Attachment 3 and Enclosure 3 Contain Proprietary Information to be Withheld from Public Disclosure Pursuant to 10 CFR 2.390

PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
Nuclear LLC

DEC 22 2017

10 CFR 50.90

LR-N17-0189
LAR H17-03

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

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

Subject: **RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (CAC NO. MF9930)**

References

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

2. NRC e-mail to PSEG, " Final Reactor Systems Branch (SRXB) Request for Additional Information – HOPE CREEK MUR (CAC MF9930)," dated December 14, 2017 (ADAMS Accession No. ML17349A081)

In the Reference 1 letter, PSEG Nuclear LLC (PSEG) submitted a license amendment request for Hope Creek Generating Station (HCGS). The proposed amendment will increase the rated thermal power (RTP) level from 3840 megawatts thermal (MWt) to 3902 MWt, and make Technical Specification (TS) changes as necessary to support operation at the uprated power level.

In Reference 2, the U.S. Nuclear Regulatory Commission staff provided PSEG a Request for Additional Information (RAI) to support the NRC staff's detailed technical review of Reference 1.

This letter provides the requested information in Attachment 1 (Non-Proprietary) and Attachment 3 (Proprietary). Also, Enclosure 1 (Non-Proprietary) and Enclosure 3 (Proprietary) provide the revised LOCA analysis results for GNF2 fuel requested in RAI SRXB-9.

Attachment 3 and Enclosure 3 contain proprietary information as defined by 10 CFR 2.390. GE-Hitachi Nuclear Energy Americas LLC (GEH), as the owner of the proprietary information, has executed affidavits (provided in Attachment 2 and Enclosure 2) identifying that the proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. GEH requests that the proprietary information in Attachment 3 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390(a)(4).

PSEG has determined that the information provided in this submittal does not alter the conclusions reached in the 10 CFR 50.92 no significant hazards determination previously submitted. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

No new regulatory commitments are established by this submittal. If you have any questions or require additional information, please do not hesitate to contact Mr. Brian Thomas at (856) 339-2022.

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

Executed on 12/20/17
(Date)

Respectfully,

Eric Carr
Site Vice President
Hope Creek Generating Station

Attachments

1. Response to SRXB Request for Additional Information Regarding MUR Power Uprate – Non-Proprietary (GEH Reference DOC-0006-0106-088 Attachment 2)

2. GEH Affidavit supporting the withholding of information in Attachment 3 from public disclosure

3. Response to SRXB Request for Additional Information Regarding MUR Power Uprate - GEH Proprietary Information (GEH Reference DOC-0006-0106-088 Attachment 1)

Enclosures

1. GE Hitachi Nuclear Energy, "Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation," 002N5176-R0-NP, Revision 0, August 2016 - Non-Proprietary

2. GEH Affidavit supporting the withholding of information in Enclosure 3 from public disclosure

3. GE Hitachi Nuclear Energy, "Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation," 002N5176-R0-P, Revision 0, August 2016 - Proprietary

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. L. Regner, Project Manager, NRC
NRC Senior Resident Inspector, Hope Creek
Mr. P. Mulligan, Chief, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. MacEwen, Hope Creek Commitment Tracking Coordinator

Response to SRXB Request for Additional Information Regarding MUR Power Uprate

Non-Proprietary

This is the non-proprietary version of Attachment 3 of this letter from which the proprietary information has been removed. Portions of the documents that have been removed are indicated by white space inside open and closed brackets as shown here [[   ]].

(GEH Reference DOC-0006-0106-088 Attachment 2)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION BY REACTOR SYSTEMS BRANCH HOPE CREEK GENERATING STATION MEASUREMENT UNCERTAINTY UPRATE DOCKET NO. 50-354

INTRODUCTION

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 SRXB-RAI 1 AND SRXB-RAI 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.

<u>Response:</u>

On September 18, 2006 (ADAMS Accession No. ML062680451), PSEG submitted the Safety Analysis Report for HCGS Constant Pressure Power Uprate (CPPU) as Attachment 4 (NEDC-33076P, Reference 7). This report is referred to as the PUSAR. PUSAR Section 4.1.2.1 states that peak drywell pressure and temperature conditions were determined at 102% of 3952 MWt. These values are shown in PUSAR Table 4-1, as summarized below. As can be seen below, the pressure and temperature values are based on 102% of 3840 MWt or 102% of 3952 MWt, either of which bound TPO. The EPU analyses bound and remain valid for TPO. No changes have occurred that would impact these results.

Hope Creek DBA LOCA Containment Performance Results

	CLTP in UFSAR	CLTP using the CPPU Method[1&2]	CPPU[1]	LIMIT
Peak Drywell Airspace Pressure (psig)	48.1	47.6	50.6	62.0
Peak Drywell Airspace Temp (°F)	291	295	298	340
Peak Bulk Pool Temperature (°F)	210	201	212.3[3]	212/340[4]

1. The CPPU analysis was performed at 102% of 3952 MWt using a realistic decay heat model (ANS/ANSI 5.1, 1979 with 2σ uncertainty), as compared to May-Witt decay heat model used in the current UFSAR analysis.
2. The values in the "CPPU Method" column are the CLTP values recalculated using the CPPU methodology for comparison to the calculated CPPU values.
3. Reported value is based on 102% of CPPU RTP (3840 MWt).
4. The peak bulk suppression pool temperature at CPPU RTP exceeds the current licensing limit, but stays below the design limit. The core spray and RHR pump NPSH margin is addressed in Section 4.2.6.

PUSAR Section 4.2.6 states that CPPU increases the reactor decay heat, which increases the heat input to the suppression pool during an event. This increased heat input increases the peak suppression pool water temperature, which may affect RHR, CS and High Pressure Coolant Injection (HPCI) pump operation. As discussed in PUSAR Section 4.1.1 and shown in Table 4-1, the calculated peak suppression pool temperature for the most limiting case is 212.3°F. Peak suppression pool temperatures for an ATWS, Appendix R, and SBO event are bounded by the most limiting case.

The net positive suction head (NPSH) requirements for the RHR and CS pumps were analyzed using 0 psig containment pressure, as required by RG 1.1, a peak suppression pool water temperature of 218°F, and at pump flows which exceed the design basis pump flows. Using these conditions, the NPSH available is greater than the NPSH required for the RHR and CS pumps. No significant changes have occurred to the RHR or CS systems since EPU that would adversely impact system parameters or performance.

In view of the above, the NRC SER for EPU states in part:

> The licensee's analyses discussed in Section 4.1.2 of the PUSAR demonstrate that the requirements of GDC 38 are satisfied at EPU operating conditions in that containment pressure and temperature following a design basis LOCA are rapidly reduced consistent with the functioning of other systems.
>
> The licensee states that the NPSH requirements for the RHR and CS pumps were analyzed assuming zero psig containment pressure, as specified by Regulatory Guide 1.1. The following table gives the peak suppression pool temperatures for the postulated accidents considered in determining that adequate NPSH margin exists.

Event	Peak Suppression Pool Temperature (°F)
Loss of Coolant Accident	212.3
Limiting ATWS (PRFO)	199.0
Station Blackout	198.0
Appendix R Fire	205.9

> In determining the NPSH margin at HCGS EPU operating conditions, the licensee employed the existing licensing basis methods for determining suction strainer debris loading and head losses. These methods have previously been found acceptable by the staff and their use for EPU is acceptable since the currently approved methods are expected to remain valid for the calculated increase in the peak suppression pool temperature of 11.3°F at the HCGS EPU operating conditions.

Since the EPU analyses bound TPO, and no changes have occurred since EPU that invalidate the analyses results, the existing margins remain valid for TPO.

In addition to the above, the following technical details are provided in response to specific NRC requests of this RAI:

(a) As stated above, no changes have occurred since EPU that would change analysis results given in PUSAR Table 4-1 and as reproduced above. The limiting break for HCGS, from an NPSH standpoint, is and remains a double-ended pipe break of a recirculation suction line. The peak suppression pool temperature of 212.3°F occurs in the long-term at 30,405 seconds (8.45 hours from event initiation) and was calculated assuming 102% of 3840 MWt. The peak short- term suppression pool temperature (conservatively calculated at 102% of 3952 MWt) is 167.3°F which is non-limiting. These numbers were established for the HCGS EPU which bound and remain valid for TPO.

For an SBO event, the peak suppression pool temperature is re-calculated for operation at TPO power level to be 204.6°F. This increase from the EPU peak temperature of 198.0°F incorporates both the increase in power and GEH Safety Communication 10-18. Both are bounded by the CS and RHR NPSH evaluations which were done at 218°F for EPU. It should also be noted that this parameter is inconsequential during an SBO since ECCS pumps are without power, and reactor level is maintained using available HPCI/RCIC systems which take suction from the CST in lieu of the suppression pool.

The 199°F maximum suppression pool temperature for the limiting ATWS event remains the same for TPO since it was established at a power level of 3952 MWt which bounds operation at the TPO power level of 3902 MWt.

(b) As stated above the TPO limiting station event for NPSH remains the same as for EPU, a double ended recirculation suction line break in the long-term. For the RHR and CS pumps, NPSH margins and margin ratios were calculated in accordance with the existing licensing basis using a bounding 218°F suppression pool bulk temperature. No credit is taken for accident pressure. Calculations meet the HCGS licensing basis, including Regulatory Guide 1.82, SRP 6.2.2 and GDC 38. The results of these analysis are summarized below for information:

RHR Loop	**$NPSH_A$ (ft.)**	**$NPSH_R$ (ft.)**	**Margin (ft.)**	**Margin Ratio**
A	6.73	3.0	3.7	2.2
B	6.85	3.5	3.4	2.0
C	5.64	3.0	2.6	1.9
D	5.67	4.0	1.7	1.4
CS Loop	**$NPSH_A$ (ft.)**	**$NPSH_R$ (ft.)**	**Margin (ft.)**	**Margin Ratio**
A (pumps A &C)	6.8*	5.6	1.2	1.2
B (pumps B & D)	6.8*	5.6	1.2	1.2

* $NPSH_A$ based on limiting pump AP206. Other pumps have more $NPSH_A$ and greater margins.

(c) & (d):

As indicated above, the NRC SER for EPU states that NPSH calculations were performed in accordance with HCGS current licensing bases that have been previously found acceptable to the NRC. This includes EPU evaluations that remain bounding for TPO. These analyses meet the requirements of Regulatory Guides 1.1 and 1.82 and Standard Review Plan 6.2.2.

NPSHr was taken from the original manufacturer’s pump curves at the maximum achievable flow. For RHR, the maximum achievable flow was determined during initial startup testing with the LPCI injection valve fully open, corrected for test conditions. For CS, the maximum achievable flow was determined using the station’s benchmarked thermal-hydraulic model, using configurations and assumptions that maximized flow through the CS pumps, corrected for model benchmark error.

SECY-11-0014, Enclosure 1, Section 6.2, Staff Guidance, Approach to Uncertainty, is found to accurately describe the pre-2011 approach taken by utilities like HCGS, to calculating NPSH margin:

> “The current approach to calculating NPSH margin assigns a bounding value to the parameters in the calculation of NPSH margin. These bounding values and assumptions are typically based on historically high or low values or on technical specifications limiting conditions for operation. The chosen accident is also limiting. For example, the worst pipe break (given the most limiting NPSH margin) is assumed for the LOCA, and the worst single failure is assumed). It is also assumed that all these limiting conditions occur simultaneously.”

Both Page 4 and Enclosure 1 state that for those utilities crediting CAP, the licensee should provide additional information to address NPSH conservatisms and explicitly account for uncertainties in the calculation of NPSH margin whenever possible. Such uncertainties associated with this newer BWROG proposed method are described in Enclosure 1 Section 6.3.1 and include the determination of $NPSHr_{eff}$ and consider items such as motor slip, field installed piping configurations, and air content of the water.

Since HCGS does not rely upon containment accident pressure (CAP) in the NPSH analysis, and the HCGS design/licensing basis pre-dates and remains unchanged from these newer expectations, the need to explicitly account for NPSH margin uncertainties for CAP applications as discussed in Enclosure 1 of SECY 11-0014 does not apply to HCGS. Since HCGS is not subject to SECY 11-0014 on use of CAP in analyzing ECCS performance, HCGS does not currently possess the information requested in SXRB-1(c) and (d) such as field motor-slip and comparisons of piping configurations between the vendor test and the plant. As discussed above, HCGS does not require this information to demonstrate adequate NPSH margin since HCGS uses a bounding approach that does not credit CAP.

For both EPU and TPO, there remains considerable margin in ECCS NPSH calculations in that:

- In all cases, NPSHr was determined at flows that exceed those that would be experienced during a design basis event.
- NPSHr is shown to be acceptable at a suppression pool temperature 5.7°F higher than that calculated for the design basis LOCA
- A minimum margin of 1.2 ft exists between NPSHa and NPSHr for the CS and RHR pumps
- No credit is taken for containment accident pressure.

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, 2007 (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, 2007 (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.

<u>Requests</u>

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.

Response:

The EPU PUSAR (Reference 7), Table 6-4, Hope Creek Appendix R Fire Event Evaluation Results, stated that the peak suppression pool temperature of 205.9°F resulted from the limiting Appendix R fire at CPPU. This evaluation was performed in February 2005 and based on an assumed initial core thermal power of 3916.8 MWt or 102% of 3840 MWt which bounds the TPO power level of 3902 MWt. The limiting Appendix R scenario is a shutdown from the Remote Shutdown Panel (RSP).

The evaluation included RCIC, SRVs and one RHR pump in LPCI, suppression pool cooling (SPC) and shutdown cooling (SDC) or alternate shutdown cooling (ASDC) from the RSP. Key assumptions included:

- One stuck open relief valve (SORV) occurs at time zero and remains open throughout the event.
- RCIC is available at 10 minutes.
- SPC starts at 20 minutes and terminates when LPCI injection is initiated.
- 3 SRVs are available for depressurization/cooldown at 100°F/hr.
- Due to a LOP, which is the limiting event for Appendix R, ASDC is started when vessel pressure reaches 80 psig.
- The analysis power is 102% EPU.

In November 2006, the evaluation was re-performed primarily to change the above stated SPC initiation time from 20 minutes to 60 minutes. With the other above listed assumptions remaining the same including an assumed analysis power of 102% of EPU, results were acceptable, and it was determined that the peak suppression pool temperature would increase to 206.3°F.

This change in assumption, and peak suppression pool temperature was reported to the NRC in Reference 8. Table 6-4 of the PUSAR was updated to show the increased suppression pool temperature of 206.3°F and other changed parameters.

The November 2006 evaluation as part of the EPU project is the analysis of record for EPU. The peak temperature (206.3°F) was at that time and continues to be bounded by ECCS NPSH evaluations which were performed at a 218°F pool temperature (refer to the answer to RAI SRXB-1 for additional discussion). Operation at TPO power is bounded by these analyses since they were performed at power levels (102% of 3840 MWt) which are above TPO power of 3902 MWt.

The Appendix R event was assessed and concluded that the Analysis of Record (AOR) for EPU bounded TPO for issues such as NPSH or peak suppression pool temperature. The TPO peak suppression pool temperature remained at 206.3°F as determined for EPU.

The HCGS MSO Compliance Assessment was completed in July 2012, after the AOR for Appendix R. It was performed based on NEI 00-01, Guidance for Post Fire Safe-Shutdown Circuit Analysis, and utilized the MSO expert panel method in developing credible MSO scenarios for HCGS. Review of the MSO scenarios shows that none of the scenarios affect the key assumptions stated above or pose any challenge to ECCS pump NPSH or result in the raising of maximum predicted suppression pool temperatures. As discussed further in the response to RAI SRXB-1, HCGS does not rely on CAP for Appendix R; thus, the concerns discussed in SECY-11-0014 (Reference 4) regarding MSO's are not applicable to HCGS.

In addition to the above, the following technical details are provided in response to specific NRC requests of this RAI:

(a) As discussed above, the peak suppression pool temperature for EPU thermal power was originally determined to be 205.9°F but changed during the EPU project to 206.3°F due to a change in the time suppression pool cooling was assumed to be placed in service (i.e. 20 minutes changed to 60 minutes). This PUSAR change was communicated to the NRC in Reference 8, and remains unchanged for TPO since both analyses assumed a power level of 102% of 3840 MWt which bound TPO power. Thus, the limiting suppression pool temperature for TPO remains 206.3°F.

As discussed above, a review of the MSO scenarios shows that none of the scenarios pose any challenge to ECCS pump NPSH or result in the raising of maximum predicted suppression pool temperatures.

(b) A summary of the Appendix R fire analysis key assumptions, inputs and results, scenarios considered is discussed above.

(c) Please see the response to NRC RAI SRXB-1 for a discussion of the limiting NPSH event (calculated at a bounding 218°F suppression pool temperature) and NPSH margin.

(d) HCGS does not credit CAP for Appendix R.

<u>NRC References for SRXB-1 and 2</u>

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.

