



10 CFR 50.90

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March 22, 2007

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: Response to Request for Additional Information
Request for License Amendment - Extended Power Uprate

Reference: 1) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC,
September 18, 2006
2) Letter from USNRC to William Levis, PSEG Nuclear LLC, February 23,
2007

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

In Reference 2, the NRC requested additional information concerning PSEG's request. Attachment 1 to this letter restates the NRC questions and provides PSEG's response to each question.

PSEG has determined that the information contained in this letter and attachment does not alter the conclusions reached in the 10CFR50.92 no significant hazards analysis previously submitted.

There are no regulatory commitments contained within this letter

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Should you have any questions regarding this submittal, please contact Mr. Paul Duke at 856-339-1466.

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

Executed on 3/22/07
(date)

Sincerely,



George P. Barnes
Site Vice President
Hope Creek Generating Station

Attachments (2)

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

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

Extended Power Uprate

Response to Request for Additional Information

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

In Reference 2, the NRC requested additional information concerning PSEG's request. Each NRC question is restated below followed by PSEG's response.

5) Piping & NDE Branch (CPNB)

5.1 Identify the materials of construction for the reactor coolant pressure boundary (RCPB) piping/safe-ends. Discuss and explain the effect of the requested power uprate on the RCPB piping/safe-end materials.

Response

The materials of construction for the reactor vessel safe ends and reactor coolant pressure boundary piping are listed below:

Location	Material
Recirculation Outlet Nozzle (N1) Safe End	SA182 F304L
Recirculation Inlet Nozzle (N2) Safe End	SA182 F316L
Steam Outlet Nozzle (N3) Safe End	SA541 C11 Mod*
Feed Water Nozzle (N4) Safe End	SA541 C11 Mod* (Type 308L**)
Core Spray Nozzle (N5) Safe End	SB166 Ni-Cr-Fe
Core Spray Nozzle (N5) Safe End Extension	SA508 C11
Head Spray Cooling Nozzle (N6a) Safe End	SA508 C12
Head Spray Cooling Spare Nozzle (N6b) Safe End	SA541 C11 Mod*
Vent Nozzle (N7) Safe End	SA541 C11 Mod*
Jet Pump Instrumentation Nozzle (N8) Safe End	SA182 F304L
Core Differential Pressure & Liquid Control Nozzle (N10) Safe End	SA182 F304L
LPCI Nozzle (N17) Safe End	SA541 C11 Mod*
Recirculation Piping	SA312, SA376 and SA358 Type 304
Main Steam Piping	SA106 Gr. B and SA-155 Gr. KCF70
Feed Water Piping	SA333 Gr. 6
Core Spray Piping	SA333 Gr. 6
Vent Line Piping	SA106 Gr. B and SA312 TP304L
Jet Pump Instrumentation Piping	SA312 TP304L

Location	Material
Core Differential Pressure & Liquid Control Piping	SA312 TP304L
LPCI/RHR Piping	SA333 Gr. 6
Drain Line Piping	SA106 Gr. B
Instrumentation Piping	SA106 Gr. B and SA312 TP304L

*ASME B&PV Code Case 1332-5

**308L stainless steel cladding applied to thermal sleeve sealing surface

Implementation of extended power uprate (EPU) conditions at Hope Creek will result in an increase in neutron fluence, flow rate and operating temperature. The primary material effects are increases in material fatigue usage, the potential for Irradiation Assisted Stress Corrosion Cracking (IASCC), the potential for Flow-Accelerated Corrosion (FAC), and the potential for flow induced vibration. These material effects are addressed as follows.

- EPU effects on material fatigue usage for safe ends and piping are discussed in section 3.2 and 3.5 of NEDC-33076P.
- The increase in fluence and the associated effect on IASCC potential are not a concern for the RCPB safe ends and piping, as they are not in a location that receives sufficient fluence to be susceptible to IASCC.
- The increases in flow rate and temperature that result from EPU, and their effect on the FAC monitoring program are discussed in Section 10.7 of NEDC-33076P.
- The increases in flow rate that result from EPU and their effects on flow-induced vibration are discussed in Section 3.4 of NEDC-33076P and in Attachment 8 of the request for license amendment (Reference 1).

- 5.2 Identify the RCPB piping/safe-end components that are susceptible to intergranular stress corrosion cracking (IGSCC). Discuss any augmented inspection programs that have been implemented and the adequacy of the augmented inspection programs in light of the EPU.

Response

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." Hope Creek was designed, fabricated, and constructed per the guidance in NUREG 0313, revision 1 so most welds are IGSCC Category A welds, that is, resistant to IGSCC due to their metallurgical properties. The below table identifies the non-Category A welds. Non-Category A welds are considered IGSCC susceptible.

Weld Description	BWRVIP-75-A IGSCC CATEGORY
Core Spray At 120 Deg Safe-End To Safe End Extension	C
Core Spray At 240 Deg Safe-End To Safe End Extension	C
RPV Core Spray Inlet At 120 Deg Nozzle To Safe End	C
RPV Core Spray Inlet At 240 Deg Nozzle To Safe End (Overlay)	E
RPV CRD Nozzle To Cap	C
RPV Jet Pump Instrumentation Nozzle To Safe-End	C
RPV Jet Pump Instrumentation Nozzle To Safe-End	C
RPV Recirc Inlet At 120 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 150 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 210 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 240 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 270 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 30 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 300 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 330 Deg Nozzle To Safe End (Overlay)	E
RPV Recirc Inlet At 60 Deg Nozzle To Safe End	C
RPV Recirc Inlet At 90 Deg Nozzle To Safe End	C
RPV Recirc Outlet At 0 Deg Nozzle To Safe End	C
RPV Recirc Outlet At 180 Deg Nozzle To Safe End	C
RPV Head Spray Nozzle To Flange	C
Recirc Loop A Weldolet to Decon Line Piping	B
Recirc Loop B Weldolet to Decon Line Piping	B

Three conditions are needed for IGSCC to occur: a susceptible material, an oxidizing environment, and tensile stress.

- Operation at EPU conditions will result in slight changes to temperature, pressure and flow for some systems comprising the RCPB but these changes will have negligible effect on the tensile stresses.
- EPU does not affect the material.
- Operation at higher power will result in a slightly higher oxygen generation rate due to radiolysis of water; however, as discussed in the response to question 5.4 below, steps will be taken to ensure that the

RCPB will continue to be mitigated from an oxidizing environment perspective.

Since the three conditions will be essentially unchanged for EPU, the IGSCC program has a negligible impact from EPU. Thus the IGSCC augmented inspection program is unchanged for EPU.

