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

November 5, 2008

Mr. Thomas Joyce
President & Chief Operating Officer
PSEG Nuclear
Post Office Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION – CORRECTION TO THE SAFETY EVALUATION FOR AMENDMENT NO. 174 RE: EXTENDED POWER UPRATE (TAC NO. MD3002)

Dear Mr. Joyce:

On May 14, 2008 (Agencywide Documents Access and Management System Accession No. ML081230540), the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 174 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consisted of changes to the Facility Operating License and the Technical Specifications (TSs) in response to your application dated September 18, 2006,[1] as supplemented by additional letters.[2] The amendment increased the authorized maximum power level by approximately 15 percent, from 3,339 megawatts thermal (MWt) to 3,840 MWt.

Due to administrative errors, the NRC determined that several pages of the related safety evaluation (SE) need to be corrected. Enclosure 1 contains the bases for the corrections. Enclosure 2 contains the corrected pages to the SE. The corrected pages to the SE do not change the NRC staff conclusions regarding Amendment No. 174 for the Hope Creek Generating Station.

Please discard the associated pages from the previous SE and replace them with the pages in Enclosure 2.

1 Agencywide Documents Access and Management System (ADAMS) Accession Number ML062680451
2 October 10, 2006 (ML062920092); October 20, 2006 (ML063110164); February 14, 2007 (ML070530099); February 16, 2007 (ML070590182); February 28, 2007 (ML070680314); March 13, 2007 (ML070790508 & ML070810360); March 22, 2007 (ML070930442); March 30, 2007 (ML071010243 & ML070960103); April 13, 2007 (ML071140157); April 18, 2007 (ML071160121); April 30, 2007 (ML071290559); May 10 (ML071360375); May 18, 2007 (ML071500294, ML071720368, & ML071500317); May 24, 2007 (ML071630305); June 22, 2007 (ML071840167); July 12, 2007 (ML072110215); August 3, 2007 (ML072250369); August 17, 2007 (ML072480515); August 27, 2007 (ML072480570); August 31, 2007 (ML072540651); September 11, 2007 (ML072640410); October 10, 2007 (ML080580475); October 23, 2007 (ML073040393); November 15, 2007 (ML073320601); November 30, 2007 (ML073460793); December 31, 2007 (ML080080577); January 14, 2008 (ML080230069); January 15, 2008 (ML080250028); January 16, 2008 (ML080290663); January 18, 2008 (ML080280531); January 25, 2008 (ML080360467); January 30, 2008 (ML080420468); March 18, 2008 (ML080860477 & ML080870082); and May 2, 2008 (ML081270387)

If you have any questions, please call me at 301-415-3100.

Sincerely,

John G. Lamb, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Bases for corrections to the SE
2. Corrected pages to SE

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BASES FOR CORRECTIONS TO THE SAFETY EVALUATION

FOR AMENDMENT NO. 174 - EXTENDED POWER UPRATE

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

On page 10, "60 ft-lbs" was changed to "66 ft-lbs" based on Table 3-2, "Hope Creek Upper Shelf Energy – 40 Year Life (32 EFPY)," contained in NEDO-33076, Revision 2, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate," August 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062690086).

On page 10, "0.15 percent" was changed to "0.14 percent" based on Table 3-2 contained in NEDO-33076, Revision 2 (ADAMS Accession No ML062690086).

On page 18, the following changes were made based on a letter dated January 16, 2008, (ML080290663) and an e-mail dated October 15, 2008 (ADAMS Accession No ML082890255):

- "18 Category C welds" was changed to "17 Category C welds"
- "9 RPV recirculation inlet nozzle to safe end welds" was changed to "8 RPV recirculation inlet nozzle to safe end welds"
- "two Category E welds" was changed to "three Category E welds"
- "two weld overlay repaired welds" was changed to "three weld overlay repaired welds"

On page 52, "Secondary Condensate Pump 85 percent permissive" was changed to "Secondary Condensate Pump 75 percent permissive" based on a letter dated January 16, 2008 (ADAMS Accession No ML080290663).

On page 53, "RBM instrumentation" was deleted since the licensee did not request any technical specification changes for the rod block monitor instrumentation.

On page 64, "10 days" was changed to "8 days" in two places based on a letter dated March 22, 2007 (ADAMS Accession No ML070930442).