Response:

The limiting break for HCGS is and remains a double-ended pipe break of a recirculation suction line.

As stated in TSAR Paragraph 4.1, “current containment evaluations for HCGS were performed at 102% of Current Licensed Thermal Power (CLTP). Although the nominal operating conditions change slightly because of the TPO uprate, the required initial conditions for containment analysis inputs remain the same as previously documented.” Further, EEQ is more specifically discussed in TSAR Section 10.3 Environmental Qualification and in Table 1-3. These sections support that “the resulting environmental conditions are bounded by the existing environmental parameters specified for use in the EQ program,” that “the current accident conditions for temperature and pressure are based on analyses initiated from at least 102% of CLTP,” and that with TPO “EQ remains within current pressure, radiation, and temperature envelopes.” These are the same evaluations that were performed for Extended Power Uprate (EPU). In all cases, the EPU evaluations were performed at power levels of 102% of CLTP or greater; therefore TPO is bounded by these evaluations. No changes have occurred since EPU that would invalidate these evaluations at the TPO power level.

Paragraph 2.3.1.1 of the NRC Safety Evaluation Report for the Hope Creek EPU (ADAMS Accession No. ML073050337) states:

> “The NRC staff’s review focused on the effects of the proposed Hope Creek EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences (AOOs), and accidents. The NRC staff’s review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed Hope Creek EPU. The NRC’s acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment.”

In view of the above, the previously reviewed bounding EEQ evaluations from the HCGS EPU remain valid for TPO since no changes have occurred that would invalidate the evaluations or the resulting conclusions. PUSAR Table 4-1 shows that peak drywell pressure and temperature, evaluated at 2% above 3952 MWt, were determined to be 50.6 psig (with a design limit of 62.0 psig) and 298°F (with a design limit of 340°F). PUSAR Table 10-2, Hope Creek Environmental Qualification for CPPU, shows that the peak accident temperature and pressure due to CPPU increase, but remain bounded by the worst-case accident profile assumed for EQ purposes. Hence, substantial margins to design limits exist at the TPO power level.

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.

Response:

The limiting break for HCGS is and remains a double-ended pipe break of a recirculation suction line.

As stated in TSAR Paragraph 4.1, “current containment evaluations for HCGS were performed at 102% of CLTP. Although the nominal operating conditions change slightly because of the TPO uprate, the required initial conditions for containment analysis inputs remain the same as previously documented.” These are the same evaluations that were performed for Extended Power Uprate (EPU). They were not changed or revised for MUR. In all cases, EPU evaluations were performed at power levels of 102% of CLTP or greater, hence MUR is bounded by these evaluations. No changes have occurred since EPU that would invalidate these evaluations at TPO power.

Paragraph 2.6.1 of the NRC SER for HCGS EPU shows portions of Table 4.1 of the HCGS PUSAR. PUSAR Table 4.1 shows that peak drywell pressure and temperature, evaluated at 2% above 3952 MWt, were determined to be 50.6 psig (with a design limit of 62.0 psig) and 298°F (with a design limit of 340°F). Hence, substantial margins to design limits exist at TPO power.

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."

Response:

TSAR Section 4.1.1, Generic Letter 89-10 Program, states that motor operated valves (MOVs) are not adversely affected by TPO “because previous analyses were either based on 102% of CLTP or are consistent with the plant conditions expected to result from TPO, there are no increases in the pressure or temperature at which MOVs are required to operate with the exception of feedwater valves (slight temperature increase, but no field modifications required). Therefore, the GL 89-10 program remains unchanged following power uprate and the MOVs remain capable of performing their design basis functions.

GL 96-05 addresses the same valve population that is discussed in Section 4.1.1, and for the same reasons that the GL 89-10 program is unaffected (i.e. previous analyses were based on 102% of CLTP or are consistent with the plant conditions expected to result from TPO), the GL 96-05 program is not affect by TPO. As part of the TPO project, modifications to plant systems since EPU were reviewed for possible impact on MOV performance. None were found to alter the conclusions from previous EPU evaluations.

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.

Response:

NEDC-33076, PUSAR Section 4.1.6, which was previously approved during EPU, states that "the Hope Creek response to Generic Letter 96-06 was reviewed for CPPU post-accident conditions. The issues identified in the GL 96-06 review were addressed through procedural changes and piping modifications (i.e. installation of pressure relief valves). These modifications and procedure changes are unaffected by CPPU. Therefore, the existing Hope Creek response to Generic Letter 96-06 remains valid for CPPU."

The evaluation which supports the PUSAR statement above was performed using system parameters corresponding to operation at 3952 MWt, which bounds operation at the TPO power of 3902 MWt. With respect to GL 96-06; the document identified three issues that could affect containment integrity and the operability of safety-related equipment during accident conditions:

1. Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of water-hammer during either a LOCA or a main steam line break,
2. Cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios, and
3. Thermally induced over-pressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions.

Item 1 was mitigated via restrictions on the use of the HCGS drywell coolers to avoid steam voids. Since the resolution of this concern removes the possibility of water hammer without relying on evaluations based on specific plant conditions, it was concluded that power uprates (EPU/TPO) do not have an effect on this issue.

Item 2 was determined not to be applicable to HCGS since the drywell coolers are not credited for LOCA or MSLB mitigation.

Item 3 (thermally-induced piping over-pressurization) was previously determined to be applicable to five drywell penetrations at HCGS. Pressure relief valves were installed in these lines to prevent over-pressurization. Accordingly, a change in maximum accident containment pressure would not affect the potential for thermal over-pressurization of these lines.

Thus, previous GL 96-06 evaluations performed for EPU assumed a power level of 3952 MWt, bounding operation at the TPO power level of 3902 MWt. The specific HCGS resolution of Items 1 through 3 contained within GL 96-06 was not based on a specific design pressure or temperature and remains valid for post-EPU and MUR conditions.

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.

Response:

Hope Creek Calculation No. BC-0052(Q), Plant Cooldown Using One RHR Heat Exchanger, was performed in 2005 in support of the station EPU project. No changes to date have been made which invalidate the conclusions. The calculation was performed assuming a decay heat associated with 102% of 3840 MWt, or 3917 MWt which bounds the TPO power level of 3902 MWt.

The calculation demonstrated that the reactor coolant temperature could be reduced to 200°F within 13.5 hours of plant shutdown using one RHR heat exchanger in the shutdown-cooling mode. This was found to meet the Technical Specification required 24 hour requirement with acceptable margin.

The calculation also demonstrated that if cooldown is conducted using the alternate shutdown cooling means, with one RHR heat exchanger, cold shutdown can be achieved in approximately 60 hours, meeting the 72 hour Appendix R requirement.

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

Response:

Hope Creek Calculation GS-0026, Hardened Containment Vent Capacity, was performed in 2016 in support of the response to NRC order EA-13-109. The calculation demonstrates that the Hope Creek Hardened Containment Vent System (HCVS) has the capacity to vent saturated steam at a flow rate equivalent to one percent of 3,917 MWt at the lesser of primary containment design pressure and Primary Containment Pressure Limit (PCPL).

This bounds the TPO uprate power level of 3902 MWt. The requirement to vent decay heat equal to 1% of the rated thermal power mirrors the requirement of a vent installed per GL 89-16. Therefore the existing hardened containment vent continues to meet the requirements of a vent installed per GL 89-16.

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.

Response:

(a) The non-proprietary and proprietary versions of GE Hitachi Nuclear Energy report 002N5176-R0, Revision 0, “Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation,” August 2016 are provided in Enclosure 1 and 3, respectively.

(b) The NRC approved GE Hitachi Nuclear Energy (GEH) Emergency Core Cooling System (ECCS)-LOCA methodology does not require a mixed core analysis. The justification for not performing a mixed core analysis is that the fuel bundles are channeled, so the different fuel bundle designs co-resident in the core do not present an opportunity for cross flow from bundle to bundle. Since the SAFER-GESTR ECCS-LOCA evaluation model (NEDC-23785-1-PA, Reference 9b-1) presents the core as composed of a modeled “hot channel” and a modeled “average channel,” the model provides further justification that a mixed fuel LOCA analysis is not necessary.

From NEDC-23785-1-PA, Volume II, “Flows in and out of the hot channel are calculated by imposing on the hot channel the plenum to plenum pressure drop from the average core calculation.” Additionally, the hot bundle is modeled assuming the bundle is operating with the maximum radial and axial power peaking factor in conjunction with artificially increasing the bundle power such that it is also operating at a bounding minimum critical power ratio. The modeling results are bounding and sufficiently conservative for any fuel bundle design since this operation would never occur in an actual core.

Since this bundle-centric approach is utilized in the model and there is no cross flow between the fuel bundles in a mixed fuel design core, an ECCS-LOCA analysis for each bundle design will be adequate to conservatively ensure that the predicted peak cladding temperatures for each bundle design are applicable or bounding for any mixed fuel design core configuration.

Reference

9b-1. General Electric Company, “The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident,” Volume II, “SAFER – Long Term Inventory Model for BWR Loss-of-Coolant Analysis,” NEDE-23785-1-PA, Revision 1, October 1984.

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.

Response:

(a) The changes or errors discovered and their effect are identified in the information referenced below.

Summary (GE14):
Base Reference: 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)

Reference	delta PCT	PCT	ADAMS No.
NEDC-33172P	NA	1380	ML053250469
LR-N011-0275	+50	1430	ML11264A035
LR-N13-0210	+45	1475	ML13253A389
LR-N14-0211	+10	1485	ML14273A209
TSAR Table 4-1	NA	1485	

Summary (GNF2):
Base Reference: GE Hitachi Nuclear Energy, "Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation," 002N5176-R0, Revision 0, August 2016.

Reference	delta PCT	PCT	ADAMS No.
002N5176-R0	NA	1610	Provided in response to SXRB-9(a)
TSAR Table 4-1	NA	1610	
LR-N17-0141	-20	1590	ML17270A087

(b) The base GE14 ECCS-LOCA report (NEDC-33172P) and the 10 CFR 50.46 notifications previously submitted to the NRC are accurate and document continued compliance to PCT limits for the GE14 fuel design. Use of the GE14 fuel design is being discontinued at HCGS. At the beginning of Cycle 22 (the first cycle proposed for MUR operation) the core will be comprised of 412 GNF2 fuel assemblies and 352 GE14 fuel assemblies. The GE14 fuel assemblies will be at higher exposures, beyond exposure times of limiting PCT, since they will be in at least their third cycle of operation. A revision to NEDC-33172P to incorporate the changes and errors previously reported to the NRC (per response to 10(a) above), would only address the higher exposure GE14 assemblies. Because of their higher exposure, the limiting PCTs for these assemblies are less than 1485 degrees F as reported in TSAR Table 4-1. Therefore, a revision to NEDC-33172P will not be performed because the currently documented PCT for the GE14 design is bounding and sufficient.

<u>SRXB-11</u>

<u>Regulatory Basis</u>: 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?

<u>Response:</u>

Using the GEH long term containment code, SHEX, identified in TSAR Table 1-1, the peak suppression pool bulk temperature was calculated as [[

]]

Using the above values, the saturation temperature corresponding to the pressure at the safety relief valve (SRV) T-quencher location is obtained as 237.5°F. Although the ATWS events are not subject to the suppression pool local temperature limits in NUREG-0783 (Reference 11-1) or Hope Creek Generating Station (HCGS) Updated Final Safety Analysis Report (UFSAR) Section 6.2.1.1.10, the 20°F subcooling criterion at the SRV T-quencher location stated in NUREG-0783 is conservatively applied to the ATWS event. This results in a suppression pool local temperature limit of 217.5°F.

The bounding temperature difference between the local and bulk suppression pool temperatures is [[ ]] scenario in NEDC-30154 (Reference 11-2). Adding this temperature differential to the calculated bulk temperature of [[ ]] results in a peak suppression pool local temperature of 215.6°F.

References

11-1. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," November 1981.

11-2. General Electric Company, "Hope Creek Generating Station Suppression Pool Temperature Response," NEDC-30154, June 1983.

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

Response:

Reference 12-1, Appendix L.5 provides a generic evaluation of a potential loss of all alternating current power supplies based on previous plant response and coping capability analyses for typical power uprate projects. Changes to parameters/requirements (Items a, b, and c of this Request for Additional Information (RAI)) were generically dispositioned and no Hope Creek Generating Station (HCGS)-specific SBO analysis was performed or required for the Thermal Power Optimization (TPO) uprate. The capacity of the Class 1E batteries, SBO compressed nitrogen requirements, and the evaluation of a loss of ventilation were not changed from the Current Licensed Thermal Power (CLTP) bases.

Reference

12-1. GE Nuclear Energy, “Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” NEDC-32938P-A, Revision 2, May 2003.

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.

Response:

The LEFM system is designed to meet the recommendations of ASME Standard PTC 19.5-2004. The individual sections referenced are addressed below.

Section 10-9.1

As-built dimensions of the Hope Creek LEFM spool piece for all acoustic path lengths and angles used in the mass flow equation were measured, documented, and analyzed for reasonableness after meter manufacture. The effect of thermal expansion on these dimensions in the specific spool piece material at pressures and temperatures encountered during plant operation is well understood and is defined in ASME codes. Bounding uncertainties for each dimensional measurement, as well as for the effect of thermal expansion, are included in the site-specific uncertainty analysis.

Section 10-9.4

The LEFM meter body is an 8-path, crossed-plane configuration as recommended by Section 10-9.4 where high accuracy measurement is required.

Section 10-9.5

Possible flow profile changes as discussed in Section 10-9.5 are addressed through calibration and continuous monitoring of hydraulic parameters in the LEFM electronics. The meter was calibrated at a NIST-traceable laboratory in a hydraulic model representative of the plant piping in which it will be installed. Parametric tests were also performed to characterize the change in meter factor under hydraulic conditions with varying amounts of swirl and transverse velocities. The LEFM electronics continuously monitor several hydraulic parameters, including the flatness of the velocity profile, and provide alarms to indicate a change in flow profile that falls outside the bounding value for the meter factor used in the site-specific uncertainty analysis.

Possible dimensional changes are addressed through plant maintenance procedures. Periodic monitoring of the pipe wall thickness will be performed to confirm that meter dimensions remain within the bounding values of the site-specific uncertainty analysis.

<u>NRC Question References for SRXB-3 through 13</u>

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)

GEH Affidavit supporting the withholding of information in Attachment 3 from public disclosure

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Lisa K. Schichlein, state as follows:

(1) I am a Senior Project Manager, NPP/Services Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Attachment 1 of GEH letter DOC-0006-0106-088, "GEH Responses to NRC TPO SRXB RAIs 9b, 11, and 12 and the GNF2 ECCS-LOCA Report in Support of the Hope Creek TPO LAR," dated December 21, 2017. The GEH proprietary information in Attachment 1, which is entitled "Response to SRXB RAIs 9b, 11, and 12 in Support of the Hope Creek TPO LAR," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. Sec. 552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975 F.2d 871 (D.C. Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary or confidentiality agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed GEH methodology for thermal power optimization for GEH Boiling Water Reactors (BWRs). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience and information databases that constitute a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

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

Executed on this 19th day of December 2017.

Lisa K. Schichlein

Lisa K. Schichlein
Senior Project Manager, NPP/Services Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Road
Wilmington, NC 28401
Lisa.Schichlein@ge.com

GE Hitachi Nuclear Energy, "Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation," 002N5176-R0-NP, Revision 0, August 2016

Non-Proprietary

This is the non-proprietary version of Enclosure 3 from which the proprietary information has been removed. Portions of the documents that have been removed are indicated by white space inside open and closed brackets as shown here [[ ]].

