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RA-20-0170 10 CFR 50.90

June 15, 2020

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
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Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55

Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Breaks Outside of the Containment Building - Responses to Request for Additional Information

References:

1. Letter to the U. S. Nuclear Regulatory Commission from J. Ed Burchfield, Jr., Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Breaks Outside of the Containment Building," dated August 28, 2019 (ADAMS Accession No. ML19240A925).
2. Letter from the U. S. Nuclear Regulatory Commission to J. Ed Burchfield, Jr., Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Breaks Outside of the Containment Building," dated May 6, 2020. (ADAMS Accession No. ML20125A361).

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) submitted a License Amendment Request (LAR) which proposes to revise the ONS current licensing basis regarding the high energy line breaks (HELBs) outside of the containment building on August 28, 2019 (Reference 1).

The Nuclear Regulatory Commission (NRC) began an audit of various HELB documentation in February 2020. As a result of the audit, the NRC has determined that additional information is needed to support its review of the HELB LAR. The draft requests for additional information (RAIs) were discussed with the NRC on May 1, 2020, to support understanding the questions and information needed. The NRC issued the RAIs on May 6, 2020 (Reference 2).

Enclosure 1 provides a revised Significant Hazards Consideration that addresses the effects of the comprehensive analysis performed for HELB. This came as an NRC request to update the Significant Hazards Consideration to include statements to reflect that the analysis and results of over 3000 postulated break locations per unit represent an improvement in the safety of the plant due to the comprehensive nature of the analysis.

Enclosure 3 provides the responses to the RAIs (Reference 2) which includes information that is proprietary to Duke Energy. The proprietary information is identified by brackets and is

annotated as such. The annotated information has substantial commercial value that provides a competitive advantage. Enclosure 2 contains a non-proprietary [redacted] version of this content. Attachment 1 contains an Affidavit attesting to the proprietary nature of the information in Enclosure 3. Note that a miscellaneous RAI was added at the end of Enclosure 2 and Enclosure 3 to address the Significant Hazards Consideration revision described above.

No changes to Technical Specifications are proposed.

The responses to the RAIs specifically have been reviewed and determined to not affect the conclusions of the Significant Hazards Consideration provided in the LAR dated August 28, 2019 (Reference 1) or the revisions in Enclosure 1.

Inquiries on this proposed amendment request should be directed to Timothy D. Brown, ONS Regulatory Projects Group, at (864) 873-3952.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 15, 2020.

Sincerely,

J. Ed Burchfield, Jr.

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Enclosure 1:	Significant Hazards Consideration
Enclosure 2:	Responses to Requests for Additional Information [Non-Proprietary]
Enclosure 3:	Responses to Requests for Additional Information [Proprietary]
Attachment 1	Duke Energy Affidavit

cc w/enclosure and attachments:

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ENCLOSURE 1

SIGNIFICANT HAZARDS CONSIDERATION

Enclosure 1
Significant Hazards Consideration

Duke Energy has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Justification: A High Energy Line Break (HELB) does not constitute a previously evaluated accident. HELB is a design criterion that is required to be considered in the design of structures, systems, or components and is not a design basis accident or design basis event. The possibility of HELBs is appropriately considered in the UFSAR and Duke Energy has concluded that the proposed changes do not increase the possibility that a HELB will occur or increase the consequences from a HELB. This LAR provides an overview of HELB reanalysis, descriptions of station modifications that will be made as a result of the HELB reanalysis, and the proposed mitigation strategies which now include normal plant equipment, the protected service water (PSW) system, and the standby shutdown facility (SSF).

   The analysis that supports the HELB LAR is a comprehensive reevaluation of HELBs that could occur in the plant. The analysis evaluated over 3000 postulated break locations per unit. The evaluations showed that for each break, the capability to reach safe shutdown is available considering the postulation of a single active failure. The evaluation results determined the plant’s ability to safely mitigate HELBs that could occur and increase overall safety of the plant.

   The PSW and SSF Systems are designed as standby systems for use under emergency conditions. With the exception of testing, the systems are not normally pressurized. The duration of the test configuration is short as compared to the total plant (unit) operating time. Due to the combination of the infrequent testing and short duration of the test, pipe ruptures are not postulated or evaluated for these systems.

   Other systems have also been excluded based on the infrequency of those systems operating at high energy conditions. Consideration of HELBs is excluded (both breaks and cracks) if a high energy system operates for less than 1% of total unit operating time such as emergency feedwater or reactor building spray or if the operating time of a system at high energy conditions is less than approximately 2% of total system operating time such as low pressure injection. This is acceptable based on the very low probability of a HELB occurring during the limited operating time of these systems at high energy conditions. Gas and oil systems have been excluded, since these systems also possess limited energy.

   The modifications associated with the HELB licensing basis will be designed and installed in accordance with applicable quality standards to ensure that no new failure

mechanisms, malfunctions, or accident initiators not already considered in the design and licensing basis are introduced. For Turbine Building HELBs that could adversely affect equipment needed to stabilize and cooldown the units, the PSW System or SSF provides assurance that safe shutdown can be established and maintained. For Auxiliary Building HELBs, normal plant systems or the SSF provides assurance that safe shutdown can be established and maintained.

As noted in Section 3.4, Oconee Nuclear Station plans to adopt the provisions of Branch Technical Position (BTP) Mechanical Engineering Branch (MEB) 3-1 regarding the elimination of arbitrary intermediate breaks for analyzed lines that include seismic loading. Guidance in the BTP MEB 3-1 is used to define crack locations in analyzed lines that include seismic loading. Adoption of this provision allows Oconee Nuclear Station to focus attention to those high stress areas that have a higher potential for catastrophic pipe failure. In absence of additional guidance, Duke Energy uses NUREG/CR-2913 to define the zone of influence for breaks and critical cracks that meet the range of operating parameters listed in NUREG/CR-2913. NUREG/CR-2913 provides an analytical model for predicting two-phase, water jet loadings on axisymmetric targets that did not exist prior in the Giambusso/Schwencer requirements.

In conclusion, the changes proposed will increase assurance that safe shutdown can be achieved following a HELB. The changes will also collectively enhance the station's overall design, safety, and risk margin; therefore, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Justification: A HELB does not constitute a previously-evaluated accident. HELB is a design criterion that is required to be considered in the design of structures, systems, or components and is not a design basis accident or design basis event. The possibility of HELBs is appropriately considered in the UFSAR and Duke Energy has concluded that the proposed changes do not increase the possibility that a HELB will create a new or different kind of accident. This LAR provides an overview of HELB analysis, descriptions of station modifications that will be made as a result of the HELB reanalysis, and the proposed mitigation strategies which now include normal plant equipment, the PSW system, and the SSF.

   The analysis that supports the HELB LAR is a comprehensive reevaluation of HELBs that could occur in the plant. The analysis evaluated over 3000 postulated break locations per unit. The evaluations showed that for each break, the capability to reach safe shutdown is available considering the postulation of a single active failure. The evaluation results determined the plant's ability to safely mitigate HELBs that could occur and increases overall safety of the plant.

The modifications associated with the HELB licensing basis will be designed and installed in accordance with applicable quality standards to ensure that no new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing basis are introduced. For Turbine Building HELBs that could adversely affect equipment needed to stabilize and cooldown the units, the PSW System or SSF provides assurance that safe shutdown can be established and maintained. For Auxiliary Building HELBs, normal plant systems or the SSF provides assurance that safe shutdown can be established and maintained.

