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LR-N08-0205

SEP 24 2008

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: Clarifications to Final Safety Evaluation for Extended Power Uprate, License Amendment No. 174

On May 14, 2008, the NRC issued Amendment No. 174 to the operating license for the Hope Creek Generating Station (HCGS), increasing the authorized maximum power level by approximately 15 percent, from 3,339 megawatts thermal (MWt) to 3,840 MWt. In reviewing the related safety evaluation, PSEG Nuclear (LLC) identified some items requiring clarification.

Attachment 1 to this letter provides clarifications to the safety evaluation. A markup of the affected pages of the safety evaluation is provided in Attachment 2. The clarifications do not invalidate the conclusions documented in the safety evaluation.

There are no regulatory commitments in this letter or attachments.

If you have any questions or require additional information, please contact Mr. Paul Duke at 856-339-1466.

Sincerely,

Christine T. Neely
Director - Regulatory Affairs
PSEG Nuclear LLC

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NRR

Attachments (2)

1. Clarifications to Final Safety Evaluation for Extended Power Uprate, License Amendment No. 174
2. Marked-up Pages

cc: S. Collins, Regional Administrator – NRC Region I
J. Lamb, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
P. Mulligan, Manager IV, NJBNE

# ATTACHMENT 1

## Hope Creek Generating Station
## NRC Docket No. 50-354

### Clarifications to Final Safety Evaluation for Extended Power Uprate, License Amendment No. 174

| Item | Location* | Existing Text | Recommended Text | Basis |
|---|---|---|---|---|
| 1. | Section 2.1.2, Pressure-Temperature Limits and Upper-Shelf Energy<br><br>Page 10 | 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 60 ft-lbs... | 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 <u>66</u> ft-lbs... | Reference: PUSAR Table 3-2 |
| 2. | Section 2.1.2, Pressure-Temperature Limits and Upper-Shelf Energy<br><br>Page 10 | 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 0.15 percent. | 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 <u>0.14</u> percent. | Reference: PUSAR Table 3-2; RAI 1.3, LR-N07-0035, 03/13/07 |
| 3. | Section 2.1.4, Reactor Coolant Pressure Boundary Materials<br><br>Page 18 | These are 18 Category C welds (9 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 two Category E welds (weld overlay repaired welds). | These are <u>17</u> Category C welds (<u>8</u> 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 <u>three</u> Category E welds (weld overlay repaired welds). | Additional weld overlay performed during most recent refueling outage.<br><br>Reference: LR-N08-0003, 01/16/08 |

* Page numbers correspond to the Final Safety Evaluation for Amendment No. 174, Accession No. ML081230640

| Item | Location* | Existing Text | Recommended Text | Basis |
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| 4. | Section 2.1.4, Reactor Coolant Pressure Boundary Materials<br><br>Page 18 | Thus, the two weld overlay repaired welds are considered adequate for Hope Creek EPU operation. | Thus, the three weld overlay repaired welds are considered adequate for Hope Creek EPU operation. | Additional weld overlay performed during most recent refueling outage.<br><br>Reference: LR-N08-0003, 01/16/08 |
| 5. | Section 2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems<br><br>Page 52 | Secondary Condensate Pump 85 percent permissive | Secondary Condensate Pump 75 percent permissive | Secondary condensate pump runback permissive will be at approximately the same FW flow in Mlb/hr as the pre-EPU setpoint.<br><br>Reference: LR-N08-0003, 01/16/08 |
| 6. | Section 2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems<br><br>Page 53 | RBM Instrumentation, | [delete] | There are no requested TS changes for rod block monitor instrumentation. |
| 7. | Section 2.5.3.1 Fuel Pool Cooling and Cleanup System<br><br>Page 64 | This limiting heat load is currently $16.1\times10^6$ British thermal units per hour (BTU/hr) at 10 days after reactor shutdown. The licensee determined that the new limiting heat load for CPPU operation is $17.2\times10^6$ BTU/hr at 10 days after shutdown. | This limiting heat load is currently $16.1\times10^6$ 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 $17.2\times10^6$ BTU/hr at 8 days after shutdown. | The assumed decay time for a batch core offload is 8 days.<br><br>Reference: HCGS UFSAR Section 9.1.3.6; RAI response 7.1, LR-N07-0056, 03/22/07 |