HITACHI

GEH Nuclear Energy

GE Hitachi Nuclear Energy
3901 Castle Hayne Road
Wilmington, NC 28401

002N5176-R0-NP
Revision 0
PLM 002N5176-R2
Class I
August 2016

Non-Proprietary Information

Hope Creek Generating Station

GNF2 ECCS-LOCA Evaluation

Prepared by:
A. V. Trofimova
Lead Engineer, LOCA & Containment Analysis

Approved by:
E. G. Thacker II
Project Manager, Technical Projects

IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

Please Read Carefully

A. Disclaimer

The only undertakings of GE | Hitachi Nuclear Energy Company, LLC (GEH) on behalf of Global Nuclear Fuel, LLC (GNF), (GNF is a Joint Venture of GE, Hitachi, and Toshiba) respecting information in this document are contained in the contract between PSEG Nuclear LLC and GNF for Fuel Bundle Fabrication and Related Services, as amended to the date of transmittal of this document, and nothing contained in this document shall be construed as changing the applicable contract. The use of this information by anyone other than PSEG Nuclear LLC authorized by GNF to have this document, or for any purpose other than that for which it is intended, is not authorized. With respect to any unauthorized use, GNF and GEH make no representation or warranty, express or implied, and assume no liability as to the completeness, accuracy or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

B. Information Notice

This is a non-proprietary version of the document 002N5176-R0-P, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

CONTENTS

TABLES

FIGURES

ACRONYMS

Acronym	Description
ADS	Automatic Depressurization System
BWR	Boiling Water Reactor
BATT	Battery
DEG	Double-Ended Guillotine
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
FFWTR	Final Feedwater Temperature Reduction
FWHOOS	Feedwater Heater Out Of Service
FWTR	Feedwater Temperature Reduction
HPCI	High Pressure Coolant Injection
ICF	Increased Core Flow
LOCA	Loss of Coolant Accident
LHGR	Linear Heat Generation Rate
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MSIV	Main Steam Isolation Valve
NA	Not Available
NFWT	Normal Feedwater Temperature
NRC	Nuclear Regulatory Commission
OLTP	Original Licensed Thermal Power
OPL	Operating Parameter List
PCT	Peak Cladding Temperature
PLHGR	Peak Linear Heat Generation Rate
Pre-EPU	Pre-Extended Power Uprate
SER	Safety Evaluation Report
SLO	Single-Loop Operation
SRV	Safety/Relief Valve

1.0 INTRODUCTION

The purpose of this document is to supplement the ECCS-LOCA evaluation results for Hope Creek Generating Station documented in Reference 8. Specifically, using the limiting cases defined in Reference 8 as a guide along with the latest ECCS-LOCA analysis input (Reference 9) and calculation procedures, results are provided for GNF2 fuel. The analysis methodology is consistent with that defined in Reference 8 and PRIME methodology, which has been approved by the NRC (References 10 and 11). The updated plant ECCS parameters (References 9) are used and the limiting cases are evaluated for GNF2 fuel.

2.0 DESCRIPTION OF MODELS

In Reference 8, the ECCS-LOCA results were generated using the standard four computer models. These models are LAMB, TASC, SAFER and GESTR-LOCA. In this analysis GESTR-LOCA is replaced by the PRIME model, which has been approved by NRC (References 10 and 11).

3.0 ANALYSIS PROCEDURE

3.1 Licensing Criteria

Consistent with Reference 8, the acceptance criteria for the ECCS-LOCA results are based on the Code of Federal Regulations, 10 CFR 50.46.

3.2 SAFER/PRIME-LOCA Licensing Methodology

Consistent with Reference 8 and Reference 10, the ECCS-LOCA analysis is generated using the SAFER/PRIME-LOCA licensing methodology which is discussed in detail in References 1 through 5 (approved by the NRC), Reference 6 and Reference 10. NRC approval of the PRIME methodology is contained in the Licensing Topical Reports (References 10 and 11).

3.3 Generic Analysis

The generic ECCS-LOCA analysis for the BWR/4 product line is described in Reference 3.

3.4 Hope Creek Generating Station Specific Analysis

The Hope Creek specific SAFER/GESTR-LOCA analysis in Reference 8 demonstrates that the nominal and Appendix K PCT trends as a function of break size are consistent with one another and with those of the generic BWR/4 break spectrum curve.

For large breaks in this analysis, the DEG recirculation suction line break with failure of the battery is the limiting break/failure combination for both Appendix K and nominal assumptions.

For small breaks, the 0.08 ft^2 recirculation suction line break with battery failure is the limiting break/failure combination for both Appendix K and nominal assumptions.

4.0 INPUT TO ANALYSIS

As discussed in Section 1, the latest ECCS-LOCA input (Reference 9) including OPL-4/5 is given in Reference 9.

The plant heat balance conditions utilized in this analysis are presented in Table 1. The GNF2 fuel parameters are given in Table 2.

All evaluation model and application updates, up to and including Notification Letter 2014-04, are resolved in this analysis.

Table 1 Plant Operational Parameters

Parameter	Nominal Assumption	Appendix K Assumption
Extended Power Uprate (EPU) (MWt)	3840	3917
Corresponding Power (% of OLTP (1))	116.6	118.9
Pre-Extended Power Uprate (Pre-EPU) (MWt)	3339	3430
Corresponding Power (% of OLTP (1))	101.4	104.2
Rated Case Core Flow (Mlbm/hr)	100	100
Rated Case Core Flow (% Rated)	100	100
MELLLA Case Thermal Power (MWt)	3840	3917
MELLLA Case Core Flow (Mlbm/hr)	94.8	94.8
MELLLA Case Core Flow (% of Rated)	94.8	94.8
SLO Case Core Thermal Power (MWt)	2338	2384
SLO Case Core Flow (Mlbm/hr)	60	60
ICF Core Flow (Mlbm/hr)	105	105
Vessel Steam Dome Pressure (psia)	1020	1055
Nominal Feedwater Temperature (°F) (2)	431.6	434.1
Feedwater Temperature Reduction (°F) (3)	102	102
Number of ADS Valves Assumed Available (4)	4	4
DEG Recirculation Suction Line Break Area (ft^2) (5)	4.096	4.096

(1) Original Licensed Thermal Power (OLTP) is 3293 MWt.
(2) Feedwater temperatures correspond to nominal and Appendix K of EPU / rated flow condition.
(3) Feedwater temperature reduction applies to either FWHOOS or FFWTR.
(4) Five ADS valves are assumed for the small break analysis. Four operable ADS valves (one non-functioning ADS in addition to the single failure) are conservatively assumed for the large break analysis and a separate small break sensitivity study to determine the impact of an ADS valve out-of-service.
(5) The recirculation suction line break area consists of following: (1) 3.54 ft2 – vessel nozzle, (2) 0.54 ft2 – jet pumps, and (3) 0.016 ft2 – bottom head drain. For the LBPCT determination the bottom head drain area is removed for the Appendix K case (4.08 ft2). It is conservatively established by sensitivity analysis to be the limiting case shown in Table 3.

Table 2 GNF2 Fuel Parameters

Fuel Parameter		Analysis Value
PLHGR (kW/ft)	-LOCA Analysis Limit	14.40
	-Appendix K	14.40 × 1.02
	-Nominal	13.75
MAPLHGR (kW/ft)	-LOCA Analysis Limit	13.78
	-Appendix K	13.78 × 1.02
	-Nominal	13.15
Worst Case Pellet Exposure for ECCS Evaluation (MWd/MTU)		14600
Initial Operating MCPR	-LOCA Analysis Limit	1.25
	-Appendix K	1.25 ÷ 1.02
	-Nominal	1.25 + 0.02
Number of Fuel Rods per Bundle		92

5.0 RESULTS

5.1 Large Recirculation Line Breaks

The power/flow points analyzed in this GNF2 fuel transition ECCS-LOCA analysis for DEG large break are EPU/rated flow and EPU/MELLLA flow. The LOCA EPU/MELLLA flow case yields the highest Appendix K PCT. The LOCA EPU/rated flow case yields comparable Appendix K PCT to the limiting case [[ ]].

The mid-peaked and top-peaked axial power shapes are considered for GNF2 large break analysis. The evaluation confirms that the mid-peaked axial power shape yields the limiting PCT for large breaks.

Several large recirculation suction line breaks are analyzed for GNF2 with nominal and Appendix K assumptions to confirm the limiting large break and single failure combination. These analyses include dryout times from LAMB/TASC calculations that determine early boiling transition. The LAMB/TASC results show that early boiling transition occurs in the top six axial fuel nodes. These results show that the limiting break and single failure combination is the DEG recirculation suction line break with battery failure for both nominal and Appendix K assumptions.

The results of these analyses are given in Table 3 and Figures 1 and 2. The most limiting Appendix K case is the DEG recirculation suction line break with battery failure at EPU/MELLLA flow condition with a mid-peaked axial power shape assumption. The corresponding nominal case at this power flow condition is the basis for a sensitivity study in the Licensing Basis PCT calculation.

For the DEG recirculation suction line break with battery failure at MELLLA flow and with nominal assumptions, there is a rapid vessel depressurization (Figure 1-b) due to vessel inventory loss through the break. This mass loss causes core uncovery (Figure 1-a). Before core uncovery there is an early bundle heatup due to the stored energy in the bundle (Figure 1-c). This is caused by early dryout of the upper axial nodes in the bundle, which is a result of the rapid core flow coastdown. As the saturation pressure drops due to vessel depressurization the nodes rewet because the water in the lower plenum saturates and flashes up through the core, causing a temperature decrease in the cladding. The LPCS and LPCI begin injection after their pressure permissives are reached for the injection valves and the injection values open (Figure 1-e). When the core uncovers, there is a second bundle heatup. As the level in the core begins to recover, secondary heat transfer mechanisms (e.g., steam cooling) are effective in overcoming the temperature excursion, resulting in a peak cladding temperature (Figure 1- c), which lasts until the core fully recovers (Figure 1- a).

The system response to the DEG recirculation suction line break with battery failure at MELLLA flow and with Appendix K assumptions is similar to the corresponding case with nominal assumptions. There is a rapid vessel depressurization (Figure 2-b) due to vessel inventory loss through the break. This vessel depressurization is faster than the nominal case due to higher break flow from the Appendix K Moody Slip Flow Model. This higher mass loss also causes an earlier core uncovery (Figure 2-a). Before core uncovery, there is an early bundle heatup due to the stored energy in the bundle (Figure 2-c). This is caused by early dryout of the upper axial nodes in the bundle, which is a result of the rapid core flow coastdown. The fuel rods remain in film boiling for a longer time than the nominal case because of the more restrictive Appendix K assumptions, which do not allow the bundle to change from film boiling to transition boiling until the cladding superheat falls below 300°F. The LPCS and LPCI begin injection (following the same initiation sequence as the nominal case). Then there is a second bundle heatup, which lasts until the water level reaches the top of active fuel (Figure 2-a and 2-d). Overall the bundle heatup for the Appendix K case is higher than the nominal case due to higher bundle power, decay heat and break flow.

The FWTR analysis of Reference 12 assumes a 102 °F reduction in the initial feedwater temperature. The PCT results for the FWTR condition at EPU/MELLLA flow are shown in Table 3 with nominal and Appendix K assumptions, respectively. These calculations show that the PCT results for the case at EPU/rated flow, at EPU/MELLLA flow and nominal feedwater temperature (NFWT) are higher than the FWTR PCT results.

5.2 Small Recirculation Line Breaks

The mid-peaked and top-peaked axial power shapes are both considered in the ECCS-LOCA small break analysis for GNF2. The evaluation confirms that the top-peaked axial power shape yields the limiting PCT for small breaks.

The most limiting single failure for small recirculation line breaks is the recirculation suction line break with battery failure. The small break cases are analyzed for GNF2 with nominal and Appendix K assumptions at rated conditions to determine the small break with the highest PCT. The results of these analyses are given in Table 4 and Figures 3 and 4. From these analyses, the most limiting small break is the 0.08 ft^2 recirculation suction line break for both nominal and Appendix K assumptions. The limiting nominal and Appendix K small recirculation break PCTs are lower than the limiting large recirculation line break PCTs. These results confirm that the recirculation suction line small break is not the limiting LOCA event.

For the 0.08 ft^2 recirculation suction line break case with battery failure and nominal assumptions, scram is assumed to occur at the start of the event on low water level. Following the scram, the vessel pressure falls to the pressure regulator setpoint (Figure 3-b). When the main steamline isolation valves (MSIV) close on low water level, the vessel pressure begins to rise until the Safety/Relief Valve (SRV) setpoint is reached. Pressure is maintained between the SRV opening and closure setpoints until the Automatic Depressurization System (ADS) actuation occurs (Figure 3-b), which has a delay after the low water level signal. Continued vessel inventory loss through the break and the ADS actuation cause core uncovery (Figure 3-a) and bundle heatup (Figure 3-c). Once the vessel pressure has dropped below the injection valve permissive pressures and the injection valves open, the LPCS and LPCI start to inject (Figure 3-e). A two-phase level is rapidly established in the bundles and terminates the bundle heatup (Figure 3-c).

The accident progression for the 0.08 ft^2 recirculation suction line break case with battery failure and Appendix K top-peaked axial power shape assumptions is similar to the nominal case. Due to higher bundle power and decay heat, the pressurization is faster following MSIV closure and the Safety/Relieve Valves (SRV) setpoint is reached sooner. Pressure is maintained between the SRV opening and closure setpoints until the ADS system actuates (Figure 4-b). Continued vessel inventory loss through the break and the ADS actuation cause core uncovery and bundle heat-up (Figures 4-a and 4-c). Once the vessel pressure has dropped below the injection valve permissive pressures and the injection valves open, the LPCS and LPCI start to inject (Figure 4-e). After ECCS injection a two-phase level is rapidly established in the bundles and terminates the bundle heatup (Figure 4-c).

5.3 Non-Recirculation Line Breaks

The analysis in Reference 8 demonstrates that the non-recirculation line break cases are clearly non-limiting. These results are a strong function of break location and size (as opposed to the fuel characteristics) and are not expected to change with the change in fuel. Therefore, these cases are not analyzed for the GNF2 transition.

5.4 Alternate Operating Modes

The limiting large break, the DEG recirculation suction line break, is evaluated for the other alternate operating modes: Increased Core Flow (ICF) and Single-Loop Operation (SLO). The Feedwater Temperature Reduction (FWTR) analysis results are discussed in Section 5.1.

The effect of ICF operation (up to 105% of rated core flow) on the ECCS-LOCA analysis is a slight delay in the onset of early boiling transition for the axial nodes in the upper part of the bundle. This results in a lower calculated PCT for these nodes. However, ICF does not affect the dryout time of the high-powered node which determines the overall bundle PCT. Therefore the effect of ICF operation on the ECCS-LOCA analysis results is negligible. Thus, the PCTs for the limiting large break cases given in Section 5.1 are applicable to the ICF condition.

The SLO analysis (60.86% of EPU and 60.0% of rated core flow) conservatively assumes the simultaneous dryout of all axial fuel nodes immediately following the initiation of the event. With the single loop operation multipliers on LHGR and MAPLHGR shown in Table 6, the analysis ensures that the nominal SLO case will be bounded by the two-loop results. With Appendix K assumptions the PCT remains below the 2200°F licensing limit.

5.5 Compliance Evaluation

5.5.1 Licensing Basis PCT Evaluation

The Appendix K results confirm that the limiting break is the DEG recirculation suction line break with battery failure (Table 3). The Licensing Basis PCT for Hope Creek is calculated for

GNF2 fuel based on the Appendix K PCT, applying the NRC approved SAFER/PRIME-LOCA licensing methodology described in References 3 and 10. Hope Creek unique variable uncertainties, including ECCS initiation signal, fuel rod gap pressure, backflow leakage and stored energy are evaluated specifically for GNF2 fuel to determine plant-specific adders. The calculated Licensing Basis PCT is shown in Table 5.

The Licensing Basis PCT was calculated for MELLLA flow and at EPU thermal power. This evaluation demonstrated that the licensing acceptance criteria and the SER requirements are met for all the operating points.

5.5.2 Plant-specific Upper Bound PCT Calculation

The primary purpose of the Upper Bound PCT calculation is to demonstrate that the Licensing Basis PCT is sufficiently conservative by showing that the Licensing Basis PCT is higher than the Upper Bound PCT. The NRC eliminated the 1600°F restriction on the Upper Bound PCT (Reference 4).

Reference 4 provided generic justification that the Licensing Basis PCT is conservative with respect to the Upper Bound PCT and that the plant-specific Upper Bound PCT calculation is no longer necessary. The NRC SER in Reference 4 accepted this position by noting that since plant-specific Upper Bound PCT calculations were performed for all plants, other means may be used to demonstrate compliance with the original SER limitations. These other means are acceptable provided there are no significant changes to the plant configuration that would invalidate the existing Upper Bound PCT calculations. For the Upper Bound PCT calculation, the plant configuration includes the plant equipment and equipment performance (e.g., ECCS pumps and flow rates), fuel type, and the plant operating conditions (e.g., core power and flow) that may affect the PCT calculation. In order to demonstrate continued compliance with the original SER limitations, the PCT impact due to the changes in the plant configuration (in this case GNF2 fuel transition) are reviewed to confirm that the conclusions based on the original Upper Bound PCT calculation are not invalidated by the changes.

The Upper Bound PCT was calculated based on the DEG recirculation suction line break for nominal assumptions with battery failure for EPU / MELLLA flow. The results show that the Licensing Basis PCT for all the power/flow points is below the 10 CFR 50.46 limit of 2200°F and the Licensing Basis PCT bounds the Upper Bound PCT. Therefore, 10 CFR 50.46 acceptance criteria and the NRC SER requirements for SAFER/PRIME-LOCA methodology are met for all the operating conditions.

Table 3 Summary of Large Recirculation Line Break Results

		Nominal Peak Cladding Temperature		Appendix K Peak Cladding Temperature	
Power/Flow Condition, Break Location, and Break Size	Failure	First Peak (°F)	Second Peak (°F)	First Peak (°F)	Second Peak (°F)
[[					
					]]

[[
]]

Table 4 Break Area Sensitivity Study for Small Recirculation Suction Line Breaks with Battery Failure

Break Area (ft²)	Nominal Peak Cladding Temperature (°F)	Appendix K Peak Cladding Temperature (°F)
[[		
		]]

[[
]]

6.0 CONCLUSIONS

The analysis contained in this report demonstrates that for GNF2 fuel transition for Hope Creek Generating Station, the limiting break and single failure combination is the DEG recirculation suction line break with battery failure for both Nominal and Appendix K assumptions.