In conclusion, the changes proposed will increase assurance that safe shutdown can be achieved following a HELB. The changes will also collectively enhance the station's overall design, safety, and risk margin; therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Justification: A HELB does not constitute a previously-evaluated accident. HELB is a design criterion that is required to be considered in design of structures, systems, or components and is not a design basis accident or design basis event. The possibility of HELBs is appropriately considered in the UFSAR and Duke Energy has concluded that the proposed changes do not involve a reduction in the margin of safety. This LAR provides an overview of HELB analysis, descriptions of station modifications that will be made as a result of the HELB reanalysis, and the proposed mitigation strategies which now include normal plant equipment, the PSW system, and the SSF.

The analysis that supports the HELB LAR is a comprehensive reevaluation of HELBs that could occur in the plant. The analysis evaluated over 3000 postulated break locations per unit. The evaluations showed that for each break, the capability to reach safe shutdown is available considering the postulation of a single active failure. The evaluation results determined the plant's ability to safely mitigate HELBs that could occur and increases overall safety of the plant.

The modifications associated with the HELB licensing basis will be designed and installed in accordance with applicable quality standards to ensure that no new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing basis are introduced. For Turbine Building HELBs that could adversely affect equipment needed to stabilize and cooldown the units, the PSW System or SSF provides assurance that safe shutdown can be established and maintained. For Auxiliary Building HELBs, normal plant systems or the SSF provides assurance that safe shutdown can be established and maintained.

The changes described above provide a HELB licensing basis and increase overall plant safety margins. The changes have no effect on limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications. Therefore, the proposed change does not involve a reduction in the margin of safety.

Based on the above, Duke Energy concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed revision to the wording in the UFSAR and operation of the unit in the proposed manner, (2) the proposed revision will be implemented in a manner consistent with the commission’s regulations, and (3) the issuance of the amendment will not be adverse to the common defense and security or to the health and safety of the public.

ENCLOSURE 2

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

[NON-PROPRIETARY]

Enclosure 2

Responses to Requests for Additional Information [Non-Proprietary]

By letter dated August 28, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19240A925), Duke Energy Carolinas, LLC (Duke Energy) submitted a license amendment request (LAR) for Oconee Nuclear Station (ONS) Units 1, 2, and 3. The LAR proposes to revise the ONS current licensing basis for high energy line breaks (HELBs) outside of the containment building.

The Nuclear Regulatory Commission (NRC) has determined that additional information is needed as discussed in requests for additional information (RAIs) given below. Responses to the RAIs immediately follow the request.

The NRC also requested an update to the significant hazards consideration to include positive statements to reflect that the analysis and results of over 3000 postulated break locations per unit represent a positive improvement in the safety of the plant due to the comprehensive nature of the analysis. This is addressed as a miscellaneous response at the end of the RAIs.

Regulatory Basis from the RAI letter dated May 6, 2020 (ADAMS Accession No. ML20125A361):

ONS Updated Final Safety Analysis Report (UFSAR) Section 3.1.6, "Criterion 6 – Reactor Core Design (Category A)," states, "The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power."

UFSAR, Section 3.1.9, "Criterion 9 – Reactor Coolant Pressure Boundary (Category A)," states, "The reactor coolant pressure boundary shall be designed and constructed to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime."

Title 10 of the Code of Federal Regulations (CFR), Section 50.46(b)(4), "Coolable geometry," states, "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The RAI contains proprietary information as originally submitted in the August 28, 2019, LAR. Attachment 4 of the August 28, 2019, letter, is the proprietary (non-public) version of "Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," and Attachment 5 is the non-proprietary (public) version. Proprietary information is identified by text enclosed within double brackets. A non-proprietary version of the RAI is also enclosed.

RAI-1:

"Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," Section 1.1, third paragraph states:

> RELAP5/MOD2 is selected for these analyses based on the potential for sustained two-phase conditions in the RCS [reactor coolant system] piping. The MS [main steam] HELB analysis can result in sufficient overcooling that leads to two-phase conditions in the RCS piping which can potentially interrupt natural circulation. To accurately predict this phenomenon the RELAP5/MOD2 code was selected to perform this analysis. The FDW [Feedwater] HELB analyses can also result in sustained two-phase conditions indicating that RELAP5 based methods are more appropriate to perform the analysis.

> RETRAN based methods are selected for analyses where sustained two-phase conditions are not expected.

As stated above, two-phase conditions in the RCS are expected in both MS and FDW HELB overcooling and overheating analysis respectively. RELAP5/MOD2 is suitable for analysis in which the two-phase conditions are expected for accurately predicting the results, and RETRAN should be selected where two-phase conditions are not expected. For the thermal-hydraulic (T-H) analysis described in Section 3.0 and 4.0 of Attachment 6, specify the code used for each analysis. In case RETRAN was used for any of the analyses, justify that sustained two-phase condition in the RCS did not exist as this code is not suitable for predicting accurate results for this condition.

RAI-1 Response:

Criterion 6 describes evaluations with respect to expected conditions of normal operation and anticipated transients. However, HELBs are not considered expected conditions of normal operation or typical anticipated transient situations as described within Criterion 6. Note that as part of the HELB analyses, the reactor core is evaluated to ensure fuel damage limits are not exceeded.

Attachment 4 to the LAR, "Thermal Hydraulic Models for High Energy Line Break Transient Analyses [Proprietary]," provides additional information regarding the selection of T-H analysis programs used in the various HELB analyses. For certain overcooling cases described below, previous RETRAN-3D work is referenced which exists through previously submitted and approved licensing actions. For modeling consistency, RELAP5/MOD2-B&W is used for the new overheating and overcooling cases presented in this licensing action. It is noted that for those cases which utilized RETRAN-3D sustained two-phase conditions do not occur. These cases are described in more detail in the following sections of Attachment 4:

Section 2.1 "Overheating Analysis Description"

Section 2.1.1.1 "4160 VAC Power Available – TB FDW HELB." For this case, RELAP5/MOD2-B&W was used.

Section 2.1.1.2 "4160 VAC Power Available – EPR FDW HELB." For this case RELAP5/MOD2-B&W was used.

Section 2.1.2 "4160 VAC Power Unavailable – SSF Mitigation." For this case RELAP5/MOD2-B&W was used.

Section 2.1.3 "4160 VAC Power Unavailable – PSW Mitigation." For this case RELAP5/MOD2-B&W was used.

These above four sections correspond to Attachment 6 sections 3.1.1, 3.1.3, 3.2 and 3.3, respectively. The section 3.1.2 analyses are not explicitly modeled because they are considered to be bounded by the analyses described in section 3.1.1.

Section 2.2 "Overcooling Analysis Description"

Section 2.2.1.1 "4160 VAC Power Available – Single MS HELB." For this case, the existing UFSAR Chapter 15 analysis was adopted as a surrogate which used RETRAN-3D. This analysis and the methods used in this analysis have been previously approved by the NRC.

Section 2.2.1.2 "4160 VAC Power Available – Double MS HELB." For this case, the existing UFSAR Chapter 15 methods using RETRAN-3D was modified by adding a second break location. The double Main Steam Line Break (MSLB) HELB analyses described in Attachment 6

sections 4.1.2 and 4.1.3 have previously been reviewed and approved as part of the Protected Service Water (PSW) Safety Evaluation Report (SER) (ADAMS Accession No. ML14206A790) (Reference 3) and are described in Section 3.3.1 of the PSW SER. This information was previously submitted in response to RAI 153 (ADAMS Accession No. ML12195A325) related to the PSW LAR (Reference 6).

