NRR-DMPSPEm Resource

From: Regner, Lisa
Sent: Thursday, December 14, 2017 4:42 PM
To: Thomas, Brian J.; Paul.Duke@pseg.com
Cc: Regner, Lisa
Subject: Final Reactor Systems Branch (SRXB) Request for Additional Informaiton - HOPE CREEK MUR (CAC MF9930)
Attachments: Final SRXB HOPE CREEK MUR (CAC MF9930).docx

Final Request for Additional Information (CAC MF9930; EPID L-2017-LLS-0002)

On October 25, 2017, the U.S. Nuclear Regulatory Commission (NRC) staff sent PSEG (the licensee) a draft Request for Additional Information (RAI) with two questions SRXB-1 and SRXB-2, as provided in the attached document. The RAI questions relate to a license amendment request (LAR) that proposes to increase the rated thermal power level from 3840 megawatts thermal to 3902 megawatts thermal, and make technical specification changes as necessary to support operation at the uprated power level. This is referred to as a measurement uncertainty recapture (MUR) uprate.

On November 1, the NRC staff held a clarification call with PSEG; further discussions were needed. On November 13, PSEG provided an email with clarification information for SRXB-2 concerning an Appendix R fire event analysis at uprate conditions. The NRC staff informed PSEG that it would issue SRXB-2 for a formal response. On November 29, 2017, the NRC staff sent PSEG a draft RAI with SRXB questions 3 through 17. A clarification call was held between PSEG and NRC staff on December 4, 2017. Clarification information was provided by PSEG on December 5 and 6, 2017, which showed that existing information was on the docket for Questions 12, 13, and 17. Additionally, since questions SRXB-9 and 11 were related, they were combined into one question. On December 4, the staff provided a revised question SRXB-1 to PSEG that resulted from the clarification call on November 1. The SRXB-2 question was unmodified.

On December 11, the licensee stated that it understood all of the SRXB questions and would respond to SRXB questions 1 and 2 by December 29, 2017; further, it requested that the remainder of the SRXB questions be combined into one RAI. The final SRXB RAI is attached, with questions SRXB-1 through SRXB-13. If PSEG does not respond by this date, the requested completion date for the MUR decision may not be met by the NRC.

The NRC staff also informed the licensee that a publicly available version of this final RAI would be placed in the NRC's Agencywide Documents Access and Management System (ADAMS).

By letter dated July 7, 2017, (ADAMS package Accession No. ML17188A259), the licensee requested an amendment to the Operating License for Hope Creek Generating Station. The proposed amendment requests a MUR uprate for Hope Creek. The NRC staff requires additional information to complete its review of this request as detailed in the attached document.

Hearing Identifier: NRR_DMPS
Email Number: 54

Mail Envelope Properties (CY4PR0901MB2211A701AFA6349FD3651CCAE50A0)

Subject: Final Reactor Systems Branch (SRXB) Request for Additional Informaiton - HOPE CREEK MUR (CAC MF9930)
Sent Date: 12/14/2017 4:42:06 PM
Received Date: 12/14/2017 4:42:07 PM
From: Regner, Lisa

Created By: Lisa.Regner@nrc.gov

Recipients:
"Regner, Lisa" <Lisa.Regner@nrc.gov>
Tracking Status: None
"Thomas, Brian J." <Brian.Thomas@pseg.com>
Tracking Status: None
"Paul.Duke@pseg.com" <Paul.Duke@pseg.com>
Tracking Status: None

Post Office: CY4PR0901MB2211.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	2574	12/14/2017 4:42:07 PM
Final SRXB HOPE CREEK MUR (CAC MF9930).docx		45631

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

REQUEST FOR ADDITIONAL INFORMATION

BY REACTOR SYSTEMS BRANCH (SRXB)