On page 134, the following sentences were deleted based on Figure 7.7-6 and Section 15.1.4.3.3 of the Updated Final Safety Analysis Report:
"Automatic recirculation flow control system increases the recirculation flow to the maximum to compensate the power reduction. Reactor power settles out at nearly the initial power level. Because the recirculation flow control can not meet the additional load demand, the pressure regulator set is automatically reduced to a lower limit, and the reactor pressure decreases eventually."

On page 140, the following was deleted: "calculated PCT for Two-Loop Operation." It was replaced with "licensing acceptance criteria." This correction was based Section 5.3.3, "Single-Loop Operation (SLO)," contained in NEDO-33172, "SAFER/GESTR-LOCA Loss of Coolant Analysis for Hope Creek Generating Station at Power Uprate" (ADAMS Accession No ML053250466).

On page 185, "Control room unfiltered intake, cfm - 350" was replaced with "Control room normal flow rate, cfm – 3300" based on a letter dated October 20, 2006 (ADAMS Accession No ML063110163).

On page 186, "Control room unfiltered intake rate, cfm - 350" was replaced with "Control room normal flow rate, cfm – 3300" based on a letter dated October 20, 2006 (ADAMS Accession No ML063110163).

On page 187, "Control room unfiltered intake rate, cfm - 350" was replaced with "Control room normal flow rate, cfm – 3300" based on a letter dated October 20, 2006 (ADAMS Accession No ML063110163).

On page 193, the following sentence was changed from:
"...source strength for a 15 percent increase in steam flow is expected to increase radiation levels due to N^{16} concentration at steam turbine components by approximately 16 percent for operation at 3840 MWt."
to
"...source strength for a 15 percent reduction in transit time is expected to increase radiation levels due to N^{16} concentration at steam turbine components by approximately 16 percent for operation at 3952 MWt, which bounds operation at 3840 MWt."
This change is based on a letter dated June 22, 2007 (ADAMS Accession No ML071840167).

On pages 243 and 244, the following was deleted since LCO 3.3.4.2, "End-of-Cycle Recirculation Trip System Instrumentation, Applicability," is not related to average bundle power: "The 25 percent RTP value is based on a generic analysis for all BWR plants with the highest average bundle power for 100 power original power level. However, the proposed EPU average bundle power for HCGS is higher than that previously assumed in the analysis. The new 24 percent RTP was established based on the new average bundle power. The analysis with the new bundle power is reviewed and accepted by the staff and is documented in Section 2.8, "Reactor Systems," of this safety evaluation."

CORRECTED PAGES TO THE SAFETY EVALUATION

FOR AMENDMENT NO. 174 - EXTENDED POWER UPRATE

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

The following pages were corrected to the safety evaluation for Amendment No. 174, "Extended Power Uprate," for the Hope Creek Generating Station. Please replace the following pages, in their entirety.

Page
10
18
52
53
64
134
140
185
186
187
193
243
244

Enclosure 2

the Hope Creek EPU are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of Power Uprate Review Standard RS-001.[27]

Technical Evaluation

The ¼ T fluence is the fluence value at ¼ T from the Inside Diameter (ID) of the vessel with T being the vessel thickness. The ¼ T fluence is used for the evaluation of Pressure – Temperature (P – T) curves and Upper Shelf Energy (USE). The ¼ T fluence includes EPU conditions.

Upper-Shelf Energy (USE) Value Calculations

Appendix G of 10 CFR Part 50 provides the NRC's criteria for maintaining acceptable levels of USE for the reactor vessel beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires reactor vessel beltline materials to have a minimum USE value of 75 foot-pound force (ft-lb) in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analyses that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI to the ASME Code. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant reactor vessel surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H reactor vessel materials surveillance program.

The licensee for Hope Creek discussed the impact of the Hope Creek EPU on the Charpy USE values for the reactor vessel beltline materials in Section 3.2.1 of the PUSAR.[28] Table 3-2, "Hope Creek Upper Shelf Energy - 40 Year Life (32 EFPY)," pp 3-35 of the Hope Creek PUSAR, indicated that the projected Charpy USE for the limiting plate (intermediate shell plate, heat 5K3025) is 66 ft-lbs, and the projected Charpy USE for the limiting weld (intermediate-lower shell-to-intermediate shell circumferential submerged arc weld, heat D55733) is 60 ft-lbs. However, the NRC staff noted that in Table 3-2, heat 10024/1 for the low-pressure coolant injection (LPCI) nozzle forging specifies a copper content of 0.14 percent. In addition, the Hope Creek UFSAR, Appendix 5A, Tables 5A-5 and 5A-19 specifies a copper content of 0.14, while the NRC Reactor Vessel Integrity Database (RVID) specifies a copper content of 0.35 percent for the LPCI forging. In response to an RAI, the licensee, in its letter dated March 13, 2007,[29] confirmed that for heat 10024/1, the copper content is 0.14 percent. This is based on the General Electric Report GE-NE-523-A164-1294R1, Tables 7-2 and 7-3. The NRC staff confirmed that the copper content is 0.14 percent based on the report and will use the reported value to update the RVID copper value for this heat of material.