These above two sections correspond to Attachment 6 sections 4.1.1, 4.1.2, and 4.1.3, respectively.

Section 2.2.2 "4160 VAC Power Unavailable – SSF Mitigation." For this case RELAP5/MOD2-B&W was used.

Section 2.2.3 "4160 VAC Power Unavailable – PSW Mitigation." For this case RELAP5/MOD2-B&W was used.

The above two sections correspond to Attachment 6, sections 4.2 and 4.3 respectively.

RAI-2:

"Thermal-Hydraulic Models for High Energy Line Break Transient Analysis," section 3.1 "Ambient Heat Losses", first paragraph, last sentence states:

> **[[**
>
> **]]**

Describe the **[[**

]]

RAI-2 Response:

The Standby Shutdown Facility (SSF) powers a subset of the pressurizer heaters. The supporting SSF analyses assume the SSF powered pressurizer heaters provide a minimum of 50 kilowatt (kW) above the ambient heat losses from the pressurizer which based on the number of powered heaters establishes a 175 kW upper limit to the ambient losses from the pressurizer. Plant testing validates this design input to verify that pressurizer ambient heat losses are less than the SSF requirements as document by Technical Specification 3.10.1. The results of this testing are provided in **[[** **]]**$_D^{a,c}$ for the overheating analysis. The heat transfer coefficients used in the overheating and overcooling RELAP5 analyses are defined to obtain the specified initial ambient heat loss.
Pressurizer ambient heat losses are determined during steady-state Mode 1 RCS conditions using measured values of voltage, current, and power for all operating pressurizer heater groups. From this data, a total heater input value (kW) is determined which is indicative of the power required to 1.) bring pressurizer spray bypass flow to saturation conditions and 2.) overcome ambient heat losses to the reactor building. Using available information regarding pressurizer spray bypass flow, a quantification of pressurizer ambient heat losses is determined. **[[**

]] $_D^{a,c}$ and provides additional details on how ambient heat losses are determined.

Ambient heat losses from the pressurizer are modeled in the overheating (after the peak RCS pressure occurs) and overcooling analyses. Ambient heat losses from the other RCS structures are not modeled in the overheating and overcooling analyses .

RAI-3:

“Thermal-Hydraulic Models for High Energy Line Break Transient Analysis,” section 3.2 describes reactor vessel (RV) head axial conduction modification in the RELAP5/MOD2-B&W ONS base model described in DPC-NE-3003-PA, Revision 1. This modification is not described in any of the subsections under section 2.1 of Attachment 5. Provide responses to the following:

(a) Describe the **[[** **]]**, and what each one represents.

(b) Since this modification is not included in those listed in any of the Subsections under section 2.1 of Attachment 5, confirm this modification was performed in the RELAP5/MOD2-B&W ONS base model for the overheating analysis.

(c) Since this feature was not included in the RELAP5/MOD2-B&W ONS base model, justify the **[[** **]]** for overheating analysis.

(d) The last sentence in Attachment 5, section 3.2 states:

> “The RV upper head includes numerous axial structures, a portion of which are modeled to allow heat transfer across node boundaries.”

Explain why there is a need for additional heat structure even though the existing ones allow heat transfer across node boundaries.

RAI-3 Response:

The ONS RELAP5/MOD2-B&W base model described in Duke Energy methodology report DPC-NE-3003-PA (Reference 10) does not contain heat structures to represent the internal metal structures within the RV upper head nodes. **[[**

]] $_{D}^{a,c}$ These structures are defined in the overcooling analysis to mitigate non-physical behavior attributed to the nodalization. The additional heat structures mitigate thermal stratification in a voided RV head, which can develop due to the absence of axial heat conduction in the base model. This modeling feature is not used in the overheating analysis.

The last sentence of Attachment 4 section 3.2 is referring to the large number of structures physically installed in the ONS RVs, that would conduct heat axially. A portion of the actual physical components are represented by the additional heat structures. By incorporating this additional modeling feature in the overcooling analyses, the previously mentioned non-physical behavior is successfully mitigated.

RAI-4:

“Thermal-Hydraulic Models for High Energy Line Break Transient Analysis,” sections 3.7 and 5.0, first paragraph in both sections state:

> The modeling approach for several of these features considers the impact of asymmetric loop conditions on the performance of the individual boundary condition.

Explain what is meant by "impact of asymmetric loop conditions on the performance of individual boundary conditions". Provide examples of asymmetric loop conditions and the boundary conditions that are impacted.

RAI-4 Response:

The discussion in Attachment 4 Section 3.7 is intended to refer to boundary conditions that are physically connected to both steam generators (SGs). These boundary conditions need to consider that the SGs pressures will be different and should be constructed accordingly. For example, a pump that is connected to the secondary side of both SGs will deliver flow based on the pump discharge pressure, and that value needs to be determined as part of the development of the boundary condition. The flow delivered to the secondary side of each SG should be defined based on the line losses from the pump to each SG, and the pressure within each SG. This example applies to the turbine driven emergency feedwater (EFW) pump and the SSF auxiliary service water (ASW) pump modeling in the single MS HELB cases.

RAI-5:

Attachment 6, for scenarios analyzed in sections 3.1.1 and 3.1.2 in which the minimum nucleate boiling ratio (DNBR) is required to meet the acceptable fuel design limits, provide the following:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values of the parameter showing that the minimum departure from DNBR meets the specified acceptable fuel design limits.

RAI-5 Response:

RAI-5 requests information about DNBR analyses for overheating events. Overheating events, such as those described in sections 3.1.1 and 3.1.2, include a pressure and temperature increase in the RCS. The pressure increase dominates and increases the available subcooling in the core to suppress any potential departure from nucleate boiling. Increasing temperatures also gain a benefit from the negative moderator temperature coefficient that decreases core power. For an event that includes a reactor trip on high RCS pressure due to a loss of main FDW flow, the minimum DNBR is bounded by the Uncontrolled Bank Withdrawal analysis statepoint included in the normal ONS reload design process. Therefore, minimum DNBR is not limiting for overheating events.

RAI-6:

Attachment 6, for scenarios analyzed in sections 3.1.1, 3.1.2, 3.1.3, 3.2, and 3.3, provide the following for the RCS pressure analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values of the RCS pressure developed comparing it with 110% of design pressure.

(d) Please provide the following figures in the response:

1. **[[**
2.
3. **]]**

(e) Confirm that the **[[**
]] are conservative. These tables are not necessary to be added in the response.

RAI-6 Response:

Sections 3.2 and 3.3 describe the overheating analyses mitigated using the SSF or PSW equipment. These analyses are **[[** **]]** D[a,c] (Reference 2). The break area is conservatively assumed to be sufficient to cause an immediate termination of main FDW flow to the SGs. The limiting peak RCS pressure is obtained by **[[**

]] D[a,c] The maximum pressure occurs during the initial pressurizer safety relief valve cycle. The acceptance criteria is assured through the conservative assumption that the reactor does not trip coincident with a loss of 4160 VAC power.