- 5.3 Identify all flawed components including overlay repaired welds that have been accepted for continued service by analytical evaluation based on American Society of Mechanical Engineers (ASME), Section XI rules. Discuss the adequacy of such analysis considering the effect of the EPU on the flaws.

Response

Hope Creek plant has two welded overlays on ASME Section XI flawed components. There are no ASME Section XI flawed components that have been accepted for continued service by analytical evaluation. The two overlays are on the reactor vessel core spray nozzle to safe end weld (N5B) and on the reactor vessel recirculation inlet nozzle to safe end weld (N2K).

The core spray overlay was verified adequate for EPU operation. At the core spray nozzle location, there is a slight (0.2%) change in temperature, but no change in pressure or flow due to EPU. Hence, the change in temperature has an insignificant effect on P + Q stresses and the fatigue usage for EPU.

The recirculation inlet overlay was verified adequate for EPU operation. At the recirculation inlet nozzle location, there is a slight increase in pressure (1.1%), a slight decrease in temperature (-0.2%), and an increase in recirculation flow (3.4%). The pressure and temperature operating conditions used in the overlay analysis bound the EPU temperature and pressure conditions. For the flow increase, the change in the heat transfer coefficient used in the analysis is 2.7%, which is considered insignificant.

The two overlays on the Hope Creek plant reactor vessel nozzles are adequate for EPU conditions.

- 5.4 Identify the mitigation processes being applied at Hope Creek to reduce the RCPB component's susceptibility to IGSCC, and discuss the effect of the requested EPU on the effectiveness of these mitigation processes. For example, if hydrogen water chemistry (HWC) was applied at the plant, it would be necessary to perform the electrochemical potential measurements at the most limiting locations to ensure that the applied hydrogen injection rate is adequate to maintain the effectiveness of HWC since oxygen content in the coolant is expected to increase due to increased radiolysis of water resulting from extended power uprate.

Response

Several mitigation processes have been applied to Hope Creek to reduce the RCPB component's susceptibility to IGSCC. Hope Creek was designed, fabricated, and constructed per the guidance in NUREG 0313, revision 1. As a result most welds were either constructed with corrosion resistant materials, solution treated, or clad with resistant materials. There are twenty areas that are exceptions. These areas contain nickel-based alloys and are not considered IGSCC resistant. One of these areas developed a leak early in plant life and a weld overlay was applied. The nineteen other areas had mechanical stress improvement process (MSIP) applied. A non-through wall indication was found in one area where MSIP was applied and an overlay was applied. Following construction, a modification was made on the two recirculation decontamination connections. The welding associated with the modification sensitized the piping, thus causing the heat-affected zone of the piping to be IGSCC susceptible. MSIP was applied at these two areas. MSIP and original construction processes used for IGSCC resistance are not affected by EPU.

Water chemistry controls are also used for IGSCC mitigation. Hope Creek is currently a Category 3a plant in accordance with BWRVIP-130, EPRI BWR Water Chemistry Guidelines, 2004 Revision. A mitigation monitoring system including Fe and Pt ECP electrodes, and 24 durability coupons (catalyst loading) was installed in January 2006. A classic NobleChem application was performed during shutdown for R13 in April 2006. All Secondary Parameters recommended by EPRI are installed and operating properly. A hydrogen benchmark test was conducted following startup from R13 in May 2006. All Secondary Parameters were also benchmarked to provide correlation with Measured ECP. The EPRI Radiolysis / ECP Model, Version 2, was updated following R13 and is also being used to provide molar ratio data at the most limiting location in the vessel, currently defined as the upper downcomer. For EPU implementation, the most significant activity with respect to mitigation will be performance of a second hydrogen benchmark test to determine the appropriate injection level. Industry experience suggests that hydrogen flow will increase by approximately 10%. Catalyst loading will continue to be monitored as required by removal and analysis of durability coupons. Following R14 (EPU), the Radiolysis / ECP Model will be updated and cases run to validate molar ratio data. These actions will ensure that EPU will not affect the water chemistry controls used for IGSCC mitigation.

6) Accident Dose Branch (AADB)

- 6.1 "Calculation No. H-1-AB-MDC-1854, Revision 11R0, Main Steam Line Break (MSLB) Accident," sheet 11, section 4.13 states that credit is not taken for the engineered safety features of the control room emergency filtration (CREF) system that mitigate airborne activity within the control room. Is the CREF designed to initiate for MSLB? If so, how are the assumptions bounding?

Response

The CREF is designed to automatically initiate when the post-accident airborne activity concentration at the CR air intake exceeds the CR ventilation radiation monitor setpoint value of $2 \times 10^{-5} \mu\text{Ci/cc}$.

The post-MSLB accident releases for the pre-accident iodine spike case and the equilibrium iodine case are postulated to instantaneously release radioactivity to the environment in a single puff. With an instantaneous puff release there is no time for the CREF intake charcoal filtration to initiate operation. Consequently, the activity present at the CR air intakes enters the CR envelope without benefit of filtration. However, the CREF system is designed with filtered recirculation flow at a flow rate of $4000 \text{ cfm} \pm 10\%$ and charcoal filter efficiencies greater than 95%. The filtered recirculation flow will be initiated in response to the high radioactivity sensed by the CR ventilation radiation monitors. The post-MSLB CR dose will be substantially reduced if credit were taken for this CREF recirculation charcoal filtration. Therefore, the calculated CR dose without crediting the CREF system initiation is bounding.

6.2 Question Deleted.

7) Balance-of-Plant Branch (SBPB)

7.1 The Hope Creek Updated Final Safety Analysis Report (UFSAR) Section 9.1.3.1 states:

"The Spent Fuel Pool Closed Cooling (FPCC) system is designed to handle the decay heat released by all anticipated combinations of spent fuel that could be stored in the fuel pool. The pool water temperature is maintained at a maximum of 135°F under the design load of $16.1 \times 10^6 \text{ Btu/h}$. This heat load is the discharge of a reload quantity of spent fuel (approximately one third of the core) at the end of a fuel cycle, plus the decay heat of the reload spent fuel from all previous refuelings."

- a) Please explain how the plant licensing basis will continue to be satisfied in this regard following the Constant Pressure Power Uprate (CPPU).

Response

The design and licensing bases of the fuel pool cooling and cleanup (FPCC) system is to maintain pool temperature at a maximum of 135°F with a batch offload (approximately one third of the core) and at a maximum of 150°F with a full-core offload. These current licensing basis values do not change at CPPU. However, the $16.1 \times 10^6 \text{ Btu/hr}$ batch heat load at 8-days after plant shutdown at CLTP (with a full SFP) changes to $17.2 \times 10^6 \text{ Btu/hr}$ at CPPU. Similarly, the full core off-load at

10-days after shutdown increases from 34.2×10^6 Btu/hr at CLTP to 43.0×10^6 Btu/hr at CPPU. HCGS calculations demonstrate that the FPCC system (with RHR assist in the case of the abnormal full-core offload) maintains bulk pool temperatures below 135°F under the above conditions.