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," has two methods for determining the percent reduction in Charpy USE. In Position 1.2, the percent reduction in Charpy USE is determined from Figure 2 in RG 1.99, Revision 2, which is based on the neutron fluence and the amount of copper in the material. In the second method, identified as Position

[27] ADAMS Accession No. ML033640024
[28] Attachment 4, page 3-3 of PSEG Letter (LR-N06-0286) to NRC dated September 18, 2006, "Request for License Amendment Extended Power Uprate, Hope Creek Generating Station Facility, Operating License NPF-57, Docket No. 50-354" ADAMS Accession No. ML062680451
[29] PSEG Letter (LR-N-07-0035) to NRC dated March 13, 2007, "Response to Request for Additional Information - Request for License Amendment – Extended Power Uprate" ADAMS Accession No. ML070790508

In response to the NRC staff's RAI provided in a letter dated March 22, 2007,[45] the licensee stated that Hope Creek was designed, fabricated, and constructed per the guidance in NUREG-0313, Revision 2, so most welds are IGSCC Category A welds which are resistant to IGSCC. However, there are 22 welds that are considered susceptible to IGSCC. These are 17 Category C welds (8 RPV recirculation inlet nozzle to safe end welds, two RPV recirculation outlet nozzle to safe end welds, two CS safe end to safe end extension welds, one RPV CS inlet nozzle to safe end weld, one RPV CRD nozzle to cap weld, one RPV head spray nozzle to flange weld, and two RPV jet pump instrumentation nozzle to safe end), two Category B welds (recirculation to decontamination line weldolets) and three Category E welds (weld overlay repaired welds). Hope Creek's IGSCC augmented inspection program is based on BWRVIP-75-A, "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules." The licensee further explained that for IGSCC to occur, three conditions must exist: the existence of a susceptible material, the presence of tensile stresses and the presence of an oxidizing environment. Operation at CPPU conditions will result in somewhat higher pressure, temperature, and flow for some systems comprising portions of the RCPB, but these changes will have negligible effect on the tensile stresses. Therefore, CPPU operation will not affect the material's susceptibility to IGSCC. Operation at a higher power level will result in a slightly higher oxygen generation rate due to radiolysis of water. However, as discussed later, steps will be taken to ensure that RCPB piping will continue to be mitigated from an oxidizing environment perspective. Since the three conditions that cause IGSCC to occur are essentially unchanged for CPPU conditions, the IGSCC augmented inspection program will remain the same for the Hope Creek EPU.

The licensee stated that Hope Creek has three weld overlay repaired welds (reactor vessel CS nozzle to safe end weld (N5B), reactor vessel recirculation inlet nozzle to safe end weld (N2A), and reactor vessel recirculation inlet nozzle to safe end weld (N2K). The weld overlay repairs were designed to ASME Code Section XI requirements. The CPPU operating conditions have no affect on the overlay repair designs because the changes in pressure, temperature and flow rate resulting from CPPU operation are considered insignificant at those locations and are bounded by the overlay design analysis. Thus, the three weld overlay repaired welds are considered adequate for Hope Creek EPU operation.

The licensee stated that several mitigation processes have been applied to Hope Creek to reduce the RCPB component's susceptibility to IGSCC. These include the use of IGSCC resistant materials, application of mechanical stress improvement process (MSIP) and the implementation of HWC with NobleChem. All Category C welds (18) and Category B welds (2) were applied with MSIP. The effectiveness of MSIP and IGSCC resistant materials are not affected by the proposed Hope Creek EPU.