HOPE CREEK GENERATING STATION

MEASUREMENT UNCERTAINTY UPRATE

DOCKET NO. 50-354

By letter dated July 7, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17188A260) (Reference 1), pursuant to Section 50.90 of Title 10 of the *Code of Federal Regulations (CFR),* PSEG Nuclear, LLC (PSEG or the licensee) submitted a License Amendment Request (LAR) for changes to the Hope Creek Generating Station (HCGS) Technical Specifications (TSs). The proposed LAR would revise Technical Specification as necessary to support an increase of approximately 1.6-percent in the rated thermal power from the currently licensed thermal power (or extended power uprate) of 3840 Megawatts thermal (MWt) to 3902 MWt. The proposed Measurement Uncertainty Uprate (MUR) represents an increase of approximately less than 20-percent above the Original Licensed Thermal Power (OLTP).

The Reactor Systems Branch (SRXB) reviewed Sections 3.1, 3.6, 3.7, 3.8, 3.9, 3.10, 4.1 (except containment coatings), 4.2, 4.3, 6.5, 9.1, 9.3, and 10.4 of the Thermal Power Optimization (TPO) Safety Analysis Report (TSAR) (Reference 2) and Enclosure 15 (Reference 3) to the licensee's letter dated July 7, 2017 and request the following additional information to complete its review.

Regulatory Basis for Questions SRXB-1 and 2

(a) NUREG-0800, Standard Review Plan 6.2.2, Containment Heat Removal Systems

(b) 10 CFR Part 50, Appendix A, GDC 38 states in part, the containment heat removal system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels. The emergency core cooling system (ECCS) and the containment heat removal system pumps are used for core cooling and containment heat removal during postulated accidents and special events.

SRXB-1

The NRC staff requests the following information for the Loss-of Coolant Accident (LOCA), Station Blackout (SBO) event, and Anticipated Transient Without Scram (ATWS) event containment analysis:

(a) For the Extended Power Uprate (EPU) analysis, or the most current analysis of record (AOR) if performed subsequent to EPU, performed at the 102-percent EPU thermal power, provide the limiting suppression pool temperature response transients for the LOCA (considering small steam line break or recirculation suction line break whichever is limiting for the suppression pool temperature), SBO event, and ATWS event. Confirm that the peak suppression pool temperature for the LOCA at 102-percent EPU thermal power is (1) 212.3°F as stated in Table 4-1 of the EPU Safety Analysis Report (SAR) (Reference 7), (2) 198.0°F for SBO event, and (3) 199.0°F for the limiting ATWS event

as stated in the table under Section 2.6.5 of the NRC EPU Safety Evaluation Report (SER) (Reference 3). If not, provide the new values and appropriate justification for the new values.

(b) Provide the AOR results for the residual heat removal (RHR) and core spray (CS) pump limiting NPSH available (NPSHa) at the pump inlet for the LOCA short-term (up to 600 seconds from its initiation in case the pumps operate with runout flows up to 600 seconds), LOCA long-term, SBO, and ATWS without crediting Containment Accident Pressure (CAP) developed during these events.

(c) Provide the vendor tested values (including test uncertainty) of the RHR and CS pumps NPSH required (NPSHr), and state its basis. In case the basis is different from the as-defined by Hydraulic Institute (HI), i.e., the NPSH corresponding to a decrease by 3-percent of the pump total dynamic head (denominated as $NPSHr_{3\%}$) at a given flow, provide justification.

(d) For the NPSHr value used to calculate the NPSH margin (NPSHa minus NPSHr), the field NPSHr (denominated as NPSHreff) is generally greater than its test-value (with test uncertainty included) obtained by testing at the pump vendor's facility due to several effects. These effects which create a field uncertainty are due to the following: (a) pump speed different in the field caused by motor slip, (b) configuration of the field-installed pump suction piping different from the configuration at the vendor's test facility, and (c) air content of the water used in the test may be lower than that of the pumped water in the field. Provide a discussion considering these effects in calculating the field NPSHr including test and field uncertainties (NPSHreff), for the LOCA short term (with pump runout flows), LOCA long term, SBO, and ATWS events AOR NPSH margins for the RHR and the CS pumps. Provide the values of NPSHreff and the NPSH margins for these events.