The heat removal capabilities to maintain SFP temperature limits are based on the most limiting design conditions for these systems. This includes the maximum service water and Safety Auxiliaries Cooling System (SACS) design temperatures, minimum flow rates, maximum fouling and allowable tube plugging for each in-service component (i.e., pumps and heat exchangers).

- b) Table 9.1-1 of the Hope Creek original Final Safety Analysis Report (FSAR) listed the heat transfer capability of the fuel pool heat exchangers as 6.0×10^6 Btu/h, and the current revision of Table 9.1-1 lists the heat transfer capability as 9.515×10^6 Btu/h. There is an apparent discrepancy in that the heat transfer capability that is listed for the fuel pool heat exchangers compared to the licensing basis fuel pool heat load of 16.1×10^6 Btu/h. Please explain.

Response

The spent fuel pool plate heat exchangers were modified in 1990 from 72 plates to 99 plates. The parameters tabulated in Table 9.1-1 are per single heat exchanger. The heat transfer capability for each spent fuel pool heat exchanger at the design conditions listed is 9.515×10^6 Btu/hr. The total heat transfer for both trains in service at these design conditions is 19.03×10^6 Btu/hr.

It should be noted that the above heat exchanger capacity is based on maximum design SACS temperature of 95°F , which would involve refueling in mid-summer. Since refueling outages are typically conducted in the fall and spring, SACS temperature would be well below the design limit and heat exchanger capacity during actual outages would be greatly increased.

- 7.2 The Hope Creek UFSAR Section 9.1.3.1 states:

"The Fuel Pool Cooling and Cleanup (FPCC) System is designed to permit the Residual Heat Removal (RHR) System to be operated in parallel with the FPCC system through a crosstie, to remove the maximum heat load and to maintain the bulk water temperature in the spent fuel pool [SFP] at or below 150°F , with a maximum anticipated heat load of 34.2×10^6 Btu/h. This heat load is the discharge of one full core of fuel at the end of a fuel cycle, plus the decay heat of the reload spent fuel from all previous refuelings. If required, one RHR pump

and one RHR heat exchanger can be aligned to augment the FPCC system through the system crosstie. For this system configuration, a heat load greater than 45 million Btu/hr can be removed from the spent fuel pool with a maximum SACS [Safety Auxiliaries Cooling System] inlet temperature to the RHR heat exchanger of 95 °F and a spent fuel pool temperature of 152 °F."

- a) Please explain the differences in RHR system function and alignment and other design parameters (if applicable) in the above paragraph where one alignment can remove 34.2×10^6 Btu/h and the other alignment can remove 45×10^6 Btu/h.

Response

The SFP is normally cooled by a combination of two (2) FPCC pumps and two (2) FPCC heat exchangers depending on the decay heat load in the pool. Each FPCC heat exchanger is rated at 9.515 MBtu/hr under design conditions of 95°F SACS cooling water and 135°F SFP water (normal batch refuelings). The design heat removal capacity of the FPCC system is $2 \times 9.515 = 19.03$ MBtu/hr.

When a full core offload is planned, additional SFP cooling is available through a crosstie to the RHR system that aligns one RHR pump to take suction from the SFP skimmer surge tanks, pump SFP inventory through one RHR heat exchanger and return to the SFP via the SFP diffusers. This alignment is called the RHR FPC Assist (RFA) mode. In this configuration, fuel in the RPV is cooled by the other RHR loop aligned in the normal shutdown cooling (SDC) mode while fuel is transferred to the SFP. [Note that fuel pool cooling at Hope Creek is particularly robust in that cooling can be supplied by two FPCC pumps and two FPCC heat exchangers, along with two RHR pumps and two RHR heat exchangers. All of this equipment is safety-related, and seismic category I.]

The RHR heat exchanger RFA mode design heat removal capacity is 41.6 MBtu/hr at a pool temperature of 150°F and 30.2 MBtu/hr at a pool temperature of 135°F. The combined SFP heat removal available in the RFA mode, therefore, is 49.2 MBtu/hr ($19.03 + 30.2$) at a pool temperature of 135°F.

An additional RHR alignment is available to cool both the SFP and the RPV during refuelings when the fuel transfer canal is flooded, the RPV head is off and level is flooded up such that the SFP and RPV water inventory is common. This configuration results in the comparable heat load of a full core offload to the SFP since the entire heat load from the fuel in the RPV and the SFP is added to the common water inventory. One RHR pump and associated RHR heat exchanger is aligned to take suction from the RCS recirculation line and return to the SFP. Both FPCC

pumps and heat exchangers are in service. A recirculation pump may be in service at minimum speed or natural circulation is used to circulate flow through the RPV core. This is referred to as the Alternate RHR FPC Assist (ARFA) mode.

The maximum CPPU design heat load for a full core offload to the SFP occurs in the ARFA alignment due to pump heat from both the recirculation and RHR pumps in service. These pumps add 4.38 MBtu/hr to the total heat load that must be removed. The background decay heat in the SFP, based on the pool partially filled with 3242 bundles after 14 consecutive refuelings prior to the full core offload, is 5.32 MBtu/hr.

The decay heat from the full core 10 days after reactor shutdown is 38.2 MBtu/hr. Therefore, the total CPPU heat load required to be removed is 47.9 MBtu/hr (4.38 + 5.32 + 38.2). As discussed above, the heat removal available with FPCC and one RHR loop is greater than this heat load following CPPU.

- b) Please explain how the plant licensing basis will continue to be satisfied in this regard following the CPPU.

Response

The licensing basis maximum SFP temperatures (i.e., 135°F batch and 150°F full core) are not changed by CPPU. As discussed, heat removal capacity is greater than the decay heat load and SFP temperature following a full core offload can be maintained within design limits after CPPU.

- 7.3 The Hope Creek UFSAR Section 9.1.3.6 states:

"Acceptance Criterion II.I.d.(4) of Standard Review Plan (SRP) 9.1.3 limits the water temperature in the fuel pool to 140 °F at the maximum heat load with the normal cooling system operating in a single active failure condition.

The bulk water temperature in the fuel pool could reach 152 °F if one FPCC pump was not available, or 174 °F if one FPCC pump and one FPCC heat exchanger were not available with a maximum normal heat load of 16.1×10^6 BTU/hr. The radiological consequences of the fuel pool temperature reaching 152 °F and 174 °F have been evaluated. The resultant doses will not exceed 10 CFR 20 limits at the site boundary. However, the RHR System can be manually aligned to provide supplemental cooling."