A NobleChem application was performed during cycle 13 refueling in April 2006. A mitigation monitoring system including iron, and platinum ECP electrodes, and 24 durability coupons (catalyst loading) was installed in January 2006. A hydrogen benchmark test was conducted following the cycle 13 reactor start-up in May 2006. All secondary parameters were also benchmarked to provide correlation with measured ECP. The molar ratio data based on EPRI Radiolysis/ECP Model was used to monitor the ECP condition at the most limiting location in the vessel, currently defined as the upper downcomer. Following the Hope Creek EPU implementation, the licensee will perform a second hydrogen benchmark test to determine the appropriate injection level, and will update the radiolysis/ECP Model and run cases to validate

[45] ADAMS Accession No. ML070930442

Stator Water Cooling System	Flow Orifice, Flow Meter, and Winding Inlet Pressure Gauge were replaced to accommodate increase Stator Water pressure and flow required for the increased generator rating
High Pressure Turbine Instrumentation	Replace instrumentation to accommodate the HP Turbine replacement
Main Steam Line Flow Instrumentation	Rescale the instrument to accommodate the input to the NSSS Isolation Logic in psid (Mlbs/hr) for EPU range.
Main Steam Line Flow Instrumentation	Rescale the instrumentation to accommodate the input to the Digital Feedwater Control System for EPU range in psid (Mlbs/hr).
Main Steam Line Flow Recorders, Indicators, Computer points	Rescale the instrumentation to accommodate the EPU range in Mlb/hr
Feedwater Flow Recorder, computer points	Rescale the instrumentation to accommodate the EPU range in Mlb/hr
Condensate pre-filter flow	Rescale the instrumentation to accommodate the EPU range in gpm
Condensate demineralizer flow	Rescale the instrumentation to accommodate the EPU range in gpm
Hydrogen Water Chemistry Injection	Setpoint is revised in terms of FW flow because of increase in total rated FW flow, but remains same in terms of percent rated thermal power
Primary Condensate Pump 75 percent permissive	The setpoint is revised because of the increase in total rated flow and full-scale range.
Secondary Condensate Pump 75 percent permissive	The setpoint is revised because of the increase in total rated flow and full-scale range.
Reactor Core Isolation Cooling turbine exhaust pressure	Change setpoint to ensure system availability for the duration assumed for the SBO event.
Neutron Monitoring	Re-calibrate APRM and RBM to reflect EPU operation

Electrohydraulic Control and Turbine Supervisory Instrumentation	Replace instrumentation.
APRM flow biased trip reference card	Replace this card to accommodate the ARTS/MELLA changes.

The above instrument changes will be made to accommodate the revised process parameters at Hope Creek EPU operating conditions. Since the instrumentation and control functions related to the above changes will be confirmed by the licensee during post-modification testing, power ascension testing, and instrument calibration, as applicable, the NRC staff has reasonable assurance that the instrumentation will continue to perform their intended process and safety functions at Hope Creek EPU operating conditions.

Instrument Setpoint Methodology

The licensee has requested TS changes associated with instrument setpoint or AVs related to APRM flow biased reactor trip, and MS Line Isolation on High Flow with this amendment request. In Section 5.3, "Technical Specification Instrument Setpoints," of the PUSAR,[84] the licensee states that none of these instruments perform a function related to the protection of a TSs SL. Therefore, the proposed changes to the TSs setpoints do not involve a limiting safety system setting (LSSS) that protects a plant SL.[85] The staff reviewed the licensee's setpoint methodology to calculate the nominal trip setpoints, acceptable as-left (AAL) band and acceptable as-found (AAF) band for these instruments. The nominal trip setpoint is established at a value which is more conservative than limiting trip setpoint. The AAL which the licensee has defined as desired range/recalibration tolerance is established by taking the square root of the sum of the squares of calibration tolerance and vendor accuracy numbers. The AAF value is established by taking the square root of the sum of the squares of the calibration tolerance, measurement and test equipment uncertainties and drift numbers. The NRC staff finds that the licensee's methodology to calculate these numbers meets the guidance provided in the RIS 2006-17[86] and therefore is acceptable to the staff.

The licensee has further stated that the instrument channel calibration is performed using approved surveillance procedures which identify the calibration tolerances. Instrument channels are calibrated at the nominal trip setpoint. If during the calibration the instrument exceeds the desired range/recalibration tolerance (AAL band) but is below the acceptable value (AAF band), the instrument will be re-calibrated. However, if the instrument is found to be outside the acceptable value (AAF band) it will also be entered in the corrective action program. If the instrument is found outside the AV, then it will be declared inoperable and the action required by

[84] Attachment 4, Page 5-8 of PSEG Letter (LR-N06-0286) to NRC dated September 18, 2006, "Request for License Amendment Extended Power Uprate, Hope Creek Generating Station Facility, Operating License NPF-57, Docket No. 50-354" ADAMS Accession No. ML062680451