SRXB-2

Background

The suppression pool temperature response for the Appendix R Fire event at the TPO power (101.6-percent EPU power) is not bounded by the current (EPU) suppression pool temperature response analyzed at 100-percent EPU power, based on the following:

- Section 6.7.1 of the EPU SAR (Reference 7) states:

  The limiting Appendix R fire event was analyzed assuming CLTP and CPPU [Constant Pressure Power Uprate].

- The NRC SER for the EPU (Reference 3), table in Section 2.6.5 states the Appendix R Fire event evaluation thermal power is 3840 MWt which is 100-percent of the EPU power level.

- In the supplement dated February 16, 2017 (Reference 8), to the EPU SAR (Reference 7), there is no mention that the analysis reported is based on 102-percent EPU power. The supplement revises Section 6.7.1 and Table 6-4 of the EPU SAR, and increases the peak suppression pool temperature by 0.4°F due to delay in the initiation of its cooling.

- In response to EPU RAI # 13.14 dated March 30, 2017 (Reference 9), the Appendix R Fire event evaluation power level is stated as 3840 MWt (100-percent EPU) and the calculated peak suppression pool temperature is 205.9°F. The response date for this RAI is subsequent to the date of the EPU supplement (Reference 8).

- The TSAR (Reference 2) Section 6.7.1 mentions Fire event peak cladding temperature and containment pressure analysis was performed at 102-percent EPU power. It does not mentions about suppression pool temperature response and NPSH analysis.

Requests

The NRC staff requests the following information for the 10 CFR 50, Appendix R Fire event containment analysis:

(a) For the analysis performed at 100-percent EPU thermal power, provide the limiting suppression pool temperature response profile for Appendix R Fire event. Based on multiple spurious operations (MSOs), provide the list of the scenarios that are considered, justifying the analysis is for the most limiting Appendix R Fire scenario. Confirm that the limiting peak suppression pool temperature for this event at 100-percent EPU thermal power of 205.9°F as stated in Table 6-1 of the EPU SAR (Reference 7) remains applicable. If not, provide their new values and appropriate justification for the new values.

(b) For the TPO power uprate, provide a summary of the Appendix R Fire analysis, including key assumptions, inputs, and results (temperature versus time from event and peak temperature). Provide a list of the scenarios that are considered, justify the analysis for the most limiting Appendix R Fire scenario. Provide justification if the conservatism in any of the inputs and assumptions from the EPU analysis is reduced.

(c) For the TPO power uprate, provide a summary of the most limiting pump inlet NPSHa analysis including assumptions, inputs, and results, such as the graph of NPSHa versus time, and the value of minimum NPSHa at the inlet of the pump. Provide the value of the required NPSH (NPSHr) and its basis, and the pump NPSH margin without crediting CAP.

(d) If CAP is credited for achieving a zero or greater than zero NPSH margin, provide responses to the applicable requirements in Section 6.0 of Reference 5.

REFERENCES for SRXB-1 and 2

1. PSEG letter to NRC dated July 7, 2017, “License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate” (ADAMS Accession No. ML17188A260).