- a) Apparently none of the calculations for CPPU fuel pool cooling as summarized in Table 6-3 of the Power Uprate Safety Analysis Report (PUSAR) considered a single failure in the FPCC system. Please explain

how the plant licensing basis will continue to be satisfied as described above including meeting the specified maximum temperatures with a single failure in FPCC without crediting RHR.

Response

As stated in Section 6.3 of the PUSAR, SFP batch temperature analyses are performed with the single-failure of the RHR assist mode of the FPCC system.

RHR-assist was selected as the single-failure because it is the limiting scenario for heat removal. Since both the RHR assist mode and the Hope Creek (HCGS) FPCC are safety-related, seismic I systems, to postulate failure of an FPCC component along with unavailability of the RHR assist mode effectively postulates two failures of safety-related components. Nonetheless, these various failure scenarios are addressed in section 9.1.3.6 of the UFSAR under a discussion of the 140°F temperature limit in SRP 9.1.3, following a single active failure.

Hope Creek continues to meet the temperatures specified in UFSAR section 9.1.3.6 under CPPU conditions. SFP bulk temperature following CPPU refueling remains below 152 °F with no RHR assist and with one FPCC pump unavailable. SFP bulk temperature following CPPU remains below 174 °F with no RHR assist and with one FPCC pump and one FPCC heat exchanger unavailable. The SFP cooling thermal hydraulic performance calculation for CPPU is based upon a batch refueling heat rate of 17.2 MBtu/hr at 8 days after plant shutdown (rather than the pre-CPPU heat rate of 16.1 MBtu/hr at 8 days). The reason that the prior analyses remain bounding with this additional heat is that the HCGS plate exchangers were expanded from 72 plates to 99 plates following the initial analyses that calculated the 152 °F and 174 °F values. The additional plates increase heat transfer areas of these heat exchangers by approximately 33%. This is more than sufficient to accommodate CPPU heat rates within the previously identified pool bulk temperatures.

Using CPPU heat rates, the calculation evaluates a number of FPCC system failure scenarios using a thermal-hydraulic computer (PROTOFLO) model. This calculation determines that with no RHR assist and the single failure of one FPCC pump, the SFP decay heat rate can be as high as 21.8 MBtu/hr before a bulk pool temperature of 152°F would be exceeded. The 21.8 MBtu/hr value is an approximate 35% increase over the original 16.1 value, which correlates to the approximate 33% increase in HX heat transfer area when plates were added. Since the calculated CPPU heat rate is 17.2 MBtu/hr, the 152 °F temperature is not exceeded at CPPU.

The calculation for CPPU also evaluates the case of SFP cooling with only one FPCC pump and one FPCC heat exchanger (with no RHR assist). In this scenario, the SFP heat rate can be as high as 21.9 MBtu/hr before a bulk temperature of 174 °F would be exceeded. Since the calculated CPPU heat rate is 17.2 MBtu/hr, the 174 °F temperature is not exceeded at CPPU. Consequently, Hope Creek continues to comply with the UFSAR statement of paragraph 9.1.3.6.

- b) The above UFSAR section indicates that the maximum normal fuel pool heat load with postulated single failures is 16.1×10^6 BTU/hr. The normal fuel pool heat load corresponds to the batch core offload (approximately one third of the core).
- b.1) Confirm that the Hope Creek normal fuel pool heat load is still valid and that the full core offload continues to be an unusual situation such that a single failure for the full core offload does not have to be assumed. Provide the frequency of performing full core offloads and explain to what extent this is limited by plant procedures.

Response

As discussed above, prior to CPPU, the peak SFP heat rate for a batch (approximately one third of the core) refueling was 16.1 MBtu/hr at 8-days following plant shutdown. For CPPU the peak pool heat rate at 8-days following plant shutdown is 17.6 MBtu/hr (including full-pool background heat and FPCC pump heat). As indicated in the response to 7.3(a), the 16.1 MBtu/hr value is not being used to evaluate CPPU conditions. It will be replaced in the UFSAR update that will follow CPPU implementation. The comparable value under CPPU conditions is 17.2 MBtu/hr.

The full core off-load at HCGS remains an abnormal event, as further discussed in b.2 below.

- b.2) Provide the frequency of performing full core offloads and explain to what extent this is limited to assure compliance with the plant licensing basis in this regard.

Response

The full core off-load remains an abnormal event at HCGS. Since a full core offload requires movement of 764 elements to and from the SFP, it is undesirable from a refueling outage perspective and is only performed when demanded by maintenance or related activities. In the past five fuel cycles (approximately 8 years) there have been two full core offloads:

RF-09 Full Core Off-Load (due to replacement of all SRM/IRM dry tubes)
 RF-10 Fuel Shuffle
 RF-11 Fuel Shuffle
 RF-12 Full Core Off-Load (due to extent of control rod work)
 RF-13 Fuel Shuffle

- 7.4 The Hope Creek UFSAR Section 9.1.3.6 states that the fuel pool loads are calculated based on SRP Section 9.1.3 and Branch Technical Position ASB 9-2 except, a) for Hope Creek "annual refueling" means 18 month refueling, and b) the decay time is assumed to be 8 days for calculating the normal heat load, and 10 days for calculating the maximum heat load.

Configurations 1, 2, and 3 of Table 6-3 of the PUSAR show the time to initiate fuel transfer to SFP as 59 hours, 24 hours, and 74 hours, respectively.

- a) Please explain the large difference between the decay times described in UFSAR section 9.1.3.6 and the fuel transfer times listed in Configurations 1, 2, and 3 of Table 6-3 of the PUSAR.

Response

The decay heat loads used in the determination of SFP temperatures shown in UFSAR Section 9.1.3.6 are based on the methodology presented in Branch Technical Position (BTP) ASB 9-2 at 8 days after reactor shutdown for a batch refueling and 10 days after shutdown for a full-core offload. CPPU decay heat analyses are based on the ANSI/ANS 5.1-1979 (+ 2 sigma), also at 8 days after shutdown for a batch and 10 days for a full-core. The UFSAR will be revised to replace BTP ASB 9-2 with ANSI/ANS 5.1-1979.

The configurations in PUSAR Table 6-3 (Hope Creek Spent Fuel Pool Parameters for CPPU) were provided to indicate minimum in-vessel decay times needed to achieve various standard review plan (SRP) limits, such as 135°F for batch offload, 140°F for full core offload with RHR assist, and 150°F for full core offload with the alternate RHR FPC assist mode. These various configurations and times are further described below.

The "time to initiate fuel transfer to the SFP" listed in Table 6-3, for Configuration 1 (59 hours after reactor shutdown) is based on the decay heat load from the "fresh" batch offload at 6 bundles per hour added to the decay heat load from 13 previous batch offloads such that the maximum SFP temperature after completion of the batch offload is 134.9°F with only FPCC in service (i.e., 2 FPCC pumps + 2 FPCC heat exchangers).