[85] Attachment 1, page 22-23 of PSEG Letter (LR-N06-0286) to NRC dated September 18, 2006, "Request for License Amendment Extended Power Uprate, Hope Creek Generating Station Facility, Operating License NPF-57, Docket No. 50-354" ADAMS Accession No. ML062680451

[86] NRC Regulatory Issue Summary, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels." August 24, 2006. ADAMS Accession No. ML051810077

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool (SFP) provides wet storage of spent fuel assemblies. The safety function of the fuel pool cooling and cleanup system (FPCCS) is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review of the FPCCS for the proposed power uprates focused on the effects of the proposed uprate on the capability of the system to provide adequate cooling for the spent fuel during all operating and accident conditions. The criteria that are most applicable to the staff's review of the FPCCS for proposed power uprates are based primarily on GDC 61, "Fuel Storage and Handling and Radioactivity Control," insofar as it requires that fuel storage systems be designed with residual heat removal capability reflecting the importance to safety of decay heat removal (DHR); and other licensing basis considerations that are applicable. The staff's review of the FPCCS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5. Acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 9.1.3 of the Hope Creek UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The licensee evaluated the FPCCS in Section 6.3 of the PUSAR for Hope Creek. The components that are necessary for performing the cooling function include two surge tanks, two half-capacity FPC water pumps, two half-capacity fuel pool heat exchangers, and associated piping, valves, and instrumentation. The system also has a cross-connection with the RHR system which allows the RHR system to provide supplemental cooling of the spent fuel. When the RHR system is operated in parallel with the FPCCS to provide FPC during a full core offload, one RHR pump takes its suction either from the skimmer surge tanks or from the reactor vessel via the shutdown cooling (SDC) suction piping, circulates the water through one RHR heat exchanger, and returns it to the SFP via the two RHR inter-tie return diffusers.

UFSAR Section 9.1.3.1 indicates that the FPCCS is designed to maintain pool temperature less than or equal to 135°F following a batch core offload (approximately one third of the core) at the end of a fuel cycle assuming a limiting heat load with all other fuel storage locations filled from previous refuelings. This limiting heat load is currently 16.1×10^6 British thermal units per hour (BTU/hr) at 8 days after reactor shutdown. The licensee determined that the new limiting heat load for CPPU operation is 17.2×10^6 BTU/hr at 8 days after shutdown.[97] The licensee stated that the two FPCCS heat exchangers were modified in 1990 from 72 plates to 99 plates per heat exchanger, which increased the design heat transfer capability of each FPCCS heat exchanger from 6.0×10^6 BTU/hr to 9.5×10^6 BTU/hr for a combined heat transfer capability of 19×10^6 BTU/hr. Therefore, the SFP heat load for CPPU operation is well within the combined design heat transfer capability of the FPCCS heat exchangers eight days after shutdown, and the licensing-basis criterion to maintain the SFP temperature less than or equal to 135°F following a batch core offload will continue to be satisfied following CPPU implementation.

[97] Response to BOP Branch Question 7.1 in PSEG letter (LR-N07-0056) to NRC dated March 22, 2007, "Response to Request for Additional Information Request for License Amendment - Extended Power Uprate" ADAMS Accession No. ML070930442

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the FWC system using water from the CST via the condenser hotwell. The NRC staff's review covered: (1) the sequence of events; (2) the analytical model used for analyses; (3) the values of parameters used in the analytical model; and (4) the results of the transient analyses. The NRC's acceptance criteria are based on: (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Inadvertent opening of a safety/relief valve will cause a decrease in reactor coolant inventory and result in mild depressurization. The pressure regulator senses the reactor pressure decrease and closes the TCV far enough trying to maintain constant reactor vessel pressure.

This event will have a slight effect on fuel thermal margins. Changes in surface heat flux are expected to be negligible indicating an insignificant change in the MCPR. According to ELTR1, the bounding event for this category (decrease in reactor coolant inventory) is LOFW. Thus, this transient is not listed in the minimum required tests in ELTR1 and hence not analyzed.