[[

]] D[a,c]

[[

]] D[a,c]

Operator actions do not occur prior to reaching the peak RCS pressure, and do not affect satisfying the maximum RCS pressure acceptance criterion. However, based on the use of SSF or PSW equipment, operator action is assumed in sections 3.2 and 3.3 to perform the following actions; 1) trip the operating reactor coolant pumps (RCPs) either 2 minutes after a loss of indicated subcooled margin, or 3 minutes after loss of RCP seal cooling, 2) align either SSF-ASW or PSW to establish SG cooling at 14 minutes, 3) establish RCP seal cooling and RCS makeup from either an SSF reactor coolant makeup (RCMU) pump or one High Pressure Injection (HPI) pump at 20 minutes, 4) energize pressurizer heaters from the SSF standby power supply or PSW switchgear for RCS pressure control at 20 minutes or 2 hours, respectively, and 5) throttle SSF ASW or PSW to maintain an RCS pressure band until SG level develops, then maintain SG level band afterwards. All of these operator actions, with the exception of aligning PSW to establish SG cooling at 14 minutes, have been previously accepted by the NRC. The NRC previously accepted the alignment of PSW to the SGs within 15 minutes as part of the PSW SER (Reference 3). The HELB LAR seeks NRC approval to change the PSW alignment time from 15 minutes to 14 minutes. This action is taken from the Main Control Room. New Time Critical Operator Actions and justifications for these actions are provided in Attachment 12 of the LAR.

Section 3.1.3 describes the overheating analyses due to a FDW line break downstream of the check valves in the East Penetration Room (EPR). The break area downstream of the check valve in the EPR is limited to 0.54 ft^2 by a guard pipe. These analyses are **[[**

]] $_{D}^{a,c}$ Peak RCS pressure is not the primary concern for these cases as the limited break area allows a fraction of the initial FDW flow to reach the SGs. The PSVs do not lift during the initial transient response. The goal of the **[[** **]]** $_{D}^{a,c}$ is to demonstrate the ability to mitigate the transient without challenging a PSV while the pressurizer is water-solid, a longer term concern. **[[**

]] $_{D}^{a,c}$ demonstrates that pressurizer spray is not required with the pressurizer power operated relief valve (PORV) unavailable. The initial conditions assumed in **[[**

]] $_{D}^{a,c}$

Operator action is assumed in section 3.1.3 to perform the following actions; 1) reduce the number of operating RCPs to a 1/1 configuration at 15 minutes, 2) open one loop high point vent (HPV) to control RCS pressure 2150-2350 psig after HPVs become available at 30 minutes, 3) energize pressurizer heaters after they become available at 50 minutes, and 4) secure pressurizer spray at 80 minutes if available. These actions do not affect the peak RCS pressure reached during this case. Approval of Time Critical Operator Actions 1 and 2 above are being sought in the HELB LAR. These new actions are described and justified in Attachment 12 of the HELB LAR. The other actions described above have been previously accepted by the NRC.

The section 3.1.2 discussion covers the critical cracks in the FDW piping upstream of the check valves in the EPR. This discussion is bounded by the section 3.1.1 analysis results.

Section 3.1.1 describes the overheating analyses due to a feedwater line break (FWLB) in the Turbine Building (TB). These analyses are **[[**

]] $_{D}^{a,c}$ provides the basis for the section 3.1.1 discussion. Peak RCS pressure is not the primary concern for this case. The goal of this case is to demonstrate the ability to mitigate the transient without challenging a PSV while the pressurizer is water-solid, a longer term concern. This case assumes a double ended guillotine break of a main FDW line in

the TB. The RCS pressure increase following the FWLB does cause a reactor trip on high pressure but does not reach the PSV lift setpoint. A PSV lift is not predicted until approximately the time EFW actuates on low SG water level. The initial conditions assumed in **[[**

]] D[a,c]

Operator action is assumed in section 3.1.1 to perform the following actions; 1) reduce the number of operating RCPs to a 1/1 configuration at 15 minutes, and 2) open one loop HPV to control RCS pressure 2150-2350 psig after HPVs become available at 30 minutes. Pressurizer spray and heaters are assumed to be unavailable in this analysis. These actions do not affect the peak RCS pressure reached during this case. Approval of Time Critical Operator Actions described above are being sought in the HELB LAR. These actions are described and justified in Attachment 12 of the HELB LAR.

The requested figures are included after the response to RAI-13.

RAI-7:

Attachment 6, for scenarios analyzed in sections 3.1.1, 3.1.2, 3.1.3, 3.2, and 3.3, provide the following for the core coolable geometry analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs of the parameters involved and their numerical limiting values showing the core remains intact and the coolable geometry is maintained.

(d) During a clarification call with the licensee it was stated: The collapsed liquid level above the core and RVLIS [reactor vessel level indicating system] indications are tracked as a function of time to validate the core remains covered for overcooling events. Please provide the following.

    1. From the above statement it appears that the reactor level is tracked during the event. If so, how, where, and according to which procedure?
    2. Explain how the analysis validates that the core remains covered during the event. Provide analysis results of the reactor level showing that the core remains covered during overcooling events.

RAI-7 Response:

RAI-7 requests information demonstrating that the core remains intact and coolable geometry is maintained. There are three criteria that demonstrate coolable geometry. They are as follows:

1. DNBR remains within established limits.
2. The RCS pressure boundary remains in intact, including meeting RV acceptable pressure and temperature limits.
3. Maintaining an adequate water level above the core.

Attachment 6 of the LAR states that the analyses referenced in the LAR demonstrate that DNBR remains within acceptable limits. For additional information regarding DNBR, see the responses to RAIs 5 and 8 respectively. The second criterion is demonstrated by satisfying pressure limits of the RCS, including the RV. The response to RAI 5 above indicates for the

bounding overheating event that the pressure remains within 110% of the design pressure of the RCS and thus acceptable. Note that overcooling events do not produce pressure increases in the RCS. Overheating events do not challenge the pressure and temperature limits for the RV. Overcooling events do challenge the pressure and temperature limits for the RV. The response to RAI 9 below addresses pressure and temperature for the RV during overcooling events. The response concludes that pressure and temperature limits are satisfied. The remaining criteria is maintaining an acceptable water level above the core. Overheating events do not uncover the core as primary coolant expands during the transient. Pressurizer level and RCS subcooling validate the core remains covered for overheating events. Overcooling events present a potential challenge for core uncovery due to coolant contraction. The collapsed liquid level above the core and the RV head RVLIS indication are calculated and plotted as a function of time to validate the core remains covered for overcooling events. Thus, the three criteria for maintaining a coolable geometry are satisfied.

The ONS inadequate core cooling monitor system provides three level indications, one for the RV and one for each hot leg. These indications are included in the ONS RELAP5/MOD2-B&W base model and are used to evaluate liquid levels in the RCS. These levels are based on the pressure differential between two level taps. The RV head level is determined between the top of the RV head, and the bottom of the hot leg. The hot leg levels are determined between the top of the hot leg U-bend and the bottom of the hot leg. All of these instrument spans are located above the core. The RV head level is affected by RCP operation, and while it does indicate 0 inches is not considered to be a valid indication while the RCPs are operating. Following RCP trip, the indication becomes valid, and measures the liquid level in the RV head. A full hot leg indication is roughly 597 inches, and a full RV head indication is about 171 inches. Additionally, the collapsed liquid level above the core is a diverse indication composed of a summation of the liquid fraction multiplied by the node height, for the RV nodes located above the core. These indications, plotted as a function of time, validate the core remains covered for overcooling events. These indications are used to evaluate the RELAP5/MOD2-B&W calculated transient analysis results to demonstrate that the design of the RCS is such that sufficient water level remains above the core. While RVLIS indications are located in the main control room, this discussion is not intended to imply these indications are monitored or used to take action by the operator as part of the mitigation strategy.