The "time to initiate fuel transfer to the SFP" listed in Table 6-3, for Configuration 2 (24 hours after reactor shutdown) is based on the decay heat load from the "fresh" full core offload at 6 bundles per hour added to the decay heat load from 14 previous batch offloads such that the maximum SFP temperature after completion of the full core offload is 139.5°F with FPCC and RFA in service (i.e., 2 FPCC pumps and 2 FPCC heat exchangers; and 1 RHR pump and 1RHR heat exchanger) cooling the SFP and RHR SDC in service during fuel movement.

The "time to initiate fuel transfer to the SFP" listed in Table 6-3, for Configuration 3 (74 hours after reactor shutdown) is based on the decay heat load from the "fresh" full core offload at 6 bundles per hour added to the decay heat load from 14 previous batch offloads such that the maximum SFP temperature after completion of the full core offload is 149.9°F with FPCC and ARFA in service (i.e., 2 FPCC pumps and 2 FPCC heat exchangers; and 1 RHR pump and 1RHR heat exchanger) with fuel in the RPV cooled by natural circulation during fuel movement.

- b) Explain how the plant will continue to meet the plant licensing basis as reflected in UFSAR Section 9.1.3.6 above for CPPU.

Response

PUSAR Table 6-3 documents the results of the analysis of FPCC, RFA and ARFA heat removal capacity with the revised heat loads resulting from CPPU. No changes are required to these systems and existing design margin is available to accommodate increased heat loads resulting from CPPU. The licensing basis SFP temperature limits are satisfied, as summarized in Table 6-3 and in plant calculations as discussed in the responses to questions 7.1, 7.2, and 7.3.

- 7.5 The Hope Creek UFSAR Section 6.4.1.1.2 of the PUSAR states:

"The SACS LOCA [loss of coolant accident] heat load calculation conservatively assumes that Spent Fuel Pool (SFP) cooling is not shed; however, an over conservatism was removed from this assumption. The CLTP [Current Licensed Thermal Power] LOCA calculation assumed the maximum SFP heat load immediately following a full fuel offload. The CPPU calculation credits the delay between offload and returning to power operation. This change results in a lower CPPU SFP heat load as well as no net increase in the total SACS LOCA heat load assumed between CLTP and CPPU."

- a) What amount of delay time is credited between offload and returning to power operation?

- b) What controls have been established to assure that the plant is not returned to service following a refueling outage until after the assumed delay time has passed?
- c) Confirm that the assumed delay time will be reflected in the UFSAR for CPPU operation.

Response

Prior to CPPU, the SACS post-LOCA heat load calculation included SFP heat immediately following completion of a normal fuel off-load. This value was considered to be unrealistically high because a design-basis LOCA can not occur during a refueling outage with the RCS depressurized and the RPV head removed. Hence in preparing the SACS evaluations for CPPU, the SFP heat rate at 12 days after plant shutdown was selected as a conservative assessment of the heat rate in the SFP during a design basis LOCA, including one following a recent refueling.

The total impact of this delay assumption is shown in the table below:

Time After Shutdown	Post-EPU SFP Heat Load	Condition
98 hours	19.0 MBtu/hr	Peak heat load immediately following fuel off-load to the SFP
12 days	14.1 MBtu/hr	Assumed SFP heat load during a DBA LOCA that occurs following the subsequent start-up after refueling (with 14 batches of offloads in the pool)

In view of the above table, the actual time delay that is assumed above is approximately 8 days (the difference between 12 days and 98 hours). However, no controls are planned to enforce this delay because the assumption is not technically significant as further described below.

The SACS LOCA calculation (EG-0020; STACS Required Flows and Heat Loads) does include a contribution from the SFP during postulated loss-of-power (LOP) and LOCA events. However, this does not directly impact the calculation of UHS temperatures.

The calculation that determines the capabilities of the ultimate heat sink to meet SACS post-accident heat removal requirements is calculation EG-0047 (HCGS Ultimate Heat Sink Temperature Limitations). EG-0047, Revision 4, was provided to the USNRC in mid-2006 as part of LCR H05-12 (Amendment 168). Assumption 3.2.7 from calculation EG-0047 (shown below) states that SFP heat

exchangers are isolated post-accident if SACS temperatures can not be maintained below limits.

For this analysis, the SFP heat exchangers are isolated if the SACS header temperature cannot be maintained below 95°F (normal conditions) or 100°F (LOCA and/or LOP). Following a LOP signal the fuel pool pumps trip and are not automatically loaded onto the EDGs; fuel pool heat exchangers would remain isolated if river temperatures were high. Following a LOCA scenario, the instrument air system is assumed to be lost (since the RACS and TACS systems that cool the air compressors would automatically be isolated). The Loss of Instrument Air (LIA) would cause the fuel pool heat exchanger outlet valves to fail closed preventing fuel pool cooling pump flow, and fuel pool heat exchangers would remain isolated if river temperatures were high. If a LOP or LIA did not occur, and SACS temperature reaches the design value (95°F or 100°F), operator action in accordance with HC.OP-AB.COOL would isolate the SFP heat exchangers.

This is the current licensing basis for the SACS system and was found to be acceptable by the NRC staff in a previous SER (Amendment 120 dated 4/19/1999). In Amendment 120, the NRC accepted SFP isolation under postulated LOCA/LOP conditions when SACS heat exchanger outlet temperatures exceed 95°F, and provided its rationale for acceptance in paragraph 3.1.1.2 of the SER. No change to this condition is proposed for CPPU.

It should be noted that refueling outages are held in the fall and spring, when ultimate heat sink (UHS) temperatures are typically well below design basis values. Hence, should a LOCA occur immediately following refueling (when SFP heat loads are greatest), SACS capabilities would be greatly enhanced by UHS temperature and also the core would lack the power-history assumed for a DBA LOCA.

In view of the above, controls need not be established to enforce the assumed delay time and the time delay need not be reflected in the UFSAR for CPPU operation.

7.6 Question Deleted.

7.7 Hope Creek EPU License Amendment Request, Attachment 10, Matrix 5, under flood protection states that the Hope Creek flooding analysis determined that CPPU may result in flood level increases of up to 36 percent in certain areas but that the equipment in the affected areas has been previously analyzed for wetting and submergence.

Section 8.1 of the PUSAR states "Hope Creek has sufficient capacity to handle added liquid increases required, i.e., it can collect and process the drain fluids. The drainage systems backflow at maximum flood levels and infiltration of radioactive water into non-radioactive water drains do not change as a result of CPPU

- a) Provide a listing of the areas that have changes in the flood level, what equipment is affected in those areas, and why the effect does not impact plant safety.