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26

As stated earlier, the CLTP core at HCGS consists of GE and SVEA-96+ fuel types. For the first EPU core (Cycle 15), there will be predominantly GE14 fuel with some remaining average thrice burned legacy fuel (SVEA-96+). In response to the staff's RAI, by letter dated March 13, 2007, the licensee stated that the SVEA fuel operating in the Cycle 15 core will be high exposure, low reactivity fuel in its fourth or fifth operating cycle. The SVEA peak bundle power will be significantly lower than that of the limiting GE14 fuel. It was stated that based on this lower power, the results of the Cycle 15 EPU core design demonstrate that the GE14 fuel is limiting for MAPLHGR (which protects PCT) for the entire operating cycle. In response to the staff's RAI, by letter dated March 30, 2007, the licensee further stated that the limiting GE14 fuel will be operating at peak exposure values consistent with the maximum (or near maximum) LHGR limit, and therefore consistent with the limiting (or near limiting) PCT, during Cycle 15. Therefore, it is expected that the SVEA PCT will be bounded by the GE14 PCT for operating cycle 15. This will be confirmed by the licensee for the cycle-specific core, and the results are documented in the SRLR for the cycle. The staff finds this acceptable.

In addition to the large break LOCA analysis, the small break LOCA response was reanalyzed using a sufficient number of break sizes in order to assure adequate ADS capacity. The licensee stated that the plant-specific analyses demonstrate the adequacy of the ADS performance at EPU conditions, and that small break LOCA event mitigation is acceptable.

For Single Recirculation Loop Operation (SLO), a multiplier is applied to the Two-Loop Operation LHGR and MAPLHGR limits. Application of the appropriate LHGR/MAPLHGR multiplier for SLO operation assures the expected SLO PCT is less than the licensing acceptance criteria.

The EPU will make a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46 (local cladding oxidation, core-wide metal-water reaction, coolable geometry). Long term cooling is assured when the core remains flooded to the jet pump top elevation and when a CS system is operating.

Based on licensee's plant-specific LOCA analysis for HCGS EPU condition with equilibrium core, and because the licensee will perform plant cycle-specific evaluations of ECCS-LOCA performance for HCGS first EPU cycle using approved methods, as required in Section 5.2 of ELTR-2, the staff agrees with the licensee that the HCGS ECCS-LOCA performance complies with 10 CFR 50.46 and Appendix K requirements.

As confirmatory evaluations, the staff performed audit calculations. As discussed above, because it is expected that the SVEA PCT will be bounded by the GE14 PCT for the EPU cycles at HCGS, the staff used only GE14 fuel to perform their LOCA audit calculations. The results of the staff's calculations are summarized below:

<u>Audit Calculation</u>

The staff performed audit calculations using the RELAP5 code to assess ECCS performance for the HCGS NSSS. The double-ended recirculation line break was reported by HCGS as the limiting break size. The audit calculation is to confirm that the PCT value reported by the HCGS is reasonable and is under the 2200 °F SL.

RELAP5 model used by the staff for HCGS was based on an existing Browns Ferry RELAP5 model. Both Hope Creak and Browns Ferry reactors are based on GE BWR4 technology. Staff

TABLE 2.9.2

Parameters and Assumptions Used in Radiological Consequence Calculations for a CRDA

Parameter	Value
Peaking factor	1.75
Fraction of core inventory in gap	
Noble gases	0.1
Iodine	0.1
Alkali metals	0.12
Amount of core with damaged fuel rods, percent	1.8
Damaged rods that fail, percent	0.77
Melted fuel release fraction to vessel	
Noble gases	1.0
Iodine	0.5
Alkali metals	0.25
Fraction of activity released to vessel that enters main condenser	
Noble gases	1.0
Iodine	0.1
Others	0.01
Fraction of activity released from main condenser	
Noble gases	1.0
Iodine	0.1
Others	0.01
Release rate from main condenser, percent/day	1
Release duration, hours	24
CREFS initiation	Not credited
Control room normal flow rate, cfm	3300
Control room χ/Q	Table 2.9.7

TABLE 2.9.3

Parameters and Assumptions Used in Radiological Consequence Calculations for an ILPBA

Parameter	**Value**
Power level, MWt	4,031
Maximum reactor coolant iodine activity concentration, μCi/gm	4.0
Mass of total coolant released from break, lb	25,000
Reactor building volume, ft^3	4.00E+06
Iodine chemical form, %	
Elemental	97
Organic	3
Type of release to the atmosphere	Ground level release from FRVS vent
CREFS initiation	Not credited
Control room normal flow rate, cfm	3300
Control room X/Qs	Table 2.9.7

TABLE 2.9.4

Parameters and Assumptions Used in Radiological Consequence Calculations for an MSLB

Parameter	**Value**
Liquid coolant release discharged mass, lb	140,000
MSIV closure time, sec	5.5
Reactor coolant activity, µCi/gm DE I-131	
Normal	0.2
Spike	4.0
Radioactivity release rate to environment	Instantaneous
Control room occupancy factor	1
CREFS initiation	Not credited
Control room normal flow rate, cfm	3300
Control room X/Qs	Table 2.9.7

dose rates in these areas. Although there are increases in dose rates, these expected increases continue to be within the shielding design margins for the condensate, FW, and other affected systems.