Based on the limiting large and small breaks, and applying the SAFER/PRIME-LOCA methodology, the Hope Creek ECCS-LOCA analysis is performed for the limiting LOCA event for GNF2 fuel with an analytical PLHGR of 14.40 kW/ft. The results are summarized in Table 5.

The analysis demonstrates that the Licensing Basis PCT occurs for the DEG recirculation suction line break under the EPU power/ MELLLA flow condition with battery failure and mid-peaked axial power shape assumption.

These results meet all licensing and SAFER/PRIME-LOCA methodology analysis requirements. The maximum local oxidation and core-wide metal water reactions in Table 5 are the maximum from all SAFER cases analyzed.

Compared to the Licensing Basis PCT for GE14 reported in Reference 8, Licensing Basis PCT is increased by 230°F for GNF2 fuel transition. This Licensing Basis PCT increase is attributed to fuel change and correction of all known ECCS-LOCA analysis Notification Letters up to and including 2014-04.

ECCS-LOCA analysis results for all alternate modes (MELLLA, FFWTR, FWHOOS, ICF, SLO) also meet all licensing limits. The thermal limits applied to the GNF2 fuel in the ECCS-LOCA evaluation are summarized in Table 6. All ECCS-LOCA results are independent of cycle length.

Table 5 Hope Creek ECCS-LOCA Analysis Results for GNF2

Parameter	GNF2 Result	Acceptance Criteria
1. Licensing Basis PCT (1)	1610 °F	≤ 2200 °F
2. Upper Bound PCT	1600 °F	≤ LBPCT
3. Maximum Local Oxidation	< 1%	≤ 17%
4. Core-Wide Metal-Water Reaction	< 0.1%	≤ 1.0%
5. Coolable Geometry	Items 1 AND 2	Satisfied by: PCT ≤ 2200 °F AND Maximum Local Oxidation ≤ 17%
6. Core Long-Term Cooling	Satisfied by: EITHER Core reflooded above TAF OR Core reflooded to the elevation of jet pump suction and one core spray system in operation (2)	Core temperature acceptably low AND Long-term decay heat removed

(1) LBPCT calculation is based on the 1591°F case.
(2) Based on Reference 7, adequate spray distribution for long-term cooling is achieved when the core spray system flow rate at 0 psid is at the original design rated system flow rate.

Table 6 Hope Creek Thermal Limits for GNF2

Fuel Parameter	Analysis Value	
Fuel	GNF2	
PLHGR – Exposure Limit Curve	MWd/MTU	kW/ft
	0	14.40
	14600	14.40
	67000	6.87
	70000	5.50
MAPLHGR – Exposure Limit Curve	MWd/MTU	kW/ft
	0	13.78
	18917	13.78
	67000	6.87
	70000	5.50
Initial Operating MCPR -Analysis Limit -Appendix K -Nominal	 1.25 1.25÷1.02 1.25+0.02	
R-Factor	0.93	
SLO Multiplier on LHGR and MAPLHGR	0.80	

7.0 REFERENCES

1 NEDE-23785-1-PA, Vol. I, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume I, GESTR-LOCA – A Model for the Prediction of Fuel Rod Thermal Performance," Revision 1, General Electric Company, October 1984.

2 NEDE-23785-1-PA, Vol. II, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume II, SAFER – Long Term Inventory Model for BWR Loss-of-Coolant Analysis," Revision 1, General Electric Company, October 1984.

3 NEDE-23785-1-PA, Vol. III, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," Revision 1, General Electric Company, October 1984.

4 NEDE-23785P-A, Vol. III, Supplement 1, "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," Revision 1, General Electric Company, March 2002.

5 NEDC-32084P-A, "TASC-03A A Computer Program for Transient Analysis of a Single Channel," Revision 2, July 2002.

6 Letter, Richard E. Kingston (GEH) to NRC, "Transmittal of Revision 1 of NEDC-32950, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," MFN 07-406, July 31, 2007.

7 Letter OG00-0382-01 from T. Hurst to BWR Owner's Group, "GE Position Summary Regarding Long-Term Post-LOCA Adequate Core Cooling Requirements", November 16, 2000.

8 NEDC-33172P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis for Hope Creek Generating Station at Power Uprate", March 2005.

9 NFS 16-053, "GNF2 New Fuel Introduction Task F0407 ECCS-LOCA Design Input Request", Revision 0, April 8, 2016.

10 NEDC-33256P-A, Part 1 - Technical Bases, Revision 1, NEDC-33257P-A, Part 2 – Qualification, Revision 1, NEDC-33258P-A, Part 3 - Application Methodology, Revision 1, "Licensing Topical Report The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance", Global Nuclear Fuel, September 2010.

11 NEDO-33173 Supplement 4-A, "Licensing Topical Report Implementation of PRIME Models and Data in Downstream Methods", Revision 1, GE Hitachi Nuclear Energy, November 2012.

12 NEDC-33506P, "Hope Creek Generating Station Operation with Final Feedwater Temperature Reduction and Feedwater Heaters Out-Of-Service", September 2010.

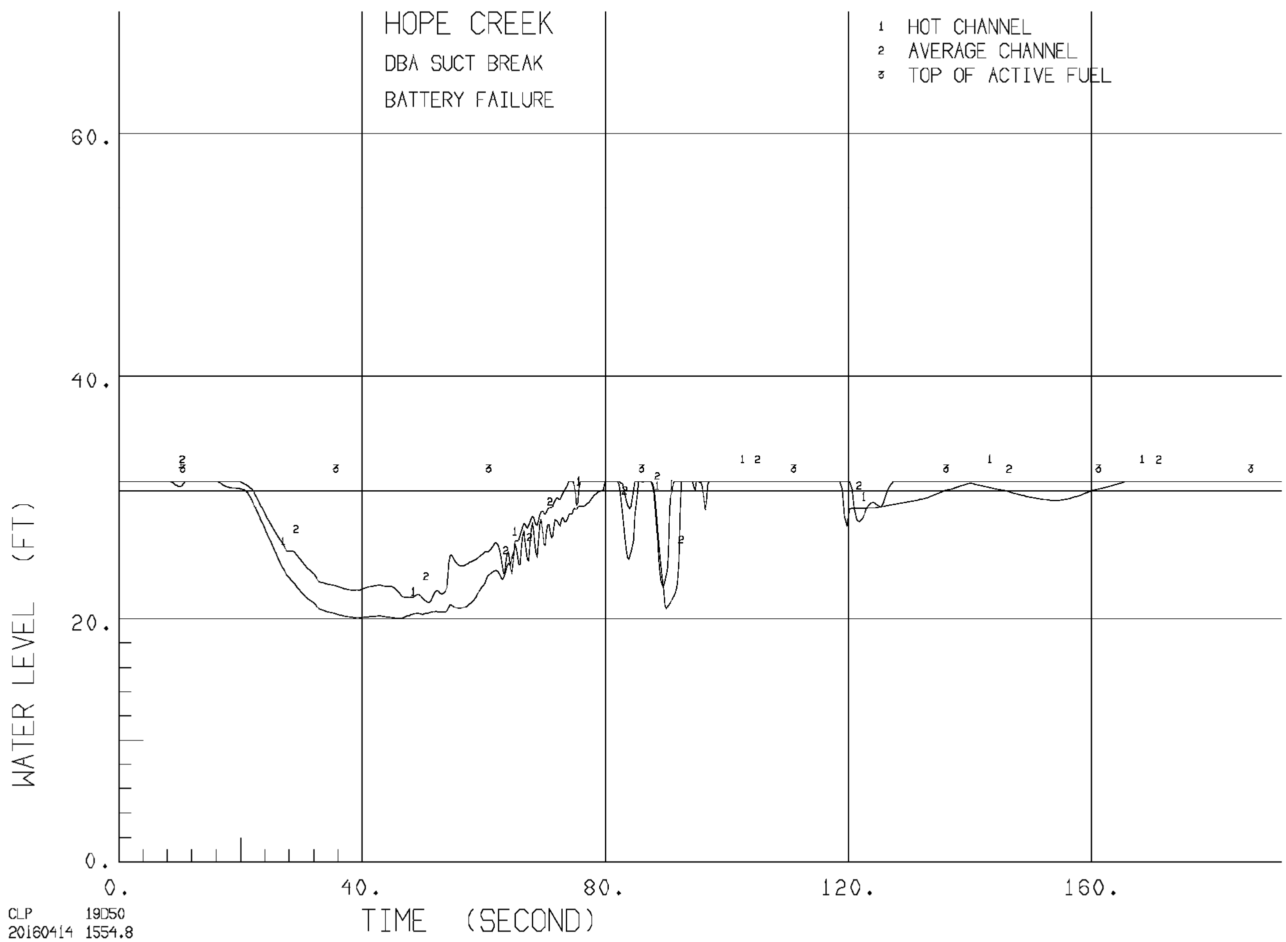

Figure 1-a Water Level in Hot and Average Channels,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Nominal Assumptions

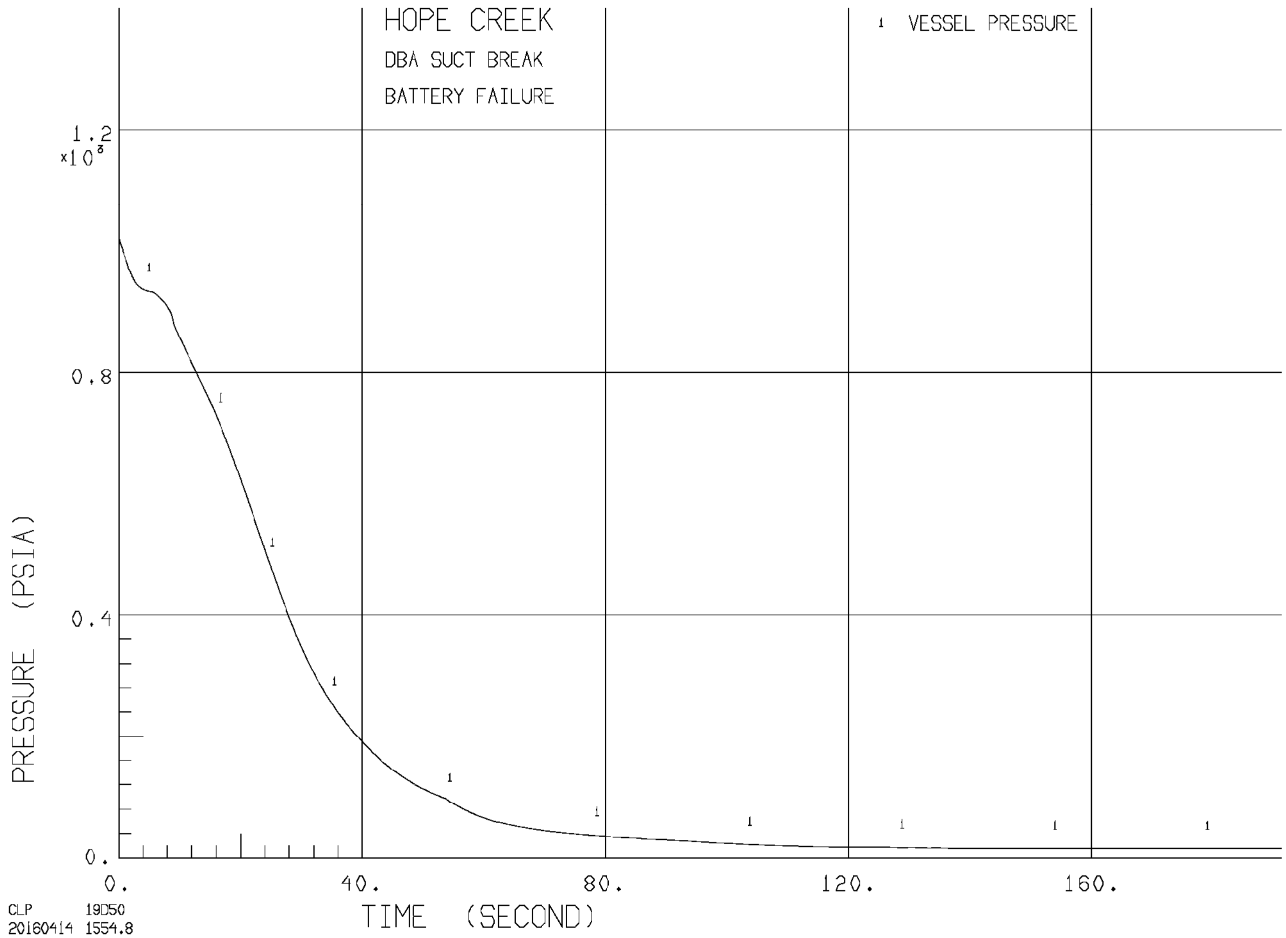

Figure 1-b Reactor Vessel Dome Pressure,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Nominal Assumptions

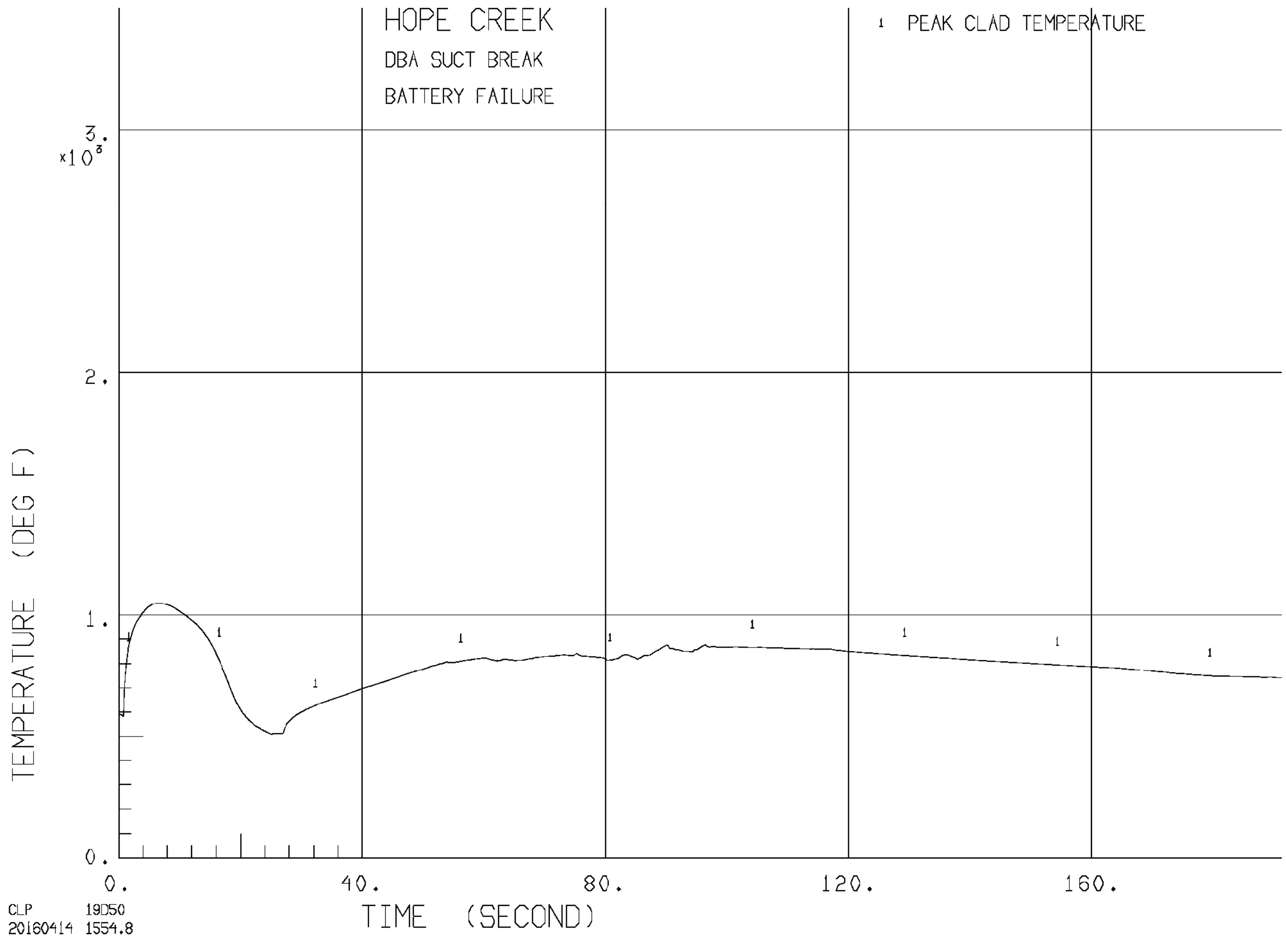

Figure 1-c Peak Cladding Temperature,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Nominal Assumptions