Response

As stated in the PUSAR (Section 10.1.2, Liquid Line Breaks, Internal Flooding) internal flooding for postulated RWCU line breaks increases due to the increased mass release at EPU conditions. The evaluation shows that for the rooms affected by postulated RWCU line breaks, the CPPU mass release will result in a maximum increase in flooding levels of 36%.

The 36% additional mass release in an RWCU line break is more the result of a change in analysis methodology than a direct affect of EPU. Specifically, the additional mass release is the result of using enthalpy values at 66% rated power/40% rated flow rather than the previous 100% power/100% flow enthalpy. The mass release rate through a pipe break at a fixed pressure is greater when the water/steam release has a lower temperature, which when condensed to water represents the worst case relative to room flood levels. The 36% mass release increase was directly converted into flood level in the RWCU Rooms. This flooding level has been evaluated and determined to have no affect on safe shutdown equipment.

The RWCU rooms affected by the flood and the resolution of those increases are shown in the following table below:

Room Number / Description / Line Designation	CLTP Flooding Level (3339 MWt)	EPU Flooding Level (3950 MWt)	Level Increase	Resolutions
4403 / EI 132 RWCU Pump Rm / BG-003-4"	3.88 feet	5.28 feet	1.4 feet	Notes 1, 2, and 3
4405 / EI 132 RWCU Pump Rm / BG-004-4"	3.48 feet	4.73 feet	1.25 feet	Notes 1, 2, and 3
4502 / EI 145 RWCU F/D Holding Pump Rm / BG-066-4"	4.37 feet	5.94 feet	1.57 feet	Notes 1, 2, and 3
4503 / EI 145 RWCU F/D Holding Pump Rm / BG-065-4"	4.97 feet	6.76 feet	1.79 feet	Notes 1, 2, and 3

Room Number / Description / Line Designation	CLTP Flooding Level (3339 MWt)	EPU Flooding Level (3950 MWt)	Level Increase	Resolutions
4506 / EI 145 RWCU HX Rm / BG-008-4"	3.25 feet	4.42 feet	1.17 feet	Notes 1, 2, and 3
4601 / EI 162 MCC Area / BG-068-2"	Not determined for HELB	1.1 inches	1.1 inches	Note 4

Note 1 Acceptable; equipment that is credited to achieve safe shutdown is above postulated flooding levels, protected/qualified for flooding, or not located in these rooms. The increased flooding levels do not affect plant safety.

Note 2: Acceptable; limiting loads in the civil calculations are not governed by flood level load

Note 3: CLTP levels multiplied by 1.36

Note 4: Areas from the break in Room 4601 includes connected Rooms 4602 through 4609, 4614, 4615, and 4625 through 4628. Rooms 4606 and 4625 through 4628 all have existing flood levels ≤ 0.1 ft. All affected rooms (including Room 4601) have been evaluated by HCGS and no safety related equipment is affected by the increased HELB flooding level.

- b) Do the maximum flood levels and the infiltration of radioactive water into non-radioactive water drains considered in section 8.1 of the PUSAR consider the flood level increases of up to 36% described in Attachment 10, Matrix 5? If not, what are the effects of the increase in flood level?

Response

As stated in the PUSAR (Section 8.1, Liquid and Solid Waste Management) the floor drainage system backflow at maximum flood levels and infiltration of radioactive water into non-radioactive water drains do not change as a result of EPU.

As discussed above, the 36% additional mass release in an RWCU line break is the result of a change in analysis methodology rather than a direct impact of EPU.

Flood level calculations conservatively neglect the head of water above the drain in calculating the flow leaving through the floor drains. Some of the calculations of flood level took no credit whatsoever for floor drains. The increased flood heights have no effect on the assumed floor drain flow rate used in the calculations. This is conservative in determining maximum flood levels.

In reality, the increased flood levels will result in higher floor drain flow rates than the conservatively calculated rates. The effect of high energy pipe breaks on the floor drain sump size and the sump pump capacity on the lowest elevation of the Reactor Building is not specifically analyzed. This 36% flood level increases are not considered adverse based on the relatively small changes in level.

- 7.8 PUSAR Section 6.4.1.1.2 states that diesel generator loads remain unchanged for a LOCA, and Section 6.1.1 states that the existing emergency power system is adequate. UFSAR Section 9.5.4 states "The standby diesel generator (SDG) fuel oil storage and transfer system provides onsite storage for at least 7 days of operation to all SDGs as they operate at their full operating loads as described in SDG loading calculation E-9(Q)."

Explain how the proposed power uprate will affect the SDG loading sequence and the duration of the SDG loads for postulated accident conditions, and describe the impact that this will have on the SDG fuel oil inventory that is required to support seven days of SDG operation. Explain how the required inventory is assured by the existing Technical Specification requirements, including consideration of usable fuel oil storage tank volume and measurement uncertainties.

Response

The proposed power uprate will not affect the SDG loading sequence or the duration of the SDG loads for postulated accident conditions. The proposed power uprate will not affect the SDG fuel oil inventory to support seven day of SDG operation. There are no electrical power increase requirements or load duration changes associated with the safety-related loads supplied by the Standby Diesel Generators (SDGs).

There are no changes to the fuel oil storage requirements as evaluated by the NRC in Technical Specification Amendment No. 59.

The diesel generator storage tank calculation considers the weight volume of fuel oil, nominal volume of tank, unusable volume, and fuel oil consumption. Fuel oil consumption for a diesel generator is conservatively assumed at a rate equivalent to 100% of the continuous electrical rating of 4430 kWe for seven days.

Each set of Diesel Generator Fuel Oil Storage tanks has storage capacity of 46,744 gallons of usable fuel at the minimum level setpoint. This exceeds the Technical Specification (T/S) requirement of 44,800 gallons. The stored usable fuel for operation of four diesel generators is 186,976 gallons. The required 7-day fuel for operation of three diesel generators continuously at full load is 172,053 gallons. This results in 8.7% design margin. As noted in the NRC

Safety Evaluation for HCGS TS Amendment No. 59, a failure of one EDG is assumed, and credit is taken for the ability to transfer fuel oil from the inoperable EDG's storage tanks.

Calculation SC-JE-0059 determines the loop accuracy of the start/stop pump setpoints and the high/low alarms for the diesel fuel oil day tanks. It established that the transfer system and alarms operate with sufficient margin above T/S limits to account for measured uncertainties.

Calculation SC-JE-0051 determines the loop accuracy of the low and low-low level alarms for the diesel fuel oil storage tanks. It established that alarm setpoints have enough margin above T/S limits to account for measurement uncertainties.