RAI-8:

Attachment 6, for scenarios analyzed in Sections 4.1.1, 4.1.2, 4.1.3, 4.2.1, 4.2.2, 4.3.1, and 4.3.2 in which the minimum DNBR is required to meet the acceptable fuel design limits, provide the following:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values of the parameter showing that the minimum DNBR meets the specified acceptable fuel design limits.

(d) Please provide following figures:

1. **[[**
2.

3.
4.]]

RAI-8 Response:

Section 4.1.1 describes the MS HELBs that do not impact the status of 4160 VAC power. These MS HELBs were not specifically analyzed for this LAR. Existing MSLB analyses as described in UFSAR Chapter 15.13, and UFSAR Chapter 15.17 similarly assume that 4160 VAC power is not affected by the break. Although MS HELBs are different and distinct from MSLBs described in Chapter 15 of the UFSAR, in this case the Chapter 15 analyses were used as a surrogate. These Chapter 15 analyses were performed with RETRAN-3D using NRC approved methods described in the Duke Energy methodology reports DPC-NE-3000-PA (Reference 4) and DPC-NE-3005-PA (Reference 5). The initial conditions for the UFSAR Chapter 15.13 without offsite power and UFSAR Chapter 15.17 analyses performed to evaluate the limiting minimum DNBR are defined by the statistical core design (SCD) methodology. This method specifies that parameters important to the DNBR are set to nominal values and incorporates the associated uncertainty in the DNBR evaluation. For UFSAR Chapter 15.13 with offsite power cases that determine whether a return to criticality condition occurs, and to determine the minimum DNBR, the initial conditions are selected to maximize the RCS depressurization and cooldown, to maximize any potential return-to-power.

The limiting minimum DNBR state points from these analyses are included in the normal ONS reload core design process. This process ensures the minimum DNBR meets the specified acceptable fuel design limits.

Section 4.1.2 describes double MS HELBs where both 4160 VAC and 6900 VAC power remain available. Section 4.1.3 describes scenarios where a LOOP is assumed coincident with the double MS HELBs, and 4160 VAC power remains available. These scenarios are analyzed using NRC approved methods described in the Duke Energy methodology reports DPC-NE-3000-PA and DPC-NE-3005-PA modified by the addition of a second break location. These cases are analyzed with the initiating event being a double-ended break of both steam lines in the TB. The objective is to determine whether a return to criticality condition occurs, and to determine the minimum DNBR. The initial conditions for the section 4.1.2 scenarios are selected to maximize the RCS depressurization and cooldown, and to maximize any potential return-to-power. The initial conditions for the section 4.1.3 analyses performed to evaluate the limiting minimum DNBR are defined by the SCD methodology. This method specifies that parameters important to the DNBR are set to nominal values and incorporates the associated uncertainty in the DNBR evaluation. Limiting single failures are considered. The operator actions assumed are 1) to trip the operating RCPs 2 minutes after a loss of indicated subcooled margin, and 2) isolate EFW flow to the affected SG at 10 minutes. These operator actions have been previously accepted by the NRC. The limiting minimum DNBR state points from these scenarios are included in the normal ONS reload core design process. This process ensures the minimum DNBR meets the specified acceptable fuel design limits.

The double MSLB HELB analyses described in Attachment 6 sections 4.1.2 and 4.1.3 have previously been reviewed and approved as part of the PSW SER (ADAMS Accession No. ML14206A790) (Reference 3) and are described in section 3.3.1 of the PSW SER. This information was previously submitted in response to RAI 153 (ADAMS Accession No. ML12195A325) related to the PSW LAR (Reference 6).

Section 4.2.1 describes the plant transient response to a single MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using SSF equipment. **[[**

]] D[a,c] (Reference 7).

The following operator actions are credited in section 4.2.1. These operator actions do not impact the DNBR evaluation. Operator action is credited to secure the RCPs 3 minutes after loss of RCP seal cooling or 2 minutes after a loss of indicated subcooling. These two considerations occur at approximately the same time for this case. The operator actions to initiate SSF operation are described in section 3.6 of the LAR. The operator starts the SSF diesel generator, aligns the SSF electrical system, starts the SSF ASW pump, and starts the SSF RCMU pump. The SSF RCMU pump is credited to restore RCP seal cooling at 20 minutes. Operator action to isolate normal letdown is also credited at 20 minutes. The SSF powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers sufficiently **[[**

]] D[a,c]

The minimum DNBR for the initial depressurization immediately following reactor trip is covered by the Chapter 15 MSLB DNBR analyses. The limiting core response is obtained with a single MS HELB with an immediate RCP trip. This case is evaluated further by a sensitivity case that does not credit boron added by the SSF RCMU pump to obtain a bounding return to power response **[[** **]]** D[a,c]. The maximum core power level reached in this sensitivity case is 2.4% power at 1501 seconds. The minimum DNBR during this return to power is evaluated using the EPRI critical heat flux (CHF) correlation to identify the limiting critical heat flux and DNBR state points. The Modified Barnett CHF correlation is then used to evaluate the limiting state points identified with the EPRI correlation and the peak heat flux state point. **[[**

]] D[a,c], with a minimum value of 5.45 at the time of the return to power. This DNBR result is consistent with the indicated core exit subcooling between 1200 and 1800 seconds of Case 2k being greater than 120°F, and consistently greater than 60°F subcooled while the core is re-critical. A reload design check has been incorporated into the current ONS reload design process for the single MS HELB scenario.

Section 4.2.2 describes the plant transient response to a double MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using SSF equipment. **[[**

]] D[a,c]

The following operator actions are credited in section 4.2.2. These operator actions do not impact the DNBR evaluation. Operator action is credited to secure the RCPs 3 minutes after loss of RCP seal cooling. The operator actions to initiate SSF operation are described in section 3.6 of the LAR. The operator starts the SSF diesel generator, aligns the SSF electrical system, starts the SSF ASW pump, and starts the SSF RCMU pump. The SSF RCMU pump is credited to restore RCP seal cooling at 20 minutes. Operator action to isolate normal letdown is also credited at 20 minutes. The SSF powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers sufficiently at **[[**

]] $_{D}^{a,c}$ to control pressurizer level.

The minimum DNBR for the initial depressurization immediately following reactor trip is covered by the double MSLB DNBR analyses included in the ONS core reload design, and described in sections 4.1.2 and 4.1.3, above. The double MS HELB case with the limiting return to criticality is provided by **[[**

]] $_{D}^{a,c}$ The brief return to criticality that occurs during this case can be seen after 1500 seconds, and results in a fission power increase to approximately 1 watt. A reload design check has been incorporated into the current ONS reload design process for the double MS HELB scenario.