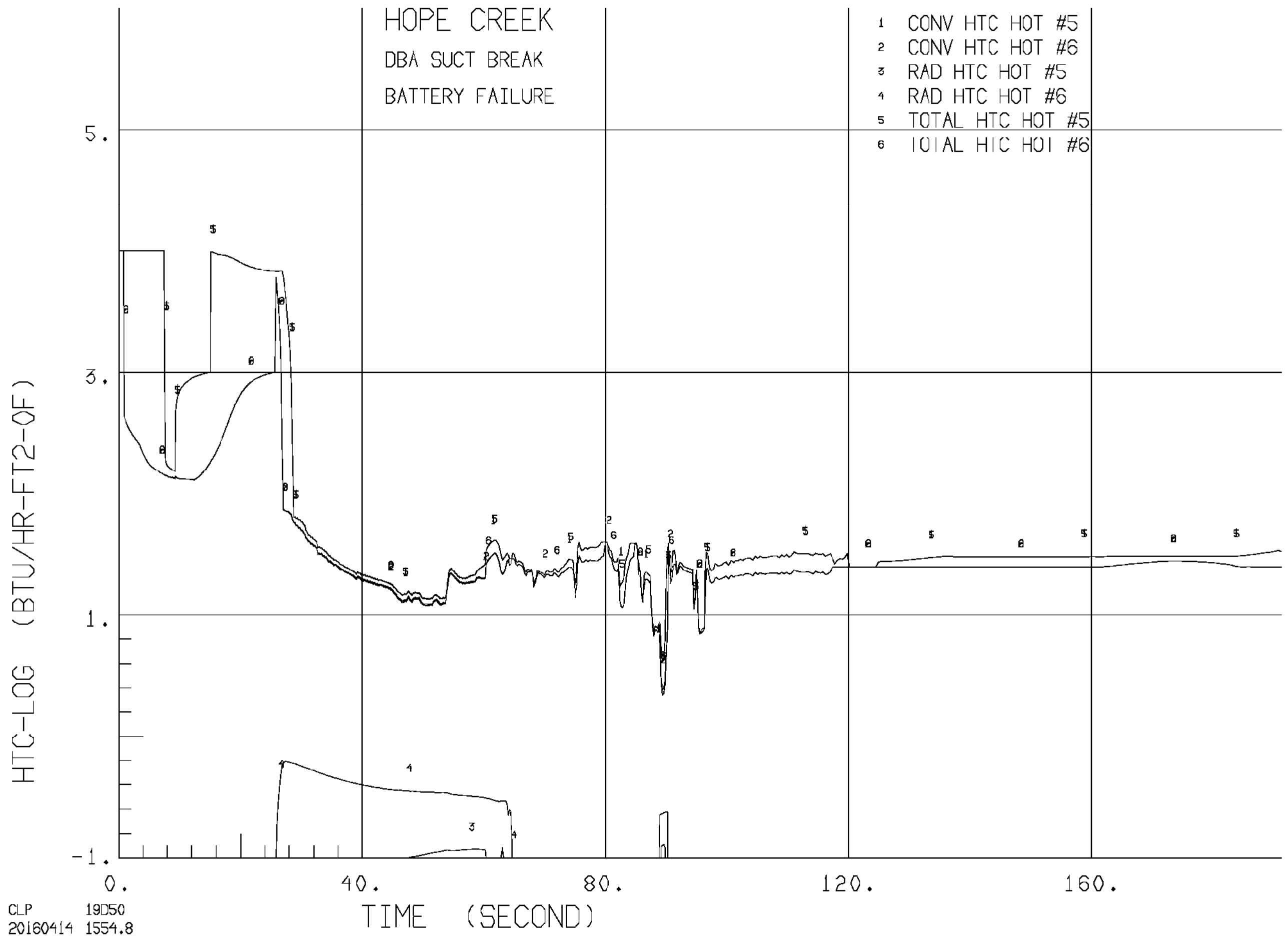

Figure 1-d Heat Transfer Coefficients,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Nominal Assumptions

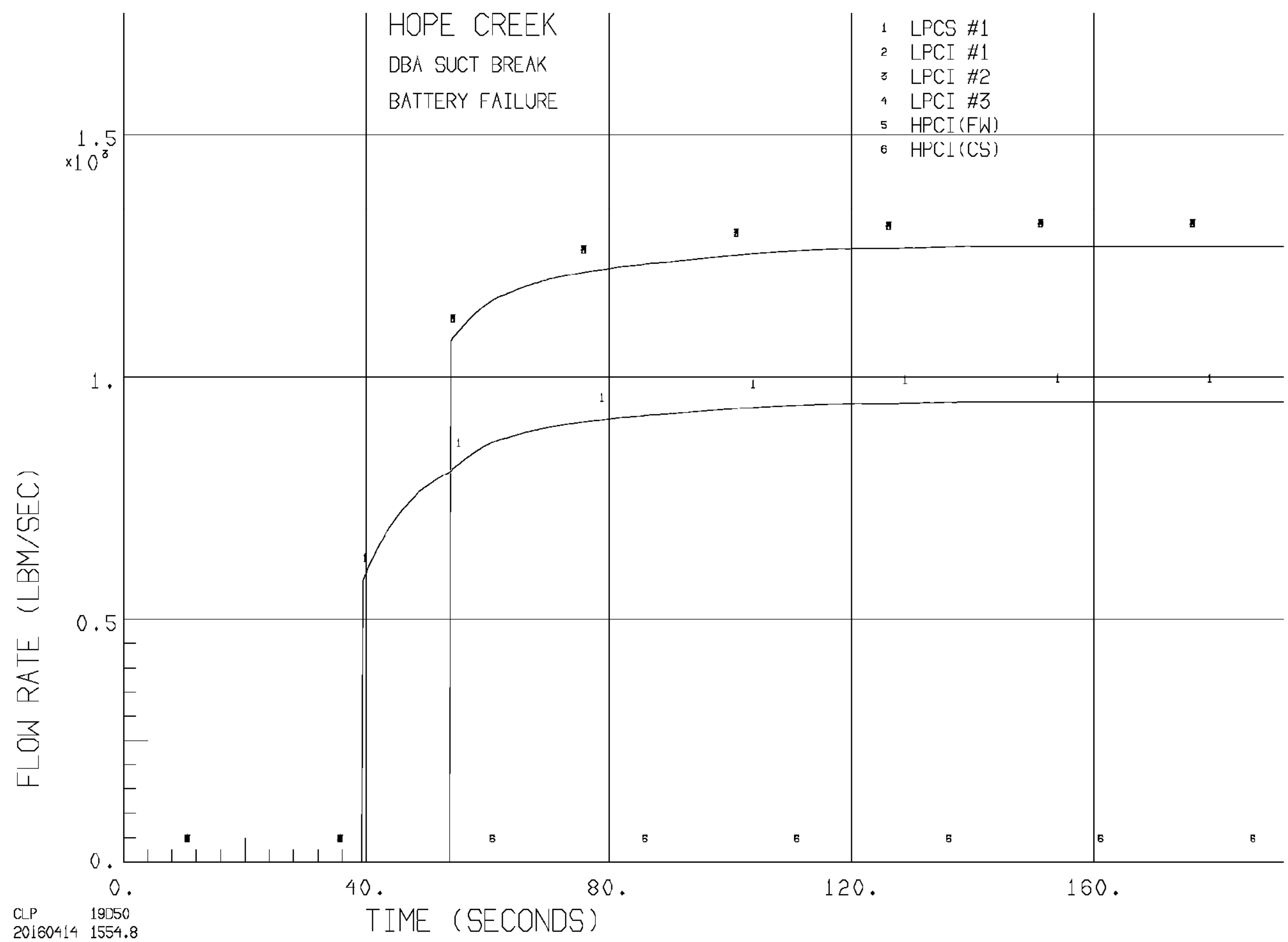

Figure 1-e ECCS Flow Rates,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Nominal Assumptions

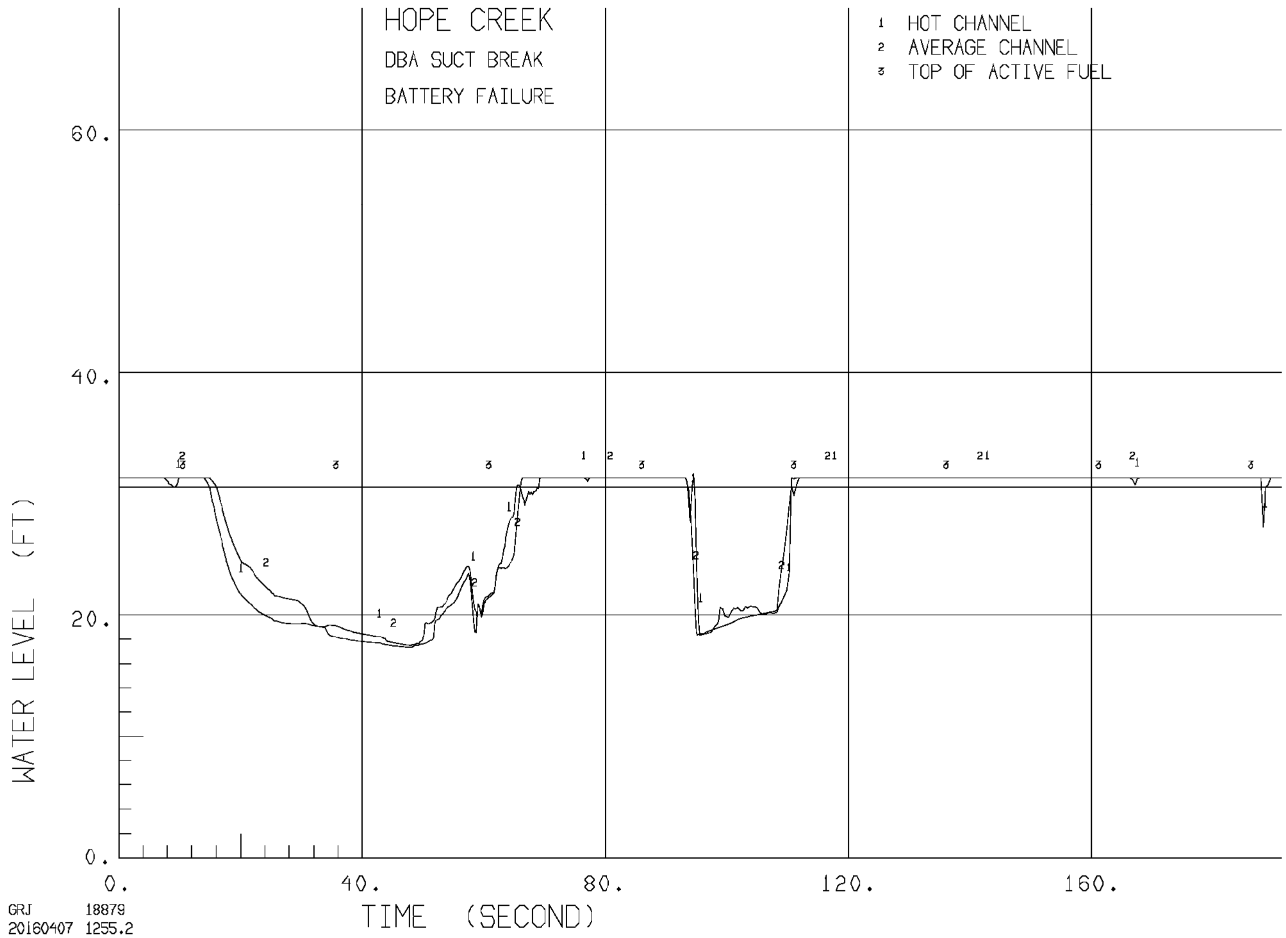

Figure 2-a Water Level in Hot and Average Channels,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Appendix K Assumptions

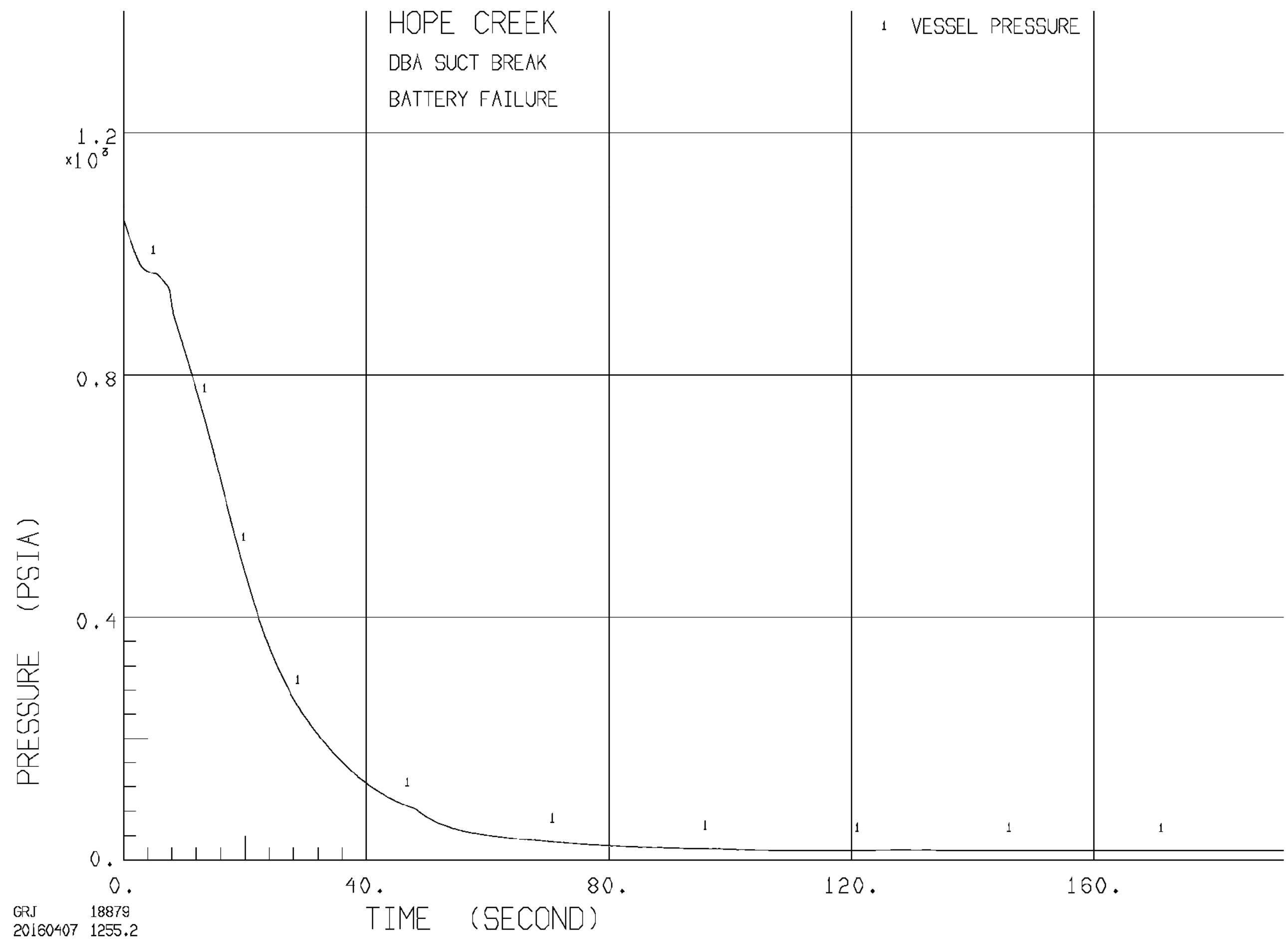

Figure 2-b Reactor Vessel Dome Pressure,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Appendix K Assumptions