8) SG Tube Integrity & Chem. Eng Br (CSGB)

- 8.1 Section 3.11 of the PUSAR states that there are slight changes in Reactor Water Cleanup (RWCU) system operating conditions due to a decrease in inlet temperature and increase in operating pressure. Please provide the magnitude of these changes.

Response

The changes in RWCU system operating conditions are shown in the table below.

RWCU System	CLTP 3339 MWt	CPPU 3840 MWt
RWCU Inlet Temperature, °F	531.6	530.8
RWCU Inlet Pressure (RPV dome pressure, neglecting head), psig	1005	1005
RWCU Outlet Temperature, °F	434.8	433.9
RWCU Outlet Pressure (at the feedwater line), psig	1049	1063
Design RWCU Flow, lbm/hr	133,000	133,000
Maximum RWCU Flow, lbm/hr	148,000	148,000

- 8.2 Section 3.11 of the PUSAR concludes that at power uprate conditions the RWCU system will perform adequately at the present flow rate. Please discuss the aspects of the system that were evaluated and the parameters evaluated to reach this conclusion (for example, the effects of changes in temperature,

pressure, chemistry, and flow rate on heat exchanger heat transfer and materials).

Response

As can be seen from the table above, the changes in RWCU system operating conditions are very small. The system flow rate is unchanged. The performance effects of these small changes are a slight increase in the calculated reactor water conductivity (from 0.068 $\mu\text{mho/cm}$ to 0.071 $\mu\text{mho/cm}$) as discussed in PUSAR Section 3.11.

- 8.3 According to PUSAR Section 3.11, the concentration of iron in the reactor water is expected to increase from 16 ppb to 19 ppb, but that this is within the design chemistry limits and does not affect performance of the RWCU system. Please discuss the remaining margin between the expected iron level and the design limit.

Response

There is not a specific design limit for reactor water iron. It is a chemistry parameter primarily controlled by input from feedwater and buildup within the reactor vessel. Full flow condensate pre-filters were installed and went into operation in June 1999. The pre-filters reduced feedwater iron to below 0.1 ppb and since that time iron oxide (Fe_2O_3) is added to the feedwater via the active zinc injection system to maintain the recommended concentration range of 0.1 to 1.5 ppb delineated in BWRVIP-130: BWR Water Chemistry Guidelines – 2004 Revision. Historical average reactor water total iron values are provided in the table below:

DATE RANGE	TOTAL IRON, ppb
June 1998 – June 1999	174
June 1999 – June 2000	92
June 2000 – June 2001	23
June 2001 – June 2002	32
June 2002 – June 2003	56
June 2003 – June 2004	16
June 2004 – June 2005	16
June 2005 – June 2006	11
June 2006 – Present	10

The other significant event impacting reactor water iron levels was the implementation of NobleChem in April 2006. The injection was performed during shutdown for R13. Upon restart in May 2006 and return to hydrogen injection, the reactor crud began a restructuring process from Fe_2O_3 to Fe_3O_4 . During this transition, reactor water soluble iron levels increased as expected. Pre-NobleChem, post-NobleChem, and present average iron values are provided in the table below:

PERIOD	INSOLUBLE, ppb	SOLUBLE, ppb	TOTAL, ppb
Pre-NobleChem Cycle 13	5.3	2.0	7.3
Post-NobleChem 05/06 – 10/06	5.7	10.4	16.1
Present 11/06 – 02/07	4.7	4.5	9.2

Since feedwater iron is controlled by chemical addition and is the primary contributor to reactor water iron, no significant changes are anticipated.

- 8.4 PUSAR Sections 3.11 and 4.1.3 state that some containment isolation valves have reduced operating margins but remain capable of performing their isolation function. Please discuss how the operating margin is reduced by the proposed power uprate and by how much.

Response

Operating margin is reduced slightly due to the small increase in containment pressure. A review of the motor operated valve (MOV) calculation database determined that 34 containment isolation valves were impacted by the increased drywell pressure (from 48.1 psig to 50.6 psig). Nine additional MOVs were identified as being impacted based on an assumed increase in torus pressure to 39.9 psig. The impact on valve margin was minimal due to the pressure increases which resulted in a small increase in required thrust.

The affected motor operated valves (MOVs) and the output capability margin, after accounting for the increased pressure, are shown in the table below:

Valve ID	Description	Margin(%)
1BD-HV-F031	RCIC TORUS SUCTION VALVE	64.0
1BJ-HV-F042	HPCI SUPPRESSION POOL SUCTION ISOLATION VALVE	49.6
1FC-HV-F059	RCIC TURBINE STEAM EXHAUST ISOLATION VALVE	111.0
1FC-HV-F060	RCIC VACUUM PUMP DISCHARGE VALVE	56.0
1FC-HV-F062	RCIC VACUUM BREAKER OUTBOARD ISOLATION VALVE	151.2
1FC-HV-F084	RCIC VACUUM BREAKER INBOARD ISOLATION VALVE	465.7
1FD-HV-F071	HPCI TURBINE EXHAUST ISOLATION VALVE	352.8
1FD-HV-F075	HPCI VACUUM BREAKER OUTBOARD ISOLATION VALVE	348.6
1FD-HV-F079	HPCI VACUUM BREAKER INBOARD ISOLATION VALVE	361.4
1GS-HV-4955A	CAC "A" H2/O2 ANALYZER UPPER DRYWELL SUCTION INBD ISO VLV	289.9
1GS-HV-4955B	CAC "B" H2/O2 ANALYZER UPPER DRYWELL SUCTION INBD ISO VLV	297.3
1GS-HV-4974	CONTAINMENT NITROGEN MAKEUP SUPPLY ISOLATION VALVE	21.4