Section 4.3.1 describes the plant transient response to a single MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using PSW equipment. **[[**

]] $_{D}^{a,c}$

The following operator actions are credited in section 4.3.1. These operator actions do not impact the DNBR evaluation. Operator action is credited to secure the RCPs 3 minutes after loss of RCP seal cooling. The operator isolates the normal letdown flowpath at 20 minutes and starts an PSW powered HPI pump to restore RCP seal cooling at 20 minutes. Full HPI flow through one HPI header is maintained until action to throttle flow is credited to maintain 150°F CETC subcooling until Pressurizer level recovers, after which controlled to maintain 100" pressurizer level. PSW pump flow is available at 14 minutes, but not aligned until RCS temperatures recover to control CETCs to the desired setpoint of 350°F. The PSW powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers. The PSW letdown path from the RCS is through the reactor vessel head vent (RVHV) and loop HPV. The RVHV is opened **[[**

]] $_{D}^{a,c}$, and is subsequently cycled to maintain a 20" pressurizer level band.

The minimum DNBR for the initial depressurization immediately following reactor trip is covered by the Chapter 15 MSLB DNBR analyses. **[[**

]] $_{D}^{a,c}$ The core remains subcritical after the rods insert for the duration of the event. The core then becomes less subcritical while the turbine driven EFW pump is operating. Once the overcooling is terminated, the boron added by the HPI pump ensures the core remains subcritical.

Section 4.3.2 describes the plant transient response to a double MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using PSW equipment. **[[**

]] $_{D}^{a,c}$

The following operator actions are credited in section 4.3.2. These operator actions do not impact the DNBR evaluation. Operator action is credited to secure the RCPs 3 minutes after loss of RCP seal cooling. The operator isolates the normal letdown flowpath at 20 minutes and starts an PSW powered HPI pump to restore RCP seal cooling at 20 minutes. Full HPI flow through one HPI header is maintained until action to throttle flow is credited to maintain 150°F CETC subcooling until pressurizer level recovers, after which controlled to maintain 100" pressurizer level. The PSW powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers sufficiently **[[** **]]** $_{D}^{a,c}$ The PSW letdown path from the RCS is through the RVHV and loop HPV. The RVHV is opened in **[[** **]]** $_{D}^{a,c}$, and is subsequently cycled to maintain a 20" pressurizer level band. PSW pump flow is available at 14 minutes, but not aligned until RCS temperatures recover to control CETCs to the desired setpoint of 350°F. The operator also cycles pressurizer heaters to maintain a desired RCS pressure band.

The minimum DNBR for the initial depressurization immediately following reactor trip is covered by the double MSLB DNBR analyses included in the ONS core reload design and described in sections 4.1.2 and 4.1.3, above. **[[**

]]$_{D}^{a,c}$ The core remains subcritical after the rods insert for the duration of the event. The core then becomes less subcritical during the overcooling while the turbine driven EFW pump is operating. The oscillations are due to hot leg spillover events refilling the core. Once the overcooling is terminated, the boron added by the HPI pump ensures the core remains subcritical.

The requested figures are included after the response to RAI-13.

RAI-9

Attachment 6, for scenarios analyzed in sections 4.1.1, 4.1.2, 4.1.3, 4.2.1, 4.2.2, 4.3.1, and 4.3.2, provide the following for the RCS pressure analysis:

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Transient graphs and the numerical limiting values showing the RCS remains within acceptable pressure and temperature limits.

(d) Please provide the following figures:

1. **[[**
2.
3.
4.
5. **]]**

RAI-9 Response:

Sub parts (a) and (b) for these sections of Attachment 6 are addressed in the response to RAI #8, above. Part (c) on the validation of acceptable pressure and temperature limits is addressed below.

Section 4.1.1 describes single MS HELBs that do not impact the status of 4160 VAC power. These MS HELBs were not specifically analyzed for this LAR. Existing MSLB analyses described in UFSAR Chapter 15.13, and UFSAR Chapter 15.17 similarly assume that 4160 VAC power is not affected by the break. Although MS HELBs are different and distinct from MSLBs described in Chapter 15 of the UFSAR, in this case the Chapter 15 analyses were used as a surrogate. The supporting analyses typically focus on determining the minimum DNBR, which provides the basis for ensuring core cooling. Consequently, the duration of many of these analyses is not sufficient for validating long term RCS pressure and temperature conditions.

The section 4.1.1 analyses are mitigated using normal plant equipment, considering a single failure, with Operations staff in the control room applying the Emergency Operating Procedures (EOP). The EOP includes guidance in Rule 5 “Main Steam Line Break” to stabilize the RCS pressure and temperature and throttle HPI flow. Rule 5 ensures Rule 8 “Pressurized Thermal Shock (PTS)” is in progress or complete. Rule 8 is used when either a cooldown below 400°F at greater than 100°F/hr as measured by the cold leg temperatures has occurred, or if HPI has injected through the safety injection flow paths with all RCPs off. The Rule 8 guidance minimizes subcooled margin using a variety of methods. Once RCS temperature is stable, a 1-hour hold of RCS temperature must be performed unless a loss of coolant accident (LOCA), SG tube rupture, or Blackout is in progress.

To illustrate the importance of RCS subcooling in the transient response, and to enable a review of analyses that do not specifically evaluate the RCS pressure/temperature limits, the information provided by **[[**
]] $_{D}^{a,c}$ is intended to provide an illustration of guidance used in the ONS EOP Blackout Tab for mitigating a FLEX event using the SSF. This figure also helps to illustrate a method for screening transients for the RCS P/T criteria. The red line on this figure is a proposed limit to be added to procedural guidance and represents the minimum temperature/maximum pressure allowable during an event to protect the nil ductility temperature (NDT) curves for all 3 ONS units. The red line on this figure is based on 250°F indicated subcooled margin below 300°F, and a maximum value of 2300 psig above 300°F. Also provided on this figure are the NDT limits for Units 1, 2, & 3, along with 0°F subcooled and 0°F superheated lines that provide an instrument uncertainty corrected prediction of subcooling. The green line is the 150°F indicated subcooled margin curve and represents the desired RCS conditions when performing a natural circulation cooldown. From this figure, three points are taken from the red line to screen the analyses: 1200 psi at 300°F, 800 psi at 250°F, 550 psi at 200°F. These pressure/temperature criteria provide obvious margin to the NDT curves as shown in Figure 8.3.5-1.

The RCS pressure and temperature limits addressed in the discussion below are the 54 EFPY NDT curves described in ONS Technical Specification 3.4.3 and used in the Units 1, 2, & 3 EOPs. For analyses that do not contain pressure/temperature figures, the approach using the pressure/temperature criteria shown above are used to screen the transient responses. NDT does not become a concern until cold leg temperatures decrease below 300°F, above 300°F the NDT curves are above 2300 psig. The section 4.1.1 analyses are depressurization events that never exceed 2300 psig. Below 300°F, the RCS needs to be more than 250°F subcooled to

reach the NDT curves. A couple of examples of MSLB analyses with a sufficient duration are evaluated using the screening criteria provided above.

The results of a long term MSLB analysis, performed for approximately 8 hours using the EOP strategy outlined above, are reviewed to evaluate the RCS pressure and temperature limits. The RCS pressure and cold leg temperature data from this analysis are plotted using the **[[**
]] $_{D}{}^{a,c}$ to verify the RCS remains less than 250°F subcooled for the duration of both the large and small size steam line break cases. The large steam line case assumes a double-ended guillotine break of the steam line. The small steam line break case assumes **[[**

]] $_{D}{}^{a,c}$ The large and small steam line break cases remain below the NDT screening criteria.