2. Enclosure 6 of Reference 1, “GE-Hitachi Nuclear Energy (GEH) Document NEDC-33871P, “Safety Analysis Report for Hope Creek Generating Station Thermal Power Optimization,” Revision 0, (Proprietary Version),” (ADAMS Accession No. ML17188A261), Enclosure 8 of Reference 1- GEH Report NEDO-33871 (Non-Proprietary Version) (ADAMS Accession No. ML17188A263

3. NRC letter to PSEG dated May 14, 2008, “Hope Creek Generating Station - Issuance of Amendment Re: Extended Power Uprate (TAC No. MD3002)” Amendment No. 174 (ADAMS Accession No. ML081230581), Safety Evaluation Report (ADAMS Accession No. ML073050337)

4. SECY 11-0014, “Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents,” (ADAMS Accession Number ML102590196)

5. Enclosure 1 to SECY-11-0014, “The Use of Containment Accident Pressure in Reactor Safety Analysis ADAMS ML102110167,” (ADAMS Accession Number ML102110167)

6. Letter from NRC to BWROG dated February 25, 2013, “Use of Containment Accident Pressure in Demonstrating Acceptable Operation of Emergency Core Cooling System and Containment Heat Removal Pumps during Postulated Accidents,” (ADAMS Accession No. ML13016A013)

7. General Electric NEDC-33076P, Revision 2, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate", August 2006, (ADAMS Accession Number ML062690073)

8. PSEG letter to NRC dated February 16, 2007, “Supplement to License Amendment Request for Extended Power Uprate,” (ADAMS Package No. ML070590178, Attachment 1- ADAMS Accession No. ML070590186)

9. PSEG letter to NRC dated March 30, 2007, “Response to Request for Additional Information Request for License Amendment - Extended Power Uprate,” (ADAMS Package No. ML071010231, Attachment 1- ADAMS Accession No. ML07101244)

SRXB-3

Regulatory Basis: 10 CFR 50 Appendix A, GDC 16, “Containment design”

The table in Section 4.1 of the TSAR (Reference 2) does not provide information regarding the containment temperature response for Equipment Environmental Qualification (EEQ). Provide the analysis results, including the limiting break size, under the TPO uprate condition.

SRXB-4

Regulatory Basis: 10 CFR 50 Appendix A, GDC 16, “Containment design”

The table in Section 4.1 of the TSAR does not provide any information regarding the peak containment wall temperature for structural analysis. Provide the analysis results, including the limiting break size, under the TPO uprate condition.

SRXB-5

Regulatory Basis: Compliance with Generic Letter (GL) 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

With respect to the TSAR, Section 4.1.2, provide justification for why the TPO uprate is determined to have no effect on the current evaluation of GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

SRXB-6

Regulatory Basis: Compliance with Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions"

Section 4.1.4 of the TSAR states:

> The containment design temperatures and pressures in the current GL 96-06 evaluation are not exceeded under post-accident conditions for the TPO uprate.

The current containment design pressure and temperature is 62 psig and 340°F respectively. Please clarify the containment design temperature and pressure at which the current GL 96-06 evaluation was performed.

SRXB-7

Regulatory Basis: NUREG-0800, Standard Review Plan 6.2.2, "Containment Heat Removal Systems", and 10 CFR Part 50, Appendix A, GDC 38, "Containment heat removal"

Referring to the TSAR tables in Section 3.10, and Section 9.1.1, confirm that the increase in the decay heat for the TPO uprate has been quantitatively verified to be within the residual heat removal (RHR) equipment heat removal capability in the 'shutdown cooling' and the 'alternate decay heat removal' modes of the RHR system.

SRXB-8

Regulatory Basis: Compliance with Generic Letter (GL) 89-16, "Installation of a Hardened Wetwell Vent"

The TSAR does not provide an evaluation of the hardened wetwell vent at the TPO uprate power installed per the NRC GL 89-16. Provide the results of a quantitative evaluation of the vent at the TPO uprate power level.

SRXB-9

Regulatory Basis: 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

(a) Refer to TSAR, Section 4.3, which refers to Table 4-1. Provide the revised LOCA analysis results for the GNF2 fuel given in Table 4-1 (i.e., the third column of Table 4-1.