Valve ID	Description	Margin(%)
1GS-HV-4983A	CAC "A" H2/O2 ANALYZER UPPER DRYWELL SUCTION OUTBD ISO VLV	308.4
1GS-HV-4983B	CAC "B" H2/O2 ANALYZER UPPER DRYWELL SUCTION OUTBD ISO VLV	300.0
1GS-HV-4984A	CAC "A" H2/O2 ANALYZER LOWER DRYWELL SUCTION OUTBD ISO VLV	306.6
1GS-HV-4984B	CAC "B" H2/O2 ANALYZER LOWER DRYWELL SUCTION OUTBD ISO VLV	300.0
1GS-HV-5019A	CAC "A" H2/O2 ANALYZER LOWER DRYWELL SUCTION INBD ISO VLV	286.3
1GS-HV-5019B	CAC "B" H2/O2 ANALYZER LOWER DRYWELL SUCTION INBD ISO VLV	297.3
1GS-HV-5050A	PRIMARY CONT H2 RECOMBINER AS205 SUCTION HEADER INBD ISO VLV	64.1
1GS-HV-5050B	PRIMARY CONT H2 RECOMBINER BS205 SUCTION HEADER INBD ISO VLV	40.5
1GS-HV-5052A	PRIMARY CNT H2 RECOMBINER AS205 SUCTION HEADER OUTBD ISO VLV	76.5
1GS-HV-5052B	PRIMARY CNT H2 RECOMBINER BS205 SUCTION HEADER OUTBD ISO VLV	11.4
1GS-HV-5053A	PRIMARY CNT H2 RECOMBINER AS205 RETURN HEADER OUTBD ISO VLV	54.3
1GS-HV-5053B	PRIMARY CNT H2 RECOMBINER BS205 SUCTION RETURN OUTBD ISO VLV	52.0
1GS-HV-5054A	PRIMARY CNT H2 RECOMBINER AS205 RETURN HEADER INBD ISO VLV	75.4
1GS-HV-5054B	PRIMARY CNT H2 RECOMBINER BS205 RETURN HEADER INBD ISO VLV	79.4
1GS-HV-5078A	CONTAINMENT HYDROGEN RECOMBINER AS205 GAS RECIRCULATION VLV	276.6
1GS-HV-5078B	CONTAINMENT HYDROGEN RECOMBINER BS205 GAS RECIRCULATION VLV	275.7
1GS-HV-5080A	CONTAINMENT HYDROGEN RECOMBINER AS205 SUCTION HDR INLET VLV	263.1
1GS-HV-5080B	CONTAINMENT HYDROGEN RECOMBINER BS205 SUCTION HDR INLET VLV	275.7
1HB-HV-F003	DRYWELL FLOOR SUMP PUMP INBOARD ISOLATION VALVE	146.1
1HB-HV-F004	DRYWELL FLOOR SUMP PUMP OUTBOARD ISOLATION VALVE	279.6
1HB-HV-F019	DRYWELL EQUIP DRAIN SUMP PUMP INBOARD ISOLATION VALVE	314.4
1HB-HV-F020	DRYWELL EQUIP DRAIN SUMP PUMP OUTBOARD ISOLATION VALVE	186.2
1KL-HV-5147	COMPRESSOR AK-202 SUCTION OUTBOARD ISOLATION VALVE	22.3
1KL-HV-5148	PCIG INBOARD CONTAINMENT ISOLATION VALVE	55.3
1KL-HV-5160A	PCIG POST ACCIDENT COMPRESSOR A SUCTION ISOLATION VALVE	22.3

Valve ID	Description	Margin(%)
1KL-HV-5160B	PCIG POST ACCIDENT COMPRESSOR B SUCTION ISOLATION VALVE	21.4
1KL-HV-5162	COMPRESSOR BK-202 SUCTION OUTBOARD ISOLATION VALVE	21.4
1SK-HV-4953	PRIMARY CONTAINMENT LEAKAGE DETECTION SUPPLY ISOLATION VALVE	30.6
1SK-HV-4957	PRIMARY CONTAINMENT LEAKAGE DETECTION INBD RETURN ISO VALVE	287.2
1SK-HV-4981	PRIMARY CONTAINMENT LEAKAGE DETECTION OUTBD RETURN ISO VALVE	302.7
1SK-HV-5018	PRIMARY CONTAINMENT LEAKAGE DETECTION INBD SUPPLY ISO VALVE	33.5

This margin provides the capability of the MOV to perform its safety related function during worst case accident and degraded voltage conditions. The margin also accounts for standard error factors such as rate of loading torque switch repeatability, which are taken into account in the MOV calculations per the requirements of GL89-10 and GL96-05. The lowest margin on the impacted MOVs is 11.4 percent.

- 8.5 According to Section 3.11 of the PUSAR, the proposed power uprate would cause an increase in the filter/demineralizer backwash frequency. Please discuss the amount of the increase relative to the capacity for processing liquid and solid radwaste.

Response

Historically, reactor water cleanup filter demineralizer backwash frequency has been approximately 13 to 14 weeks (90 to 100 days). The frequency is typically longer in the winter (colder) months and shorter in the summer (warmer) months. This is controlled by reactor vessel introduction rates from the condensate demineralizers that perform better with lower temperatures. Following NobleChem at the start of refueling outage RF13 (Spring 2006), the backwash frequency was reduced due to restructuring of the oxide film resulting in high levels of soluble iron in the reactor water. Backwash frequency initially dropped to 2 to 3 weeks following startup from RF13 and is continuing to recover. Backwash frequency as of February 2007 is averaging approximately 9 to 10 weeks and is expected to continue to improve to the pre-NobleChem frequency. During the immediate period following NobleChem with very short backwash frequency, radwaste was not challenged due to the increase relative to the capacity for processing liquid and solid radwaste. The increase due to EPU will be considerably less than the impact due to NobleChem and will present no challenge to processing radwaste, either solid or liquid. Since the increase in feedwater flow will be approximately 10%, it is anticipated that a similar reduction in backwash frequency will occur resulting in a backwash frequency of 11 to 13 weeks (80 to 90 days). The impact of EPU will be further reduced by increasing

the size of the condensate demineralizer anion underlays to 100 ft³ and revising the mixed bed above the underlay to a stoichiometric mixture, to be performed during normal resin bed replacement.

- 8.6 According to NRC Regulatory Guide 1.183, the analysis release duration for a LOCA is 30 days, and a pH greater than 7 will prevent iodine re-evolution. The suppression pool pH analysis provided to the staff in 2001, which was part of a request to use an alternate source term, was performed for a power level of 3458 MWth. Please discuss whether the pH analysis bounds conditions at the proposed EPU power level of 3840 MWth. If the previously analysis does not bound the proposed EPU conditions, please provide an updated evaluation showing the suppression pool pH will be greater than 7 for the 30-day LOCA period.

Response

The suppression pool pH analysis was revised to support the increase in rated thermal power. The conclusion is unchanged from the analysis provided to the staff in 2001. The pH of the suppression pool for the Hope Creek plant radiological design basis accident (DBA) LOCA is greater than 8 over a period of 30 days following accident initiation.

The analysis was performed for a reactor power of 4031 MWt which bounds the proposed EPU conditions. The STARpH computer code was used for both the original suppression pool pH analysis and the revised analysis. The completed HCGS post-accident suppression pool pH calculation for EPU is proprietary to Polestar Applied Technology, Inc. and is not marked in accordance with 10 CFR 2.390(b)(1)(i)(B). The calculation is available for audit in PSEG's offices.

References

1. PSEG letter LR-N06-0286, Request for License Amendment: Extended Power Uprate, September 18, 2006
2. NRC letter, Hope Creek Generating Station - Request for Additional Information Regarding Request for Extended Power Uprate (TAC NO. MD3002), February 23, 2007