The concentration of noble gases and other volatile fission products in the MSL will not change. The increased production rate of these materials in the reactor core is offset by the corresponding increase in steam flow; therefore, the concentration of these materials in the steam line remains constant. Although the EPU will result in an increase in the rate these materials are introduced into the Main Condenser and Off Gas systems, these expected increases continue to be within the design margins of the Off Gas system.

For the short lived activities, the most significant is N^{16}, the decreased transit (and decay) time in the MSL and the increased mass flow of the steam results in a larger increase in these activities in the major turbine building components. Based on the change in travel time of the steam to travel from the RPV nozzle to the steam components, the licensee estimates that the post-EPU N^{16} source strength for a 15 percent reduction in transit time is expected to increase radiation levels due to N^{16} concentration at steam turbine components by approximately 16 percent for operation at 3952 MWt, which bounds operation at 3840 MWt.

Radiation Protection Design Features

Occupational and onsite radiation exposures.

The staff has reviewed the licensee's plan for EPU with respect to its effect on the facility radiation levels and on the radiation sources in the core and coolant. The radiation sources in the core include radiation from the fission process, accumulated fission products, and neutron reactions as a result of neutron activation. The radiation sources in the core are expected to increase in proportion to the increase in power. This increase, however, is bounded by the existing safety margins of the design basis sources. Since the reactor vessel is inaccessible to plant personnel during operation and due to the design of the shielding and containment surrounding the reactor vessel, an approximate increase of 15 percent in the radiation sources in the reactor core will have no effect on occupational worker personnel doses during power operation.

In addition, the radiation shielding provided in the steam-affected areas of the plant is conservatively sized such that the increased source terms discussed above are not expected to significantly increase the dose rates in the normally occupied areas of the plant. Radiation dose rates in steam-affected areas of the plant are estimated to increase by 16 percent. These areas (including the reactor and turbine steam tunnels, moisture separator rooms, turbine rooms, high and LP heater rooms, condenser rooms, moisture separator drain pump and tank rooms, steam jet air ejector rooms, and hydrogen recombiner rooms) are all currently designated as high radiation areas and personnel access to them is restricted and controlled accordingly. The existing radiation zoning design (i.e., the maximum designed dose rates for each area of the plant), for areas outside the steam-affected areas, will not change as a result of the increased dose rates associated with this EPU.

During EPU testing, plant area radiation and process monitors are used to monitor radiation levels at 90 percent and 100 percent of CLTP and at 2.5 percent reactor power intervals above CLTP. In addition, as part of the ascension test plan, normally accessible areas adjacent to steam affected areas in the Turbine and RBs and Radwaste area of the Auxiliary Building will be

LCO 3.2.4 - LHGR, Applicability; LCO 3.2.4 - LHGR, Action; and SR 4.2.4.a

The licensee proposed to revise the LHGR RTP thermal monitoring threshold value to 24 percent.

The existing 25 percent of RTP limit for the LCO Applicability is based on generic analyses, evaluated up to approximately 50 percent of original RTP for the plant design with highest average bundle power (the BWR6) for all of the BWR product lines. This average bundle power (at 100 percent RTP) was 4.8 MWt. For the Hope Creek EPU, the average bundle power is 5.03 MWt. Therefore, the LCO Applicability for EPU conditions is reduced to 24 percent RTP. The proposed changes to the Action and SR maintain consistency with the change to the LCO Applicability. The staff finds the proposed change acceptable.

Table 3.3.1-1 - Reactor Protection System Instrumentation Table Notations, Note (j), and TS Table 3.3.4.2-1- EOC -RPT Trip instrumentation, Note (b)

The licensee has revised the reactor thermal power (RTP) value to 24 percent. The AL for the EPU is maintained at the same absolute power as the current setpoint. The licensee has reduced this value further in the conservative direction. Based on this, the 24 percent RTP value is acceptable to the staff. The staff has reviewed this analysis and finds the proposed change acceptable.