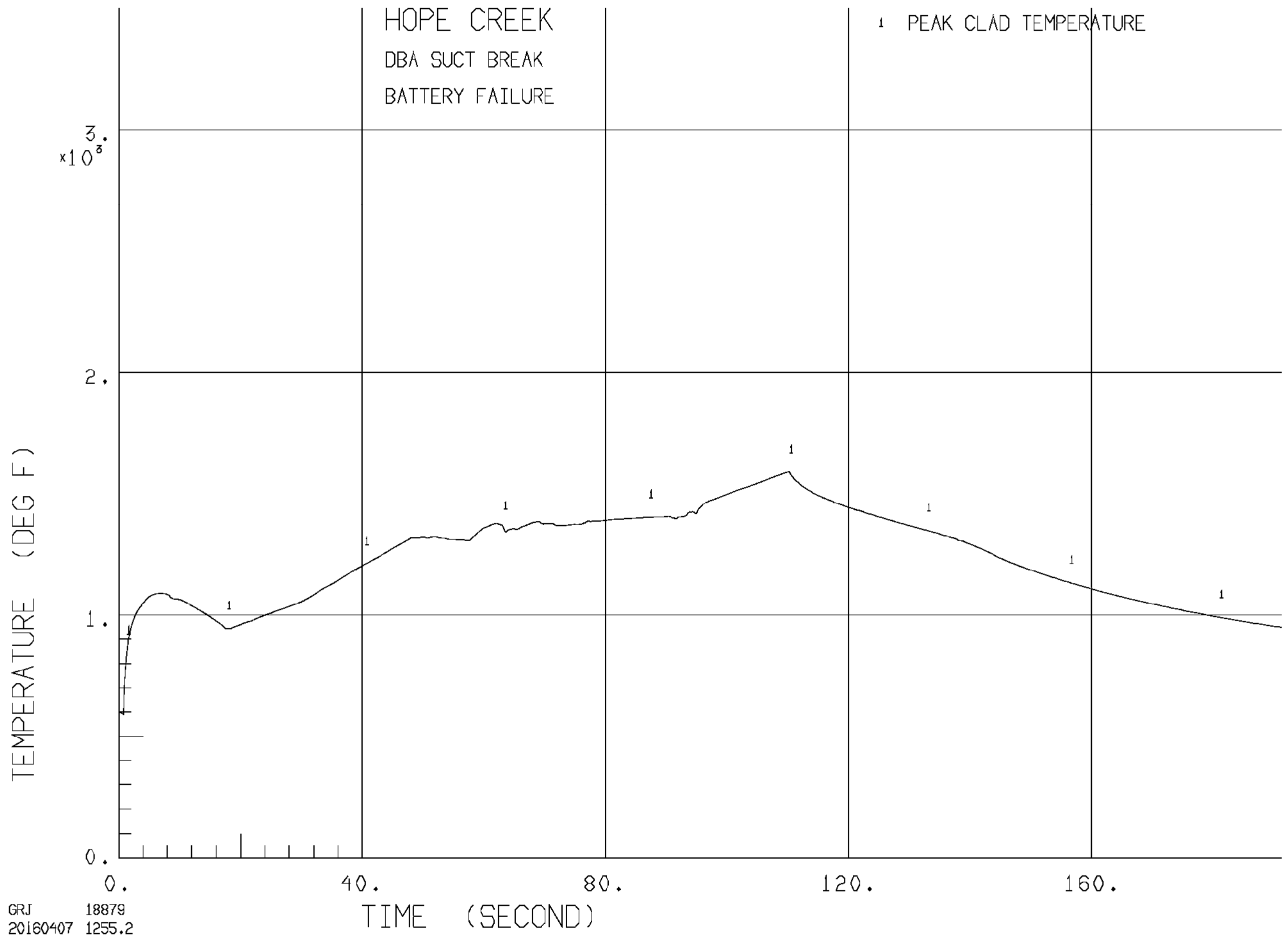

Figure 2-c Peak Cladding Temperature,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Appendix K Assumptions

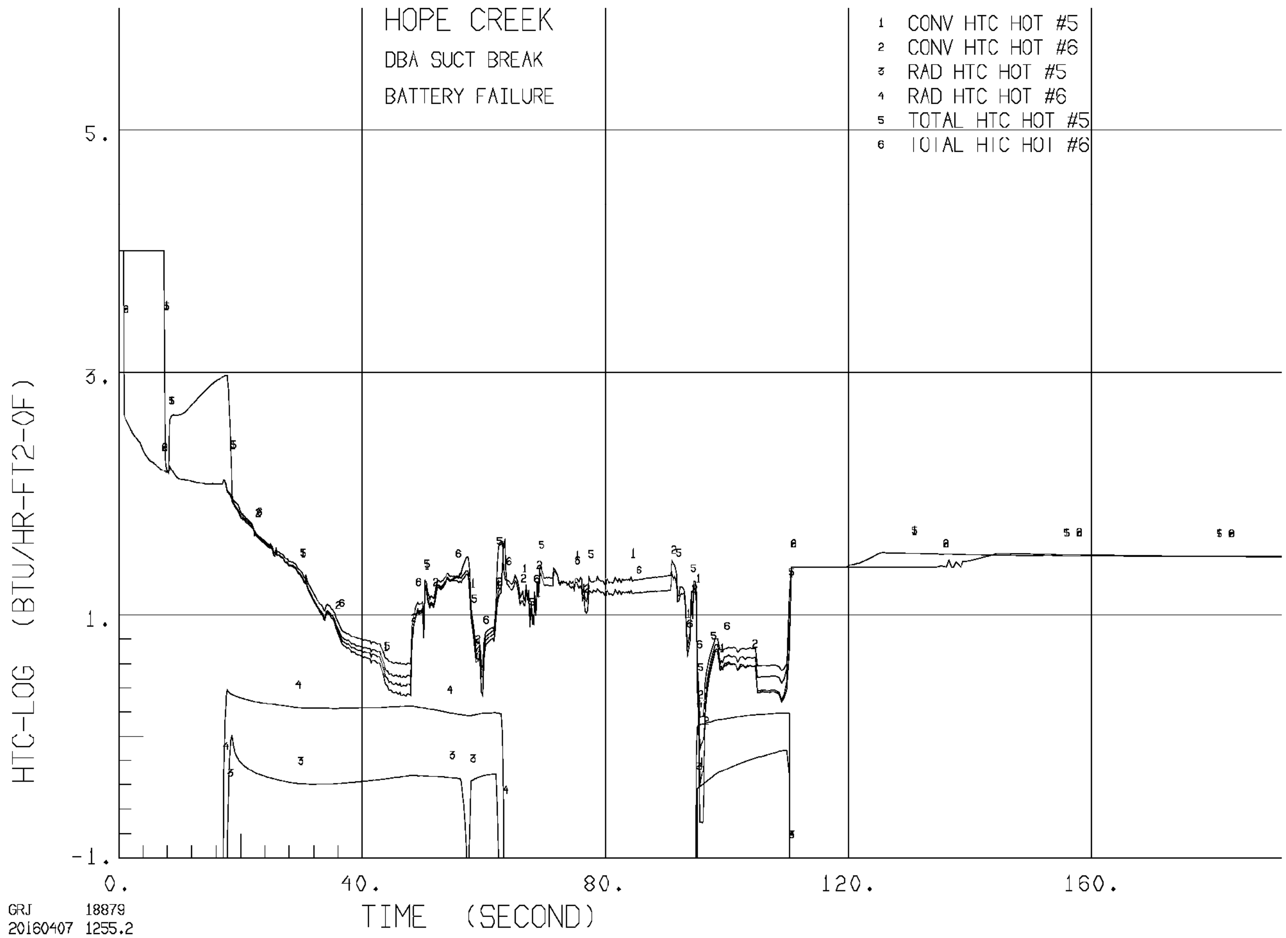

Figure 2-d Heat Transfer Coefficients,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Appendix K Assumptions

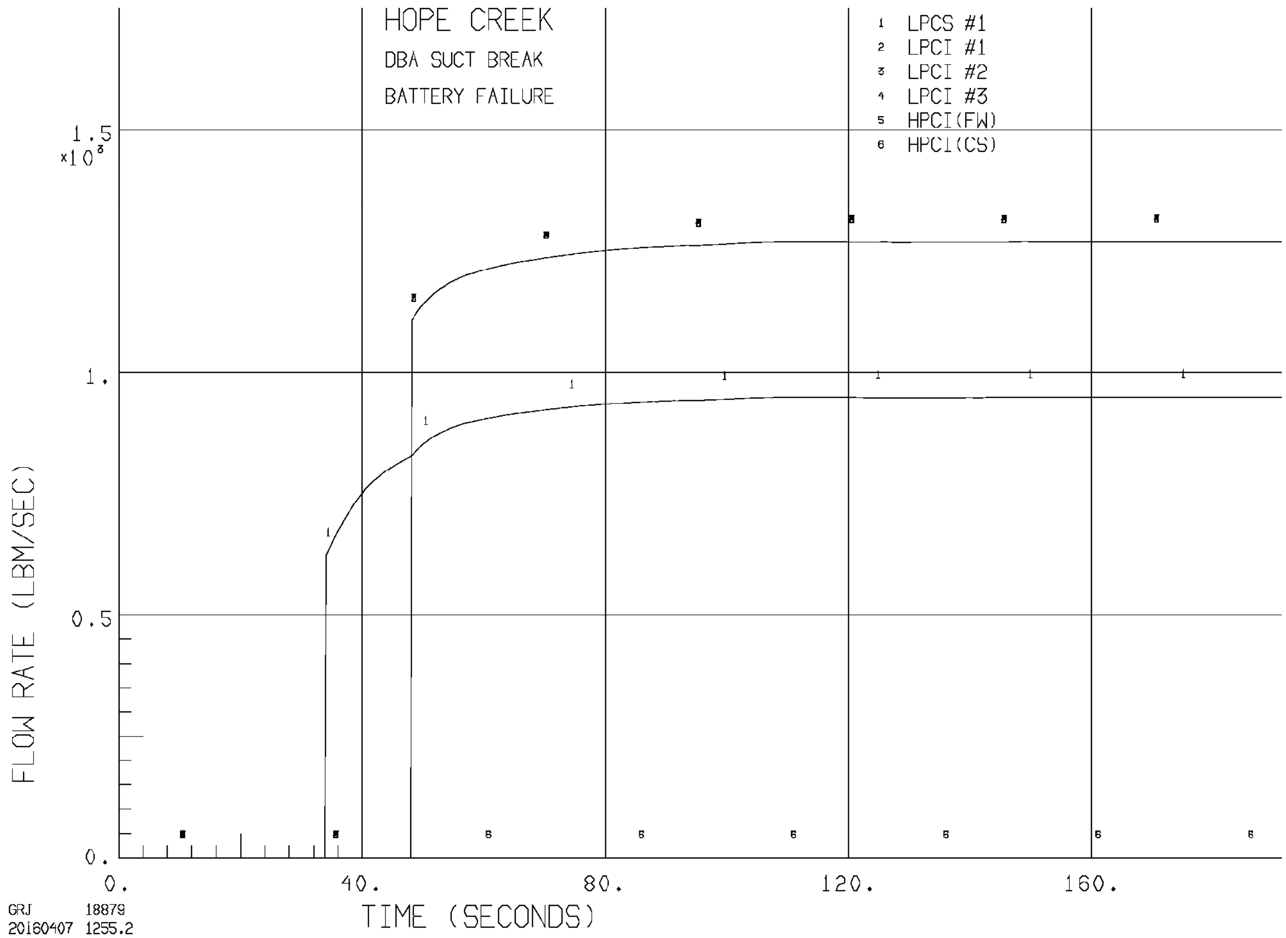

Figure 2-e ECCS Flow Rates,
Limiting Large Recirculation Suction Line Break (DEG), Battery Failure, MELLLA, GNF2 Fuel
1LPCS + 3LPCI + 4ADS Available, Appendix K Assumptions

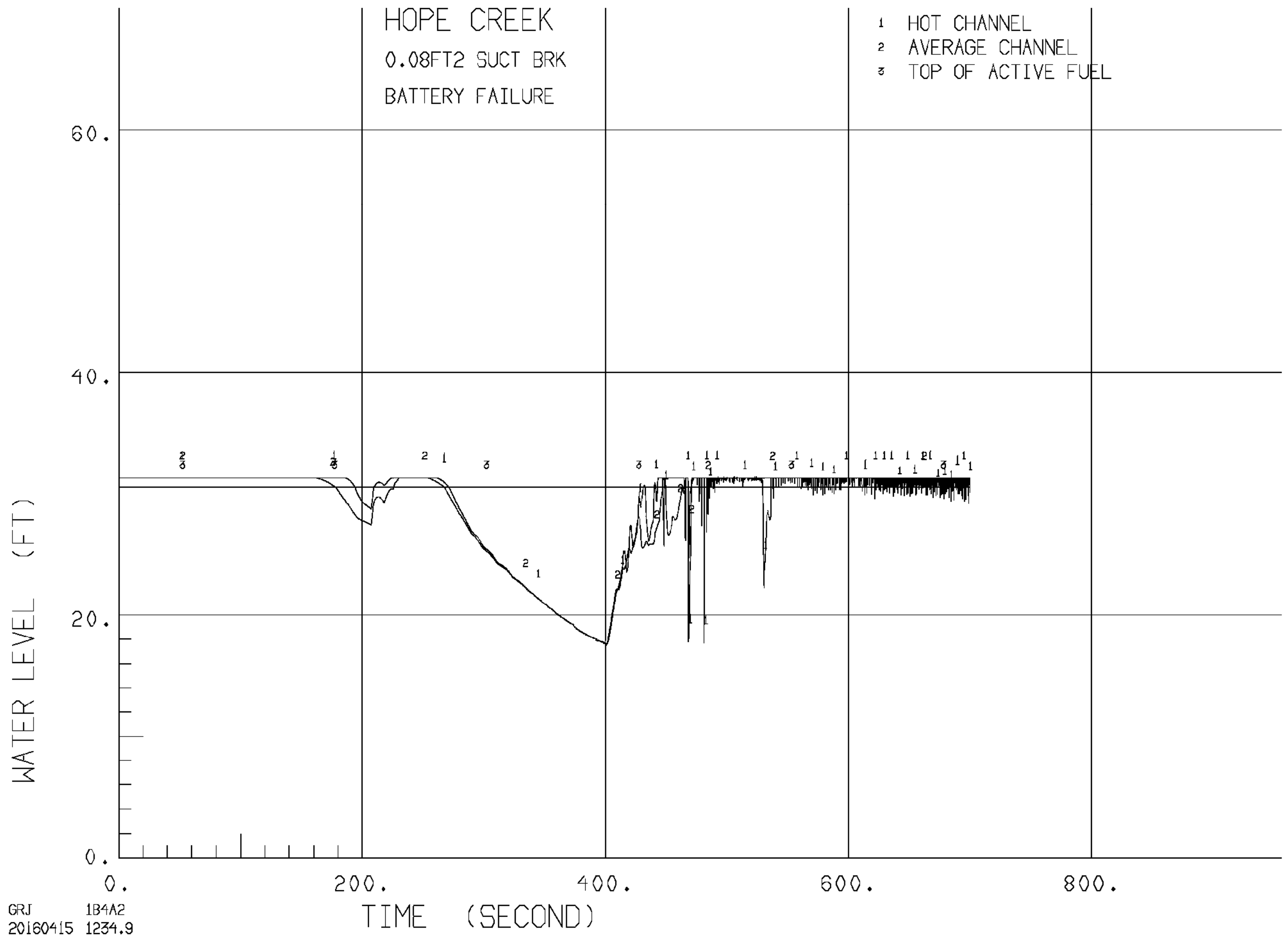

Figure 3-a Water Level in Hot and Average Channels,
Limiting Small Recirculation Suction Line Break (0.08 ft²), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Nominal Assumptions

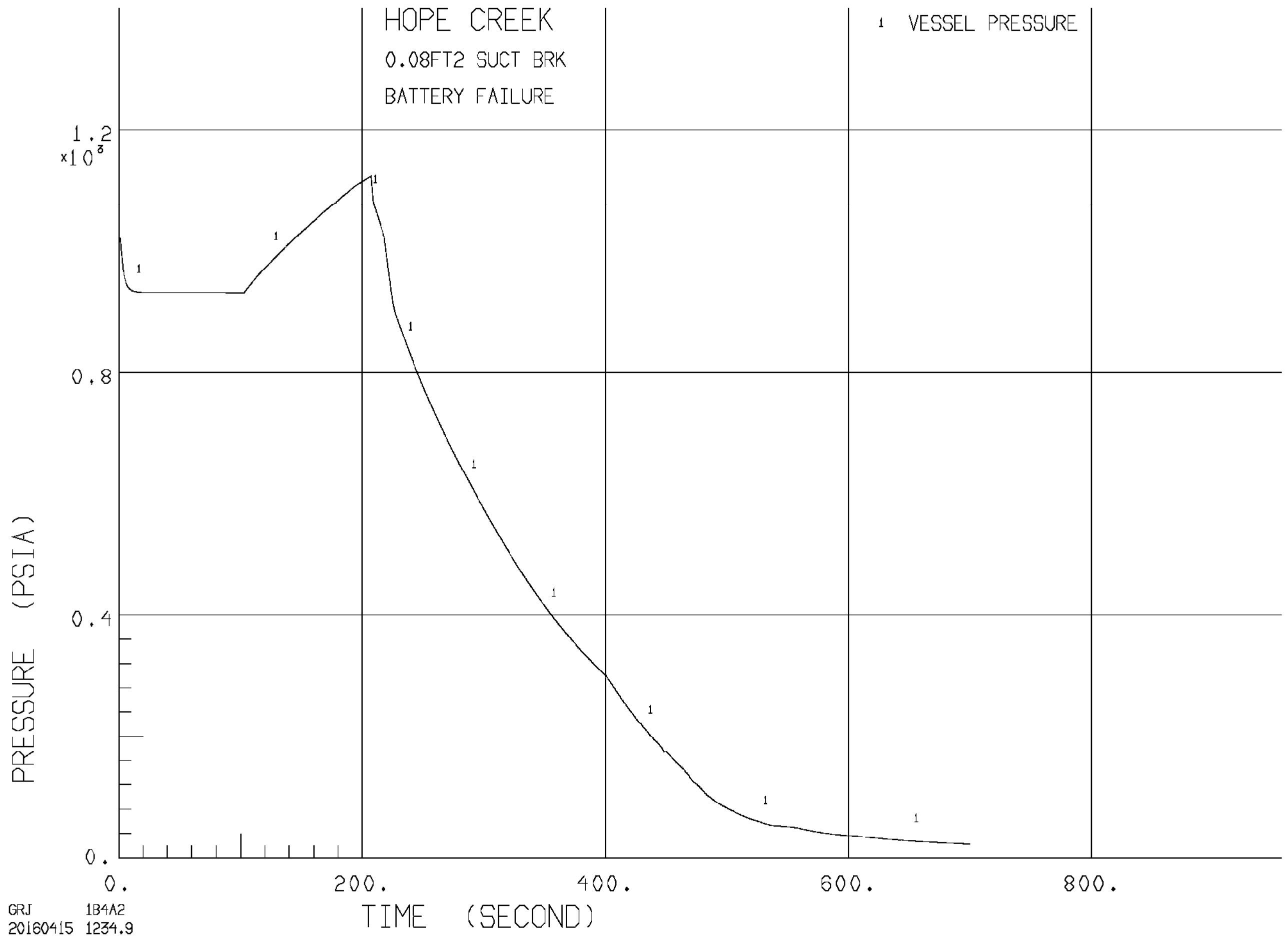

Figure 3-b Reactor Vessel Dome Pressure,
Limiting Small Recirculation Suction Line Break (0.08 ft²), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Nominal Assumptions

