed in NEDC-33506P Section 6.6 (Attachment 4 of this submittal), HCGS evaluated the existing low power scram bypass setpoint, based on turbine first stage pressure and the calculated change in steam flow. At a reduced feedwater temperature, HCGS concluded that the reactor scram bypass setting for turbine first stage pressure was not sufficiently conservative relative to the TS value of 24% rated thermal power. Therefore a new setpoint of approximately 21.4% has been calculated. The new set-point increases the low power bypass set-point conservatism at normal feedwater temperature (NFWT) and maintains the same conservatism at FFWTR conditions.

4.4 ODYSY Application

Relative to the stability application, HCGS will be referencing for the first time the *ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions,* NEDE-33213P-A, Revision 0, GE Hitachi Nuclear Energy (Proprietary), April 2009, for the implementation of FFWTR/FWHOOS, and for reference in subsequent cycle specific reload analyses. As indicated in the Attachment 4 (Table 1-3) listing of computer codes used, HCGS is referencing the most current NRC approved topical report for the determination of the backup stability protection regions.

## 5.0 REGULATORY ANALYSIS

### 5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated consistent with the requirements of General Design Criteria (GDC) 4, 10, 12, 15, 16, 20, 29, 35, and 50.

### 5.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," PSEG Nuclear LLC (PSEG) requests an amendment to Facility Operating License No. NPF-57 for Hope Creek Generating Station (HCGS). The amendment request proposes changes to paragraph 2.C (11) of the Facility Operating License (FOL). The proposed change would allow HCGS to operate with Final Feedwater Temperature Reduction (FFWTR) for the purposes of extending the normal fuel cycle. In addition, the analysis provided would

also allow operation with Feedwater Heaters Out-Of-Service (FWHOOS) at any time during the operating cycle. The term 'feedwater temperature reduction' (FWTR) encompasses both FFWTR and FWHOOS.

To support the amendment request, analyses and evaluations have been prepared by PSEG Nuclear LLC 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. The analyses and evaluations support 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 431.6°F to 329.6°F; the 60°F temperature reduction corresponds to 371.6°F.

This amendment request also proposes changes to Technical Specification (TS) Surveillance Requirement (SR) 4.3.11.5, related to testing of the Oscillation Power Range Monitor (OPRM). SR 4.3.11.5 requires, per an 18 month frequency, 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 less than or equal to the value corresponding to 60% of rated core flow.* For the implementation of FFWTR/FWHOOS at HCGS, 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). Transferring the value of the requirement is consistent with Generic Letter 88-16.

In addition, relative to the stability application, HCGS will be referencing for the first time the *ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions,* NEDE-33213P-A, Revision 0, GE Hitachi Nuclear Energy (Proprietary), April 2009, for the implementation of FFWTR/FWHOOS, and for reference in subsequent cycle specific reload analyses. NEDE-33213P-A, Revision 0, is the most current NRC approved topical report which describes the methods that are acceptable for use in determining the backup stability protection regions.

PSEG has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three conditions set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The effect of FWTR on the probability and consequences of accidents, Anticipated Operational Occurrences (AOO) and events in the Updated Final Safety Analysis (UFSAR) were reviewed.

   The impact of FWTR on the Design Basis Accident (DBA) Loss Of Coolant Accident (LOCA), was considered. Evaluations and analyses were performed to determine that the current Licensing Basis PCT remains applicable for operation of HCGS with FWTR. The calculated maximum fuel element cladding temperature does not exceed 2,200°F, the calculated total local oxidation does not exceed 17% times the total cladding thickness, the calculated total amount of hydrogen generated from a chemical reaction of the cladding with

water or steam is less than 1% times the hypothetical amount if all the metal in the cladding cylinder were to react, the core remains amenable to long term cooling, and there is sufficient long term core cooling available. Analysis also demonstrated that FWTR operation at HCGS continues to meet design limits for the DBA-LOCA peak drywell pressure and temperature. Therefore, there is no increase in the consequence of an accident previously evaluated in the UFSAR.