| Item | Location* | Existing Text | Recommended Text | Basis |
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| 8. | Section 2.8.5.6.1, Inadvertent Opening of a Pressure Relief Valve<br><br>Page 134 | Automatic recirculation flow control system increases the recirculation flow to the maximum to compensate the power reduction.<br><br>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. | [delete] | HCGS recirculation flow control system does not include the master auto mode.<br><br>Reference: HCGS UFSAR Figure 7.7-6; UFSAR section 15.1.4.3.3 |
| 9. | Section 2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents<br><br>Page 140 | 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 calculated PCT for Two-Loop Operation. | 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 <u>licensing acceptance criteria</u>. | SLO PCT is not required to be less than the TLO PCT.<br><br>Reference: NEDO-33172, section 5.3.3, Accession No. ML053250466 |
| 10. | Table 2.9.2 Parameters and Assumptions Used in Radiological Consequence Calculations for a CRDA<br><br>Page 185 | Control room unfiltered intake, cfm 350 | Control room <u>normal flow rate, 3300 cfm</u> | Operation of the Control Room Emergency Filtration System is not credited for CRDA<br><br>Reference: Attachment 7 to LR-N06-0418, 10/20/06 |
| 11. | Table 2.9.3 Parameters and Assumptions Used in Radiological Consequence Calculations for an ILPBA<br><br>Page 186 | Control room unfiltered intake rate, cfm 350 | Control room <u>normal flow rate, 3300 cfm</u> | Operation of the Control Room Emergency Filtration System is not credited for ILPBA<br><br>Reference: Attachment 4 to LR-N06-0418, 10/20/06 |

| Item | Location* | Existing Text | Recommended Text | Basis |
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| 12. | Table 2.9.4 Parameters and Assumptions Used in Radiological Consequence Calculations for an MSLB<br><br>Page 187 | Control room unfiltered intake rate, cfm 350 | Control room normal flow rate, 3300 cfm | Operation of the Control Room Emergency Filtration System is not credited for MSLB<br><br>Reference: Attachment 3 to LR-N06-0418, 10/20/06 |
| 13. | Section 2.10.1, Occupational and Public Radiation Doses<br><br>Page 193 | 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 N16 source strength for a 15 percent increase in steam flow is expected to increase radiation levels due to N16 concentration at steam turbine components by approximately 16 percent for operation at 3840 MWt. | 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 N16 source strength for a 15 percent reduction in transit time is expected to increase radiation levels due to N16 concentration at steam turbine components by approximately 16 percent for operation at 3952 MWt, which bounds operation at 3840 MWt. | 16 percent increase is calculated for operation at 3952 MWt.<br><br>Reference: RAI 11.1, LR-N07-0069, 03/30/07; RAI 11.11, LR-N07-0154, 06/22/07 |
| 14. | Section 3.2 Technical Specifications<br><br>LCO 3.3.4.2 - End-of-Cycle Recirculation Trip System Instrumentation, Applicability<br><br>Page 243 | 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. | [delete] | Reduction to 24 percent RTP is consistent the power level at which the Technical Specification Thermal Limits must be monitored; however, Applicability of LCO 3.3.4.2 is not related to average bundle power. |

# ATTACHMENT 2

## Hope Creek Generating Station
## NRC Docket No. 50-354

## Clarifications to Final Safety Evaluation for Extended Power Uprate, License Amendment No. 174

## Marked-up Pages

<u>Page</u>

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

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.

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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 60 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 0.15 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

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[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

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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 18 Category C welds (9 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 two 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 two weld overlay repaired welds are considered adequate for Hope Creek EPU operation. three

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

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|---|---|
| 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 75 Condensate Pump 85 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 |

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| 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, RBM Instrumentation, 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 $16.1 \times 10^6$ British thermal units per hour (BTU/hr) at ~~10~~ 8 days after reactor shutdown. The licensee determined that the new limiting heat load for CPPU operation is $17.2 \times 10^6$ BTU/hr at ~~10~~ 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 $6.0 \times 10^6$ BTU/hr to $9.5 \times 10^6$ BTU/hr for a combined heat transfer capability of $19 \times 10^6$ 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. ~~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 cannot meet the additional load demand, the pressure regulator set is automatically reduced to a lower limit, and the reactor pressure decreases eventually.~~

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 calculated PCT for Two-Loop Operation. 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 unfiltered intake, cfm~~ | ~~350~~ |
| Control room χ/Q | Table 2.9.7 |

Control room normal flow rate 3300 cfm

**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 unfiltered intake rate, cfm~~ | ~~350~~ |
| Control room X/Qs | Table 2.9.7 |

Control room normal flow rate 3300 cfm

**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 unfiltered intake rate, cfm~~ | ~~350~~ |
| Control room X/Qs | Table 2.9.7 |
| Control room normal flow rate | 3300 cfm |

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.

reduction in transit time

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 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~~. 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. ~~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 3.3.6-2 - 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