Table 4.3.1.1-1, Reactor Protection System Instrumentation Surveillance Requirements, Note (d)

The licensee has revised note (d) for the TS Table 4.3.1.1-1 to change thermal power ≥25 percent of rated thermal power to ≥24% of rated thermal power. The 25 percent RTP value is based on a generic analysis for all BWR plants with the highest average bundle power for 100 power original power level. However, the proposed EPU average bundle power for HCGS is higher than that previously assumed in the analysis. The new 24 percent RTP was established based on the new average bundle power. The analysis with the new bundle power is reviewed and accepted by the staff and is documented in Section 2.8, "Reactor Systems," of this safety evaluation. Based on this, the staff finds the proposed change acceptable.

Table 3.3.2-2 - Isolation Actuation Instrumentation Setpoints, Trip Function 3.d

The licensee has revised the trip setpoint and AV for main steam line flow instrumentation from 108.7 psid and 111.7 psid to 162.8 psid and 169.3 psid respectively. The AL in percent of rated steam flow is unchanged. The licensee has calculated the instrument setpoint and AV with an acceptable methodology as discussed in Section 2.4, "Instrumentation and Controls;" therefore, the staff finds the proposed change acceptable.

LCO 3.3.4.2 - End-of-Cycle Recirculation Trip System Instrumentation, Applicability

The licensee has revised the applicability of this LCO to thermal power greater than or equal to 24 percent of RTP. The proposed value is more conservative than the current value in terms of absolute power.

Based on this, the staff finds the proposed change acceptable.

Table 3.3.6-2 - Control Rod Block Instrumentation Setpoints, Trip Function 2.a, and Table 3.3.6-2 -Control Rod Block Instrumentation Setpoints, Trip Function 2.d

The licensee has revised the trip setpoint and AV for Flow Biased Neutron Flux-Upscale, (Functional Unit 2.a) and Neutron Flux - Upscale, Startup (Functional Unit 2.d). Based on the staff's review, the staff finds the proposed change acceptable.

LCO 3.3.11 - Oscillation Power Range Monitor Instrumentation, Applicability; and LCO 3.3.11, Action c

The licensee has revised the RTP from 25 percent to 24 percent. The 25 percent RTP value is based on a generic analysis for all BWR plants with the highest average bundle power for 100 power original power level. However, the proposed EPU average bundle power for HCGS is higher than that previously assumed in the analysis. The new 24 percent RTP was established based on the new average bundle power. The analysis with the new bundle power is reviewed and accepted by the staff and is documented in Section 2.8, "Reactor Systems," of this safety evaluation. Based on this, the staff finds the proposed change acceptable.

SR 4.3.11.5

The licensee has revised the thermal power to 26.1 percent from 30 percent RTP. The licensee has justified this change based on the fact that this new value maintains the same absolute power/flow region boundaries for the OPRM trip-enabled region. The analysis is reviewed and accepted by the staff and is documented in Section 2.8, "Reactor Systems," of this safety evaluation. Based on this, the staff finds the proposed change acceptable.

LCO 3.4.1.1 - Recirculation Loops, Action a.l.b; and SR 4.4.1.1.1.a

The licensee proposed to change the maximum power for SLO to 60.86 percent. The proposed changes maintain the existing licensed region for SLO. The analysis is reviewed and accepted by the staff and is documented in Section 2.8, "Reactor Systems," of this safety evaluation. Based on this, the staff finds the proposed change acceptable.

LCO 3.4.1.2 - Jet Pumps, SRs 4.4.1.2.a and 4.4.1.2.c

The licensee proposed to change 25 percent RTP to 24 percent RTP. The proposed changes are consistent with changes to the applicability of power distribution limits for ECCS performance analyses. The analysis is reviewed and accepted by the staff and is documented in Section 2.8, "Reactor Systems," of this safety evaluation. Based on this, the staff finds the proposed change acceptable.

LCO 3.6.1.2.c - Primary Containment Leakage

The licensee proposed to change 48.1 psig to 50.6 psig. The proposed change reflects the updated containment pressure response. Short-term and long-term containment analyses results are reported in the UFSAR. The short-term analysis is directed primarily at determining

If you have any questions, please call me at 301-415-3100.

Sincerely,

/RA/

John G. Lamb, Senior Project Manager
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Docket No. 50-354

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If you have any questions, please call me at 301-415-3100.

Sincerely,

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