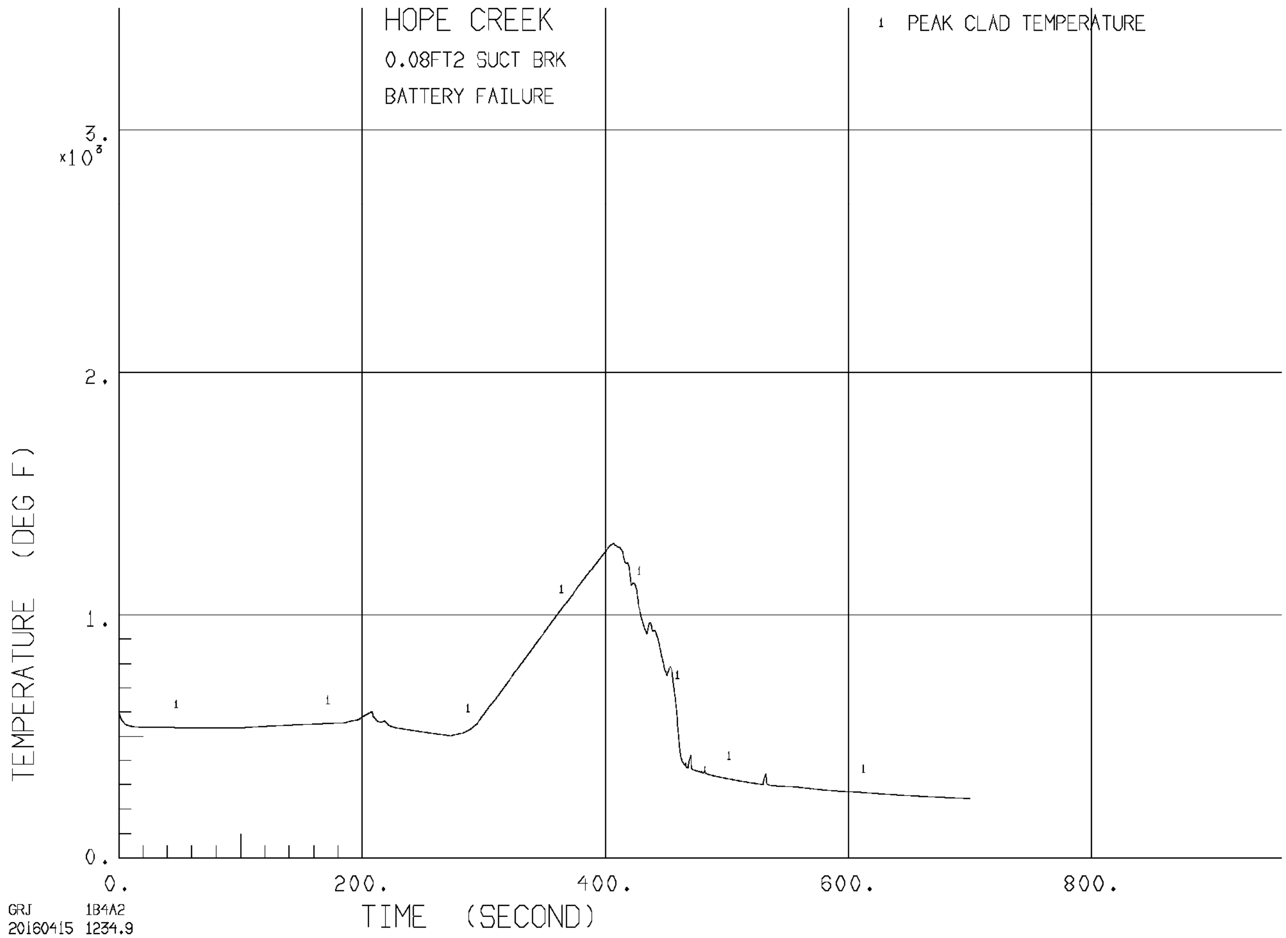

Figure 3-c Peak Cladding Temperature,
Limiting Small Recirculation Suction Line Break (0.08 ft²), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Nominal Assumptions

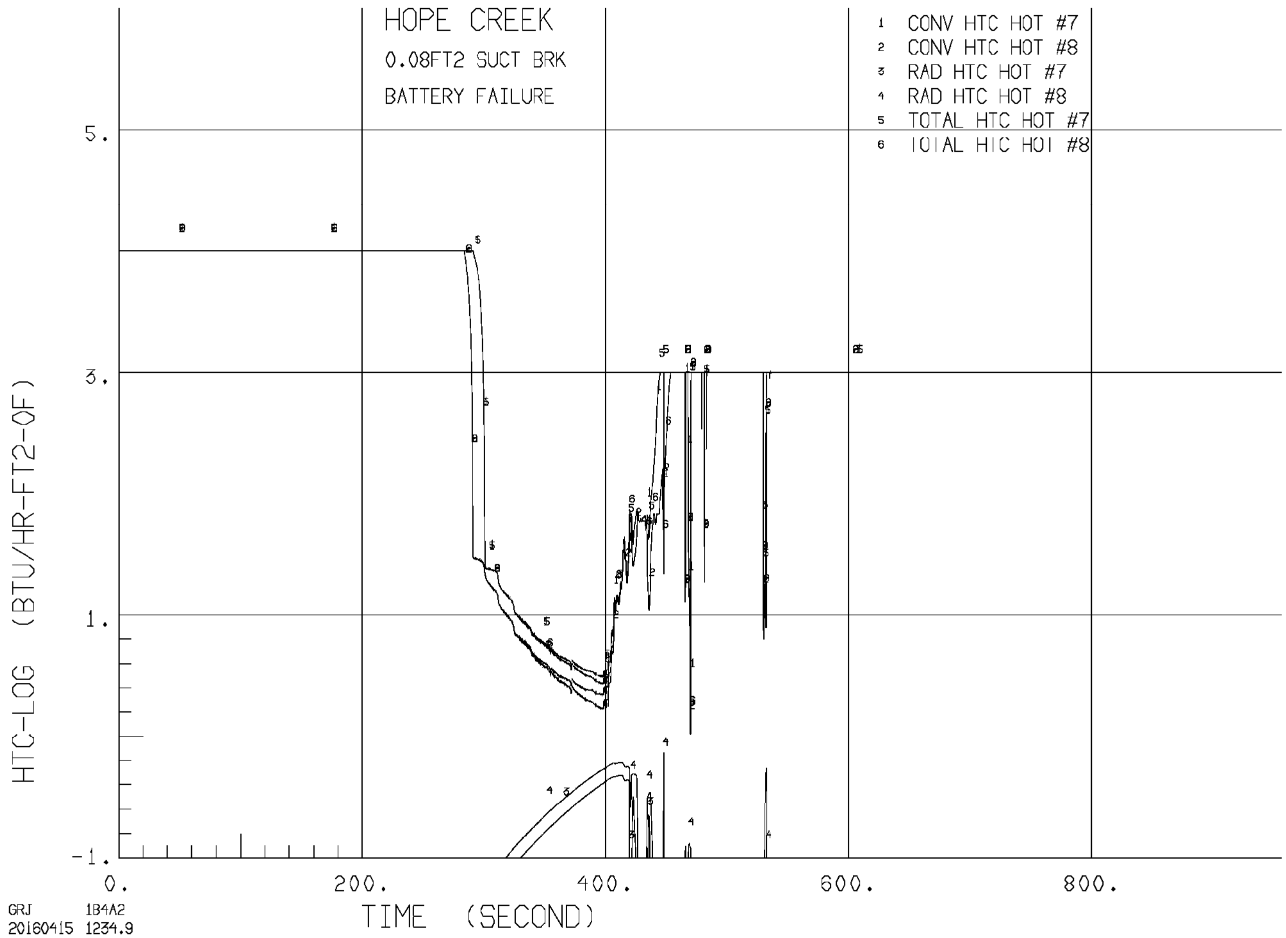

Figure 3-d Heat Transfer Coefficients,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Nominal Assumptions

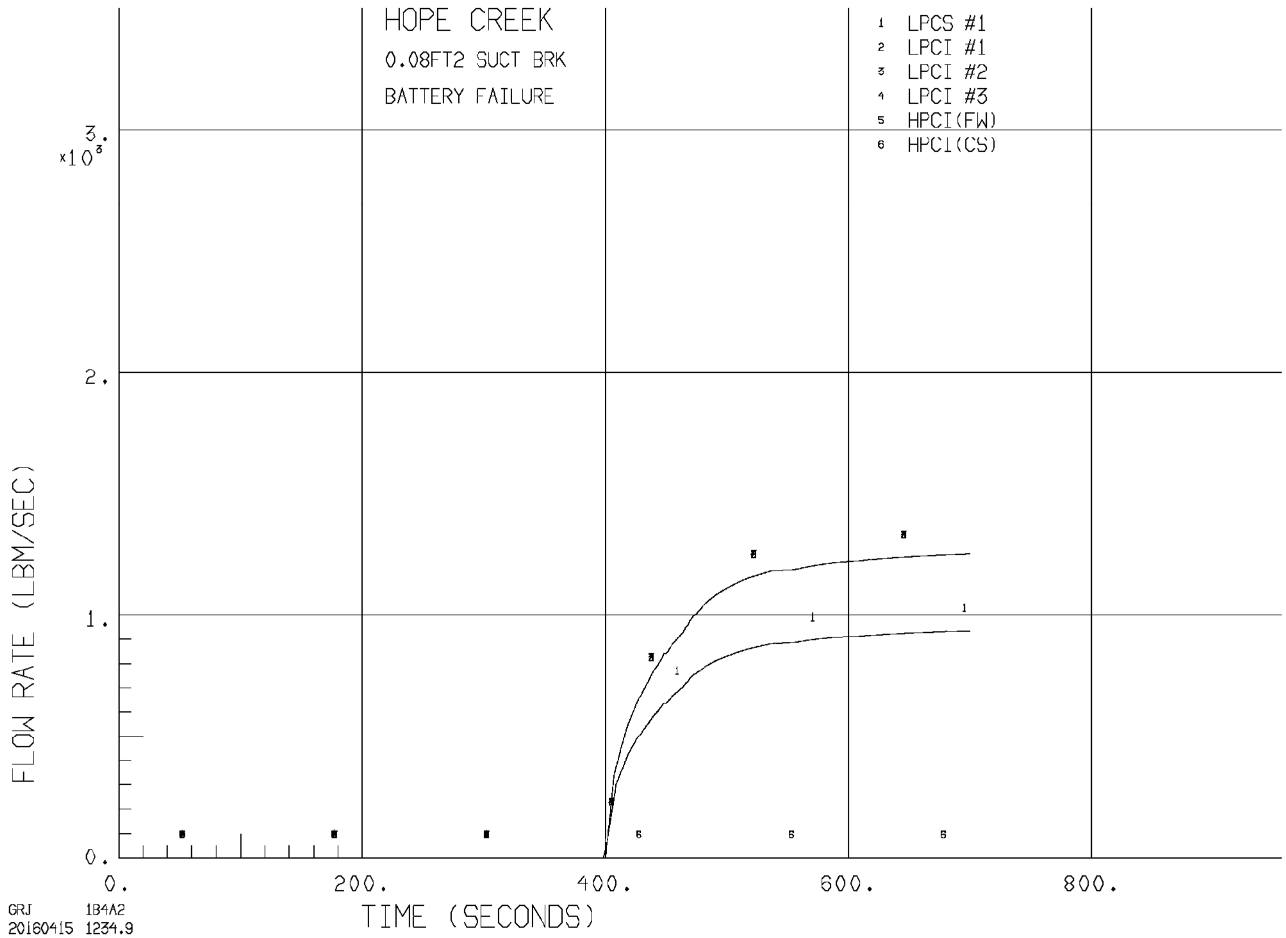

Figure 3-e ECCS Flow Rates,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Nominal Assumptions

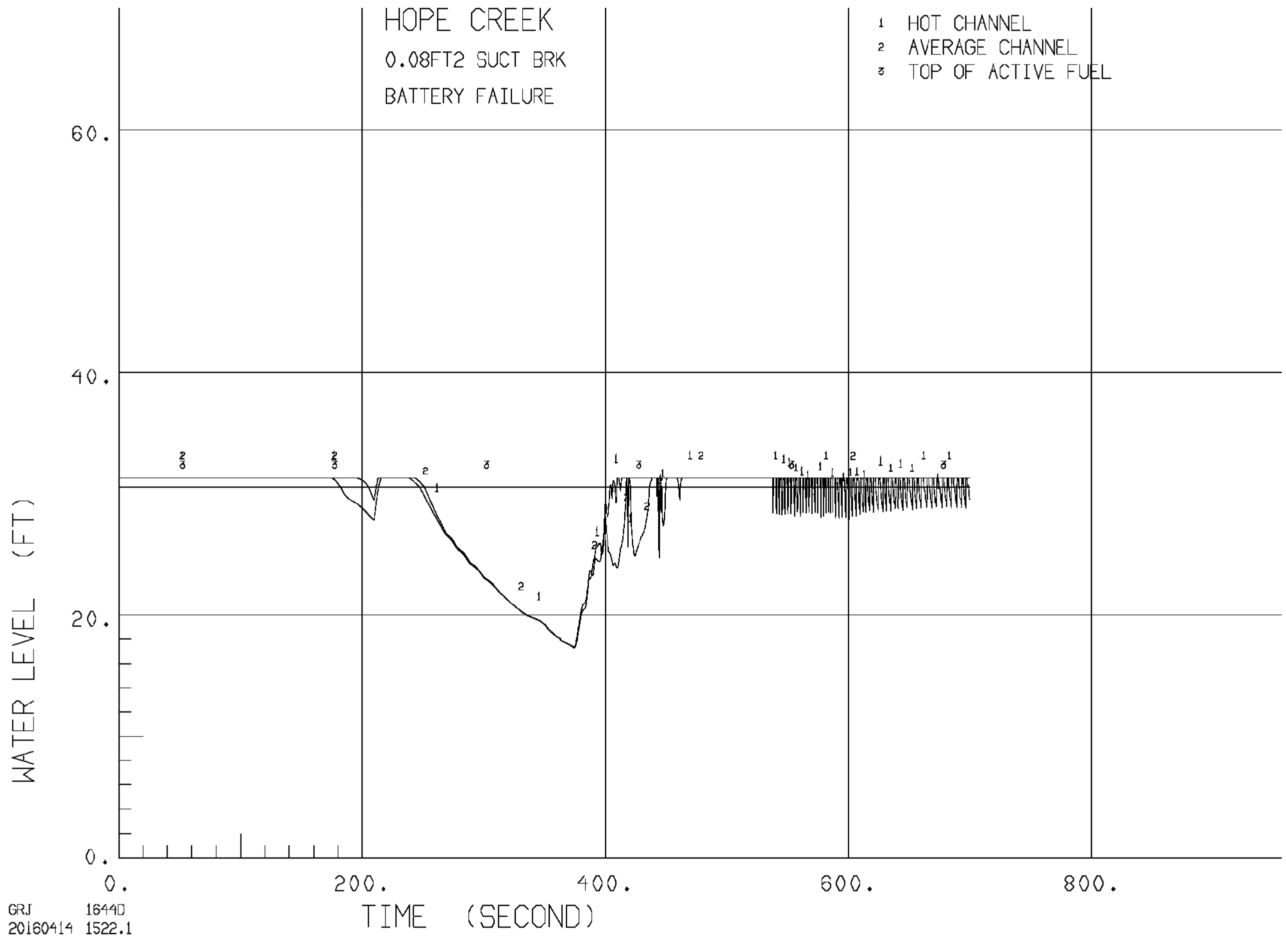

Figure 4-a Water Level in Hot and Average Channels,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Appendix K Assumptions