DPC-NE-3005-PA (Reference 5) section 15 describes the large steam line break analysis method and provides a demonstration case with offsite power maintained that is 600 seconds in duration. For this case, Figure 15-5 provides RCS temperature and Figure 15-8 provides RCS pressure. ONS UFSAR Figures 15-42 and 15-158 provide RCS temperature and RCS pressure for the current analysis of record for the with offsite power maintained case. When RCS temperatures begin to decrease below 300°F after 300 seconds, RCS pressure is well below 500 psig. This response passes the screening criteria described above for the duration of the case.

Section 4.1.2 describes double MS HELBs where both 4160 VAC and 6900 VAC power remain available. Section 4.1.3 describes scenarios where a LOOP is assumed coincident with the double MS HELBs, and 4160 VAC power remains available. The overcooling resulting from the section 4.1.2 scenarios bounds that obtained from the section 4.1.3 scenarios due to the continued operation of the RCPs. These scenarios are analyzed using NRC approved methods described in the Duke Energy methodology reports DPC-NE-3000-PA (Reference 4) and DPC-NE-3005-PA (Reference 5) modified by the addition of a second break location. These cases are analyzed with the initiating event being a double-ended break of both steam lines in the TB. The initial conditions for the section 4.1.2 scenarios are selected to maximize the RCS depressurization and cooldown, and to maximize any potential return-to-power. The initial conditions for the section 4.1.3 analyses performed to evaluate the limiting minimum DNBR are defined by the SCD methodology. This method specifies that parameters important to the DNBR are set to nominal values and incorporates the associated uncertainty in the DNBR evaluation. Limiting single failures are considered. Transient results of indicated RCS subcooling as a function of time indicate the subcooled margin remains less than 200°F for the duration of the limiting case, which is the RCS remains less than 250°F subcooled screening criteria. The EOP response to the operator actions for mitigating a potential PTS event is previously described for section 4.1.1, above. These actions are also applied to the sections 4.1.2 and 4.1.3 scenarios. Therefore, using the screening criteria discussed above, it is concluded the RCS remains within the pressure and temperature limits.

Section 4.2.1 describes the plant transient response to a single MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using SSF equipment. **[[**
]] $_{D}{}^{a,c}$ provides an illustration of where this case falls in the P/T envelope. The Unit 1, 2 & 3 54 EFPY NDT curves, as well as the 0°F subcooled and superheat curves are provided for reference. This figure shows two curves for the duration of the simulation, the thin red line is the minimum cold leg temperature as a function of RCS pressure, and the thin black line is the RV downcomer temperature as a

function of RCS pressure. This figure shows that minimum cold leg temperatures do not approach the NDT curves during this case.

Section 4.2.2 describes the plant transient response to a double MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using SSF equipment. **[[**

]] $_{D}^{a,c}$ provides an illustration of where this case falls in the P/T envelope. The Unit 1, 2 & 3 54 EFPY NDT curves, as well as the 0°F subcooled and superheat curves are provided for reference. This figure shows two curves for the duration of the simulation, the thin red line is the minimum cold let temperature as a function of RCS pressure, and the thin black line is the RV downcomer temperature as a function of RCS pressure. This figure shows that minimum cold leg temperatures do not approach the NDT curves during this case.

Section 4.3.1 describes the plant transient response to a single MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using PSW equipment. **[[**

]] $_{D}^{a,c}$ provides an illustration of where this case falls in the P/T envelope. The Unit 1, 2 & 3 54 EFPY NDT curves, as well as the 0°F subcooled and superheat curves are provided for reference. This figure shows two curves for the duration of the simulation, the thin red line is the minimum cold leg temperature as a function of RCS pressure, and the thin black line is the RV downcomer temperature as a function of RCS pressure. This figure shows that minimum cold leg temperatures do not approach the NDT curves during this case.

Section 4.3.2 describes the plant transient response to a double MS HELB in the TB due to a HELB that results in damage to, and loss of, the 4160 VAC ES switchgear and other equipment in the TB. This scenario is mitigated using PSW equipment. **[[**

]] $_{D}^{a,c}$ provides an illustration of where this case falls in the P/T envelope. The Unit 1, 2 & 3 54 EFPY NDT curves, as well as the 0°F subcooled and superheat curves are provided for reference. This figure shows two curves for the duration of the simulation, the thin red line is the minimum cold leg temperature as a function of RCS pressure, and the thin black line is the RV downcomer temperature as a function of RCS pressure. This figure shows that minimum cold leg temperatures do not approach the NDT curves during this case.

The requested figures are included after the response to RAI-13.

RAI-10:

For all the scenarios analyzed in Reference 1, Attachment 6, Sections 4.1.1, 4.1.2, 4.1.3, 4.2.1, 4.2.2, 4.3.1, and 4.3.2, provide the following for the steam generator (SG) tube stress analysis.

(a) Initial conditions, including break area and flow rate, used in the analysis justifying they are conservative.

(b) Describe the operator actions credited in the analyses.

(c) Maximum induced stress in the SG tubes and the allowable stress for the SG tube material.

RAI 10 Response:

Sub parts (a) and (b) for these sections of Attachment 6 are addressed in the response to RAI 8, above. Part (c) on the validation of SG tube stress limits is addressed below.

SG tube stress is evaluated in Attachment 6 using the calculated temperature difference between the SG tubes and the SG shell. The results obtained for the overcooling analysis described in section 4 are compared to the temperature differences determined for the SG design transients. This approach confirms the analysis results are within the analyzed tube stress limits.

The limiting compressive and tensile tube stress analyses for the ONS replacement once-through steam generators (ROTSG) are used for this evaluation. The limiting SG tube compressive limit is based on an analysis of the RCS remaining at nominal hot zero power conditions while a dry depressurized SG is allowed to cool. This situation will result in the plant experiencing the compressive tube stress analytical limit defined by the RCS at 555°F and the SG shell at 212°F for a +343°F tube-to-shell ΔT (Reference 8, section 1.0).

The tensile SG tube stress limits are defined by a large break LOCA tube stress analysis that assumes a double-ended RCS pipe break at the top of the hot leg U-bend. This break location allows cold emergency core cooling system fluid to flow backwards through the tubes to the break as the RCS is refilled. The limiting tube-to-shell temperature difference in this analysis is -374°F (Reference 9). Reference 9 contains several design calculations performed by B&W Canada for the ONS ROTSGs. **[[**

]] $_{D}^{a,c}$

MSLB analyses are performed to determine the limiting SG tube stress. These analyses assume an initial core power level of 102% of 2568 MW and hot full power conditions. RCS integrity is demonstrated by determining the limiting SG tube compressive and tensile stresses remain within design limits, and that the RCS pressure and temperature remains within the acceptable cooldown limits during the transient evolution. The maximum tensile stress resulting from a single MSLB is significantly less than the limiting tensile stress that results from a large break LOCA. With 4160 VAC power available, normal plant equipment is able to maintain the plant within limits during the cooldown.

For the single MS HELB cases described in section 4.1.1, the ROTSG tube-to-shell temperature difference is determined to be acceptable by an analysis performed from 102% of 2568 MW with initial conditions selected to be conservative for the application. The limiting tube-to-shell ΔT of -233.3°F for a single MSLB is obtained, which is significantly less than the limiting large break LOCA tube-to-shell ΔT of -374°F.