The only AOO that requires consideration in assessing the effect of FWTR on event consequences is the feedwater controller failure – increasing flow (FWCF). This is based upon the finding that the other AOOs are less sensitive to a reduction in feedwater temperature. The rated power and off-rated Power Distribution Limits, Critical Power Ratio and Linear Heat Generation Rate, for the FWCF event are validated on a cycle specific basis to ensure compliance with the Safety Limit Minimum Critical Power Ratio (SLMCPR) and compliance with the fuel rod thermal mechanical acceptance criteria of avoiding fuel centerline melt and 1% cladding plastic strain. Consequently, there is no increase in the consequences of an AOO previously evaluated.

The impact of FWTR on the consequences of the following events was also considered: Anticipated Transient Without Scram (ATWS), vessel overpressure, thermal-hydraulic stability, and High Energy Line Break (HELB). The evaluation of ATWS and vessel overpressure concluded that the consequences of the events at normal feedwater temperature remain bounding for FWTR. The evaluation of HELB determined the impact was bounded by the current design basis. Thermal-hydraulic stability considerations, as impacted by FWTR, involve both the determination of a cycle specific OPRM setpoint and determination of a cycle specific backup stability protection (BSP) regions and corresponding adequacy of the OPRM trip enabled region. The cycle specific determinations and validations performed in accordance with NRC approved methods ensure that the SLMCPR will be protected if a thermal hydraulic stability event were to occur. Therefore, there is no increase in the consequence of these events previously evaluated in the UFSAR.

In addition, the following areas were also evaluated. The reactor power level and operating pressure are not changed. FWTR has no effect on the decay heat. Current design limits associated with long-term containment analyses, including RSLB, loss of offsite power (LOOP), intermediate break accident (IBA), small break accident (SBA), and NUREG-0783 safety relief valve (SRV) steam discharge events continue to be supported without change. Therefore, there is no increase in the consequence of these events previously evaluated in the UFSAR.

The probability of an accident is not affected by the proposed changes since no structures, systems or components (SSC) which could initiate an accident are affected. Therefore, the proposed changes do not significantly increase the probability of any previously evaluated accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not alter the design function of any SSC. The implementation of FWTR operation does not create the possibility of a new or different kind of accident. Power Distribution Limits on CPR, LHGR and APLHGR, and OPRM setpoints, which are determined in accordance with NRC approved methods and are included in the Core Operating Limits Report (COLR), as part of the normal reload licensing process will continue to assure that core operation is in accordance with the conditions currently assumed for event initiation. FWTR was reviewed against the accidents, AOOs and events in the UFSAR and it was determined there would be no adverse impact; the existing design basis remains bounding. In addition, the proposed changes do not involve new system interactions or equipment modifications to the plant. FWTR does not involve any new type of testing or maintenance. Therefore there are no new design basis failure mechanisms, malfunctions, or accident initiators created by the proposed changes.

   The existing low power scram bypass setpoint, based on turbine first stage pressure and the calculated change in steam flow was evaluated. At a reduced feedwater temperature, it was concluded that the reactor scram bypass setting for turbine first stage pressure was not sufficiently conservative relative to the TS value of 24% rated thermal power. Therefore a new setpoint of approximately 21.4% has been calculated. The new set-point increases the low power bypass set-point conservatism at normal feedwater temperature (NFWT) and maintains the same conservatism at FFWTR conditions.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No.

   The AOOs and accidents described in the UFSAR were evaluated for effects caused by the reduced feedwater temperature. For cycle independent considerations, the evaluations determined that the consequences of the events are either bounded by the current design and licensing basis results, are within design acceptance criteria, or will not change in a manner that would reduce the margin of safety. For cycle specific considerations, cycle specific analyses utilizing NRC approved methods that produce the values of the limits documented in the COLR will continue to assure that core operation is maintained within the within the existing design basis and safety limits. No design basis or safety limit is altered by the proposed change.

   The existing low power scram bypass setpoint, based on turbine first stage pressure and the calculated change in steam flow was evaluated. At a reduced feedwater temperature, it was concluded that the reactor scram bypass setting for turbine first stage pressure was not sufficiently conservative relative to the TS value of 24% rated thermal power. Therefore a new setpoint of approximately 21.4% has been calculated. The new set-point increases the low power bypass set-point conservatism at NFWT and maintains the same conservatism at FFWTR conditions.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PSEG concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

### 5.3 *Conclusions*

In conclusion, based on the considerations discussed above, (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.