(b) Section 2.1 of the TSAR states that the Cycle 21 core will have the first reload of GNF2 fuel and residual GE14 fuel. Reference 34 (NEDC-33172P, ADAMS Accession No. ML053250469) in the TSAR provides the SAFER/GESTR-LOCA Loss-of Coolant Accident (LOCA) analysis at the 102-percent extended power uprate (EPU) based on GE14 fuel. Provide a LOCA analysis report based on the mixed fuel to be used in Cycle 21 at the 102-percent EPU power to support the TPO uprate. In case the LOCA analysis for the mixed core was not performed, provide justification that the current LOCA analysis based on GE14 fuel is bounding.

SRXB-10

Regulatory Basis: 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

The footnote number 2 under TSAR Table 4-1 states:

> "Where applicable, includes the effects of any change to or error discovered in the acceptable evaluation models previously reported to the NRC.

(a) What are changes or errors discovered and what are its effects?

(b) The previous LOCA analysis reported to the NRC is in Reference 6, which was submitted on the docket. If errors were discovered in the Reference 6 LOCA analysis, submit a revised corrected version for NRC review.

SRXB-11

Regulatory Basis: 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants."

Refer to TSAR, Section 9.3.1, item 4, specify the basis for the acceptance criteria for the peak local suppression pool temperature less than 217.5°F. Refer to TSAR Table 9-1, "ATWS Acceptance Criteria Results", explain how the TPO value of 215.6°F for peak local suppression pool temperature was determined?

SRXB-12

Regulatory Basis: 10 CFR 50.63, "Loss of all alternating current power", Regulatory Guide 1.155

Refer to TSAR (Reference 2), Section 9.3.2; under the TPO uprate Station Blackout (SBO) conditions, provide the following:

(a) The capacity of the Class 1E batteries
(b) The SBO compressed nitrogen requirements
(c) The evaluation of a loss of ventilation on rooms that contain equipment essential for plant response to an SBO event

SRXB-13

Regulatory Basis: Missing information

ASME [American Society of Mechanical Engineers] Standard PTC 19.5-2004, "Flow Measurement," Section 10-9 provides installation consideration for the Ultrasonic Flow Meters (UFMs). Referring to the as-built drawing 1-P-AE-01(Q)-21 in Enclosure 15, "LEFM [Leading Edge Flow Meter] Flow Meter Installation Location Drawings" (Reference 3) of Reference 1, provide a discussion if the guidelines in sections 10-9.1, "Acoustic Path Length and Angle", 10-9.4, "Secondary Flow and Distorted Velocity Profiles", and 10-9.5, "Integration" of the above ASME standard were followed. In case the ASME guidelines were not followed, specify the standard and the requirements according to which the LEFM was installed and justify their equivalence to the ASME standard guidelines.

REFERENCES for SRXB-3 through 13ahs

1. PSEG letter to NRC dated July 7, 2017, “License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate” (ADAMS Accession No. ML17188A260).

2. Enclosure 6 of Reference 1, “GE-Hitachi Nuclear Energy (GEH) Document NEDC-33871P, “Safety Analysis Report for Hope Creek Generating Station Thermal Power Optimization,” Revision 0, (Proprietary Version),” (ADAMS Accession No. ML17188A261), Enclosure 8 of Reference 1- GEH Report NEDO-33871 (Non-Proprietary Version) (ADAMS Accession No. ML17188A263)

3. Enclosure 15 of Reference 1, “LEFM Flow Meter Installation Location Drawings”, (ADAMS Accession No. ML17188A270)

4. PSEG Letter 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), and Attachment 4, NEDC-33076P, Revision 2 "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate," August 2006 (ADAMS Accession No. ML062690073)

5. Cameron Engineering Report: ER-157(P-A) Rev. 8 and Rev. 8 Errata, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System,” dated May 2008, (ADAMS Accession No. ML102950246)

6. GE Nuclear Energy, “SAFER/GESTR-LOCA Loss of Coolant Accident Analysis for Hope Creek Generating Station at Power Uprate,” NEDC-33172P, March 2005, (ADAMS Accession No. ML053250469)