For the double MS HELB cases described in sections 4.1.2 and 4.1.3, the ROTSG tube-to-shell temperature difference is determined to be acceptable by an analysis performed from 102% of 2568 MW with initial conditions selected to be conservative for the application. The limiting tube-to-shell ΔT of -269.8°F for a double MSLB is obtained, which is significantly less than the limiting large break LOCA tube-to-shell ΔT of -374°F.

For the single MS HELB cases mitigated using SSF equipment described in section 4.2.1, the ROTSG tube-to-shell temperature difference is provided by **[[** **]]** $_{D}^{a,c}$ (Reference 7). The maximum compressive tube-to-shell ΔT for Loops A and B are 245.5°F and 58.3°F, respectively. These values are well below the 343°F compressive limit. The maximum tensile tube-to-shell ΔT for Loops A and B are -169.4°F and -40.0°F, respectively. These values are well below the tensile limit of -374°F.

For the double MS HELB cases mitigated using SSF equipment described in section 4.2.2, the ROTSG tube-to-shell temperature difference is provided by **[[ ]]**$_{D}^{a,c}$ The maximum compressive tube-to-shell ΔT for Loops A and B are 105.1°F and 157.9°F, respectively. These values are well below the 343°F compressive limit. The maximum tensile tube-to-shell ΔT for Loops A and B are -151.3°F and -142.2°F, respectively. These values are well below the tensile limit of -374°F.

For the single MS HELB cases mitigated using PSW equipment described in section 4.3.1, the ROTSG tube-to-shell temperature difference is provided by **[[ ]]**$_{D}^{a,c}$ The maximum compressive tube-to-shell ΔT for Loops A and B are 54.7°F and 58.8°F, respectively. These values are well below the 343°F compressive limit. The maximum tensile tube-to-shell ΔT for Loops A and B are -152.0°F and -67.8°F, respectively. These values are well below the tensile limit of -374°F.

For the double MS HELB cases mitigated using PSW equipment described in section 4.3.2, the ROTSG tube-to-shell temperature difference is provided by **[[ ]]**$_{D}^{a,c}$ The maximum compressive tube-to-shell ΔT for Loops A and B are 81.4°F and 96.7°F, respectively. These values are well below the 343°F compressive limit. The maximum tensile tube-to-shell ΔT for Loops A and B are -198.0°F and -149.7°F, respectively. These values are well below the tensile limit of -374°F.

RAI-11:

Attachment 6, section 3.2 states:

> Pressurizer level increases with increasing RCS temperatures and goes off scale high. The pressurizer does not become water solid prior to SSF [standby shutdown facility] ASW [auxiliary service water] being aligned to the SGs at 14 minutes, and liquid relief is not predicted through the PSVs [pressurizer safety valves] or PORV [power operated relief valve]. Maintaining a steam space in the pressurizer is dependent on the timing of providing a heat sink.

Attachment 6, section 3.3 states:

> Pressurizer level increases with increasing RCS temperatures and goes off-scale high. The pressurizer does not become water solid prior to PSW [protected service water] being aligned to the SGs, and liquid relief is not predicted through the PSVs or PORV when PSW is started within 14 minutes. Maintaining a steam space in the pressurizer is dependent on the timing of providing a heat sink.

In above statements, it is predicted that the pressurizer would not become water-solid prior to the alignment of SSF ASW or PSW system for the mitigation of FDW HELB in the turbine building (TB) during the overheating event so that liquid relief would not take place through the PSVs or PORVs. Explain how it is determined that the pressurizer would not become water-solid. Describe the analysis and results on which this prediction is based.

RAI-11 Response:

Sections 3.2 and 3.3 describe the overheating analyses mitigated using the SSF or PSW equipment. These analyses are **[[**

]] This figure indicates the pressurizer does not become water-solid.

RAI-12

Attachment 6, section 3.2 and section 3.3 states the analysis is based on "ANS [American National Standard]-79 with uncertainty)" decay heat. The decay heat and its uncertainty are not specified in any of the remaining analyses.

(a) Specify what uncertainty value was used in the analysis.

(b) State the decay heat standard with its uncertainty value used for the remaining Thermal-Hydraulic analysis described in Attachment 6.

(c) Please provide the following figures:

1. **[[**
2. **]]**

RAI-12 Response:

Sections 3.2 and 3.3 describe the overheating analyses mitigated using the SSF or PSW equipment. The decay heat is based on ANS-79 with a +2σ uncertainty. A high 24-month EOC decay heat is used in these analyses to increase post-trip RCS temperatures and the resulting pressure excursion.

The overheating analyses described in sections 3.1.1 and 3.1.3 use the same decay heat assumptions as sections 3.2 and 3.3.

The overcooling analyses described in sections 4.1.1, 4.1.2, and 4.1.3 use the guidance provided in DPC-NE-3005-PA (Reference 5). DPC-NE-3005-PA Section 15 describes the large steam line break analysis method used to perform the MSLB analyses described in UFSAR Chapter 15.13. To maximize the RCS cooldown, a low decay heat power level assuming a multiplier of 0.9 is applied to the 1979 ANS Standard 5.1 decay heat power. The UFSAR Chapter 15.17 analyses do not use this guidance as decay heat is not an important parameter for these scenarios.

The overcooling analyses described in sections 4.2 and 4.3 use decay heat assumptions based on the DPC-NE-3005-PA Section 15 guidance. The decay heat is based on ANS-79 with a +2σ uncertainty and a 0.9 multiplier.

The requested figures are included after the response to RAI-13.

RAI-13:

Attachment 6, section 4.1.2, fourth paragraph, page 7 states:

> Since the RCPs continue to operate in this case, the DNBR is bounded by the double MS HELB without offsite power case where RCPs [reactor coolant pumps] are lost immediately.

Describe the relationship of the DNBR with RCP in operation. Explain why the DNBR for a double MS HELB with offsite power available is bounded with offsite power not available during the same scenario.

RAI-13 Response:

The single MS HELB analyses described in section 4.1.1 and the double MS HELBs described in section 4.1.2 are both analyzed using the NRC approved methods described in the Duke Energy methodology reports DPC-NE-3000-PA (Reference 4) and DPC-NE-3005-PA (Reference 5). The limiting minimum DNBR statepoints from these analyses are included in the normal ONS reload core design process.

The time frame over which DNBR is a concern occurs prior to the completion of control rod insertion. For MS HELBs at ONS, due to the amount of time required to reach the low RCS pressure reactor trip setpoint, this leads to short duration analyses, on the order of seconds. For scenarios where offsite power remains available and the RCPs remain operating, the mass flow entering the core increases as the core inlet temperature decreases. The potential decrease in core exit pressure is limited by the liquid saturation temperature. The limiting DNBR during this time frame occurs when the core inlet mass flow decreases while core discharge pressure decreases, before the heat flux from the fuel has time to decay. This discussion presumes a significant return-to-power does not occur. Should a return to power occur, the ONS MS HELB analyses evaluate the scenario in detail. DNBR for a scenario including a return-to-power is evaluated and determined to be acceptable. Thus, scenarios where RCPs are assumed to lose power and coastdown bound scenarios where the RCPs remain operating.

Figures referenced by Duke Responses:

The following figures are provided to support the responses provided above where noted. As these figures are identified by calculation number, all figures are bracketed as proprietary.

[[

]] D[a,c]

* Note that figure added to request by Duke Energy.

[ ]D[a,c]

[ ] a,c

[ ]a,c