## 6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 7.0 REFERENCES

References for this submittal are provided in Attachment 4.

## Mark-up of Proposed Operating License and Technical Specification Pages

The following Technical Specifications and Facility Operating License pages for Facility Operating License NPF-57 are affected by this change request

| **Facility Operating License** | **Page** |
|---|---|
| License Conditions 2.C (11) | 5 |

| **Technical Specification** | **Page** |
|---|---|
| SR 4.3.11.5 | 3/4 3-110 |

(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 Section 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 ~~reduced~~ a rated thermal power feedwater temperature less than 329.6 °F for the purpose of extending the normal fuel cycle. ~~unless analyses supporting such operation are submitted by the licensee and approved by the staff.~~

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

a. PSE&G shall submit for staff review Detailed Control Room ~~Design Review Summary Reports II and III on a schedule~~ consistent with, and with contents as specified in, its letter of January 9, 1986.

b. Prior to exceeding five percent power, PSE&G shall provide temporary zone markings on safety-related instruments in the control room.

## 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 at least once per 184 days.

4.3.11.2 Calibrate the local power range monitor once per 1000 Effective Full Power Hours (EFPH) in accordance with Note f, Table 4.3.1.1-1 of TS 3/4.3.1.

4.3.11.3 Perform CHANNEL CALIBRATION once per 18 months. Neutron detectors are excluded.

4.3.11.4 Perform LOGIC SYSTEM FUNCTIONAL TEST once per 18 months.

4.3.11.5 Verify OPRM is enabled when THERMAL POWER is ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to ~~60%~~ the percentage as specified in the CORE OPERATING LIMITS REPORT of rated core flow once per 18 months. 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. 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. Neutron detectors are excluded.

* 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.

## Mark-up of Proposed Technical Specification Bases Pages

The following Technical Specification Bases pages for Facility Operating License NPF-57 are affected by this change request

| TS Bases | Page |
|---|---|
| SR 4.3.11.5 | B3/4 3-17 |

3/4.3.11 Oscillation Power Range Monitor (OPRM)

SURVEILLANCE REQUIREMENTS (continued)

SR 4.3.11.5

This SR ensures that trips initiated from the OPRM system are not inadvertently bypassed when the capability of the OPRM system to initiate an RPS trip is required. The trip capability of the OPRM system is only required during operation under conditions susceptible to anticipated T-H instability oscillations. The region of anticipated oscillation is defined by THERMAL POWER ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to **the percentage** ~~60%~~ of rated core flow **as specified in the COLR. The value specified in the CORE OPERATING LIMITS REPORT shall not be less than 60% of rated core flow.**

The trip capability of individual OPRM modules is automatically enabled based on the APRM power and flow signals associated with each OPRM channel during normal operation. These channel specific values of APRM power and recirculation drive flow are subject to surveillance requirements associated with other RPS functions such as APRM flux and flow biased simulated thermal power with respect to the accuracy of the signal to the process variable. The OPRM is a digital system with calibration and manually initiated tests to verify digital input including input to the auto-enable calculations. Periodic calibration confirms that the auto-enable function occurs at appropriate values of APRM power and recirculation flow signal. Therefore, verification that OPRM modules are enabled at any time that THERMAL POWER ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to **the percentage** ~~60%~~ of rated core flow **as specified in the COLR (not less than 60%)** adequately ensures that trips initiated from the OPRM system are not inadvertently bypassed. **Note 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 feedwater temperature reduction (FWTR) implemented for a particular operating cycle**.

The trip capability of individual OPRM modules can also be enabled by placing the module in the non-bypass (Manual Enable) mode. If placed in the non-bypass or Manual Enable mode the trip capability of the module is enabled and this SR is met. The frequency of 18 months is based on engineering judgment and reliability of the components.

SR 4.3.11.6

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis (Ref. 8). The OPRM self-test function may be utilized to perform this testing for those components it is designed to monitor. The RPS RESPONSE TIME acceptance criteria are included in Reference 8.

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. RPS RESPONSE TIME tests are conducted 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. This Frequency is based upon operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.