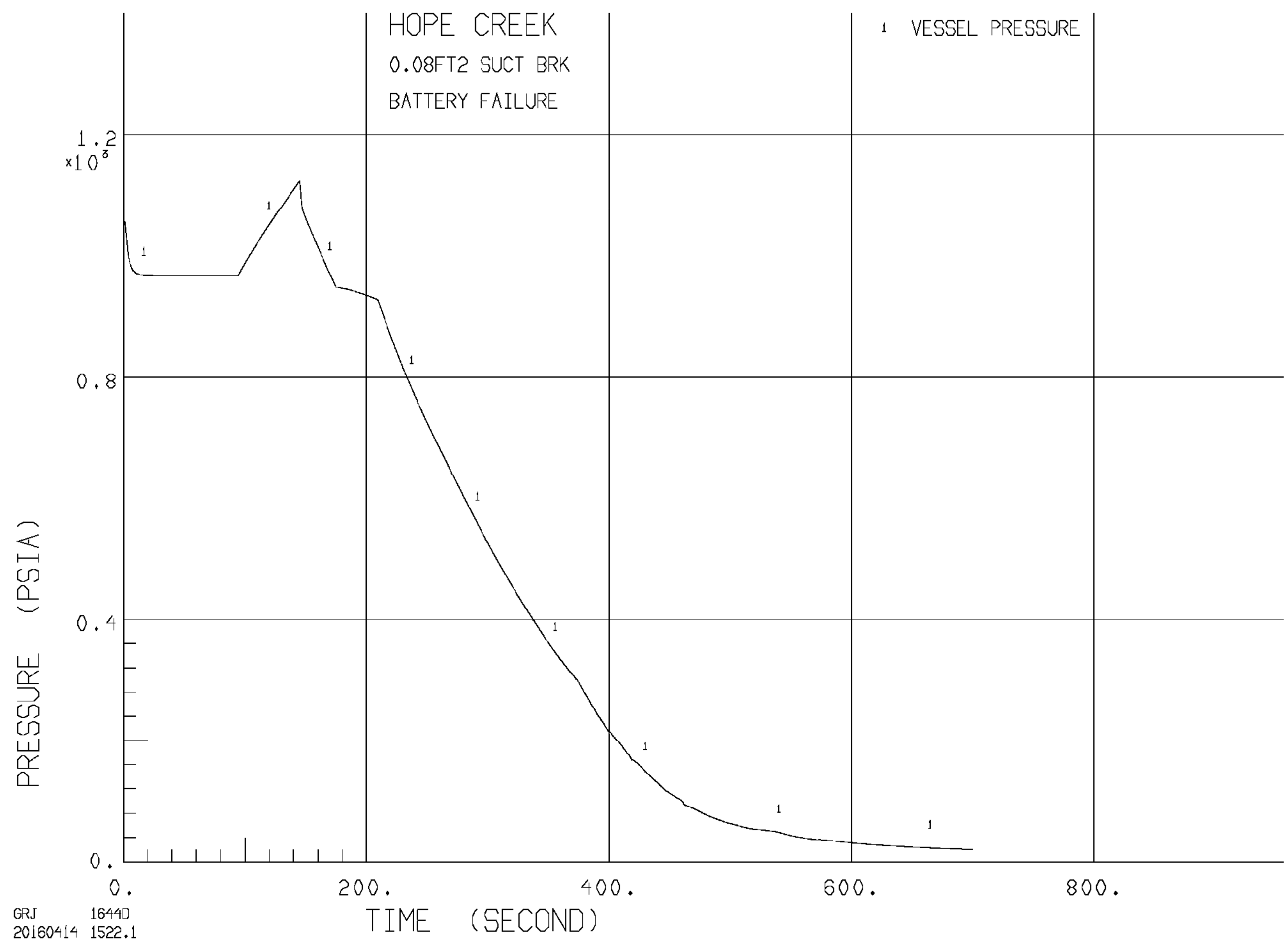

Figure 4-b Reactor Vessel Dome Pressure,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Appendix K Assumptions

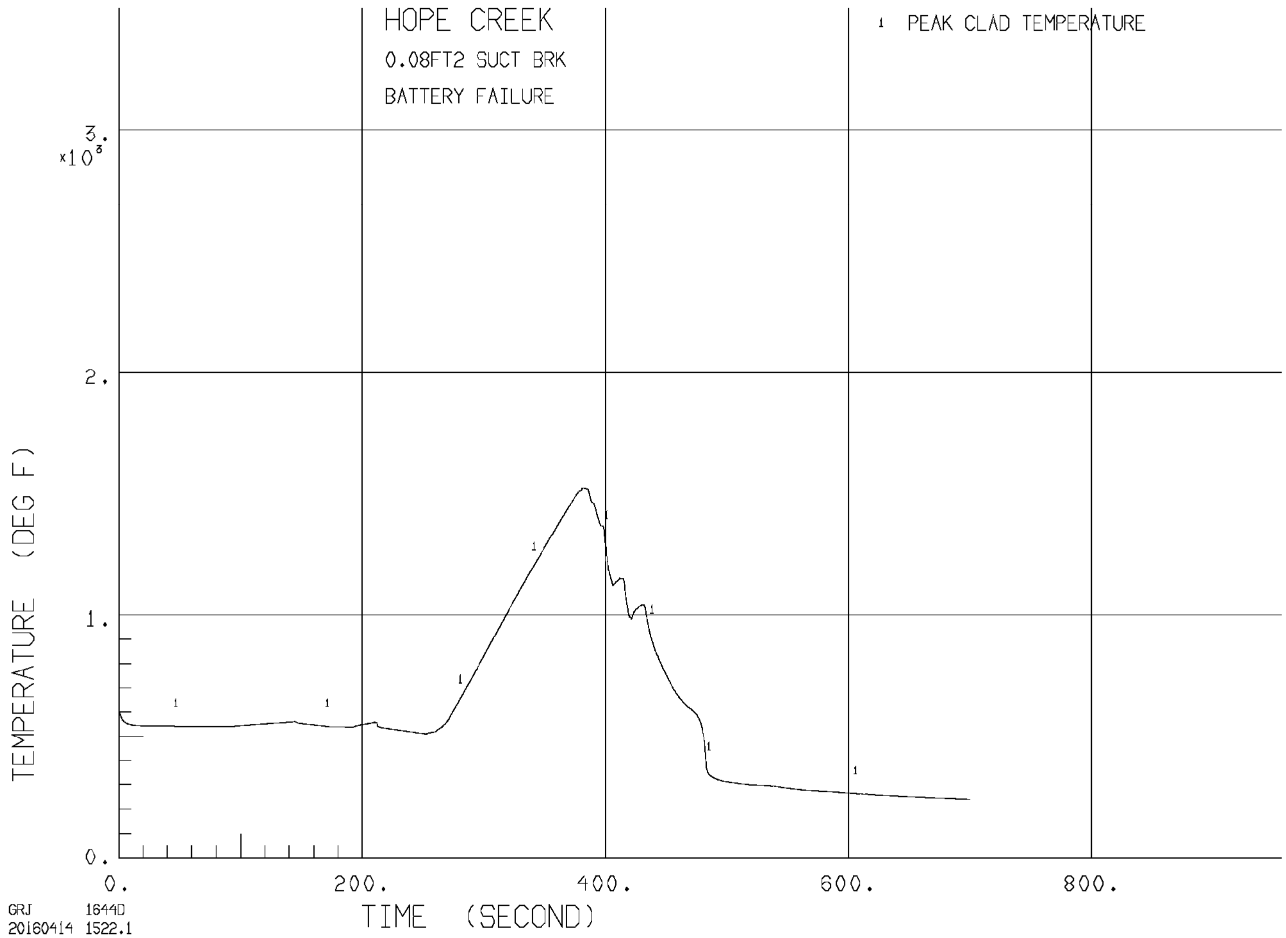

Figure 4-c Peak Cladding Temperature,
Limiting Small Recirculation Suction Line Break (0.08 ft²), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Appendix K Assumptions

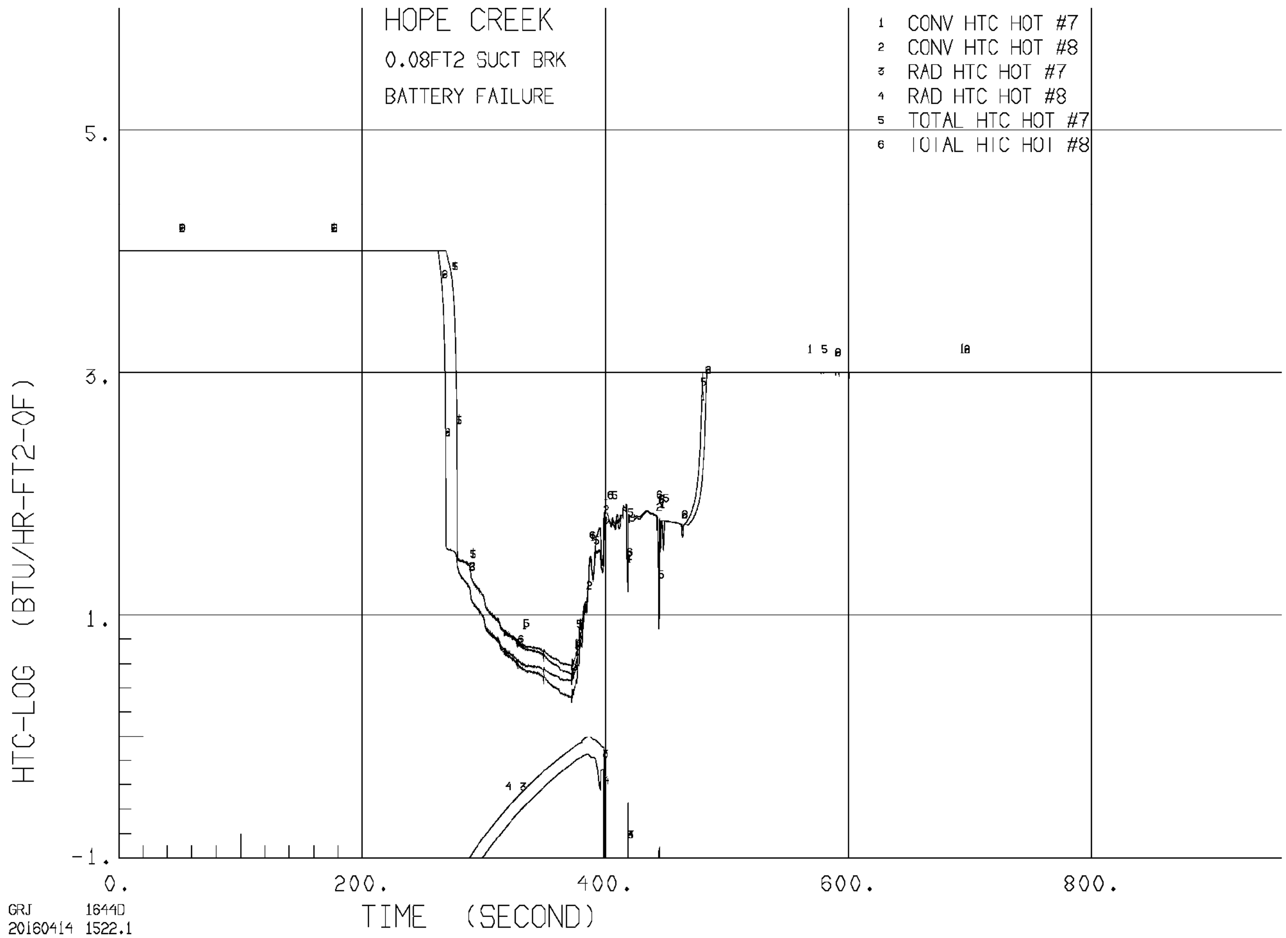

Figure 4-d Heat Transfer Coefficients,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Appendix K Assumptions

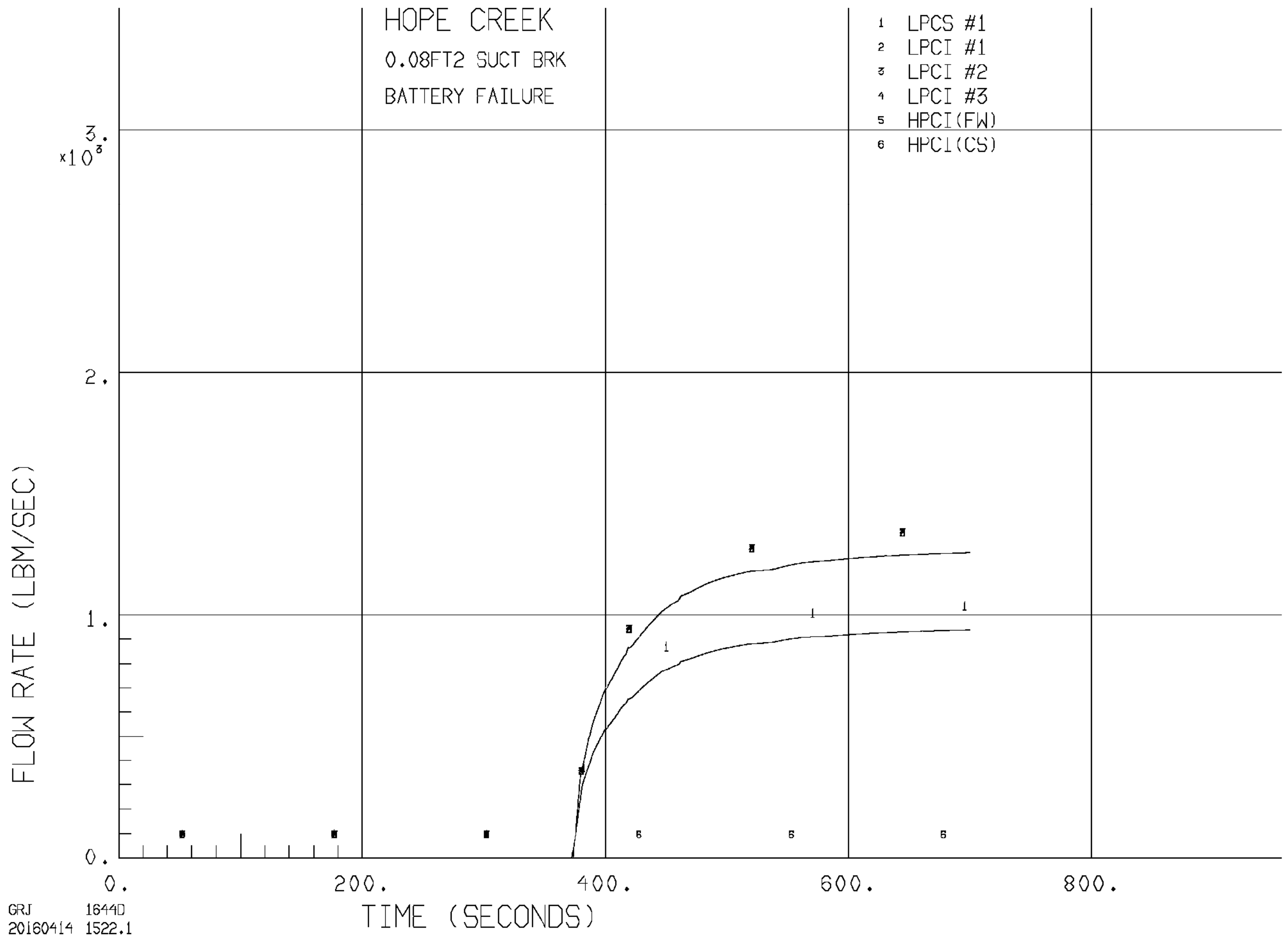

Figure 4-e ECCS Flow Rates,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), Battery Failure, GNF2 Fuel
1LPCS + 3LPCI + 5ADS Available, Appendix K Assumptions

GEH Affidavit supporting the withholding of information in Enclosure 3 from public disclosure

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Lisa K. Schichlein, state as follows:

(1) I am a Senior Project Manager, NPP/Services Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH proprietary report, "Hope Creek Generating Station GNF2 ECCS-LOCA Evaluation," 002N5176-R0-P, Revision 0, August 2016. GEH proprietary information in 002N5176-R0-P, Revision 0 is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. Sec. 552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my

knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary or confidentiality agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed information and results for the loss-of-coolant-accident analysis for GEH Boiling Water Reactors (BWRs) using GEH methodology. Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience and information databases that constitute a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

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

Executed on this 19th day of December 2017.

Lisa K. Schichlein

Lisa K. Schichlein
Senior Project Manager, NPP/Services Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Road
Wilmington, NC 28401
Lisa.Schichlein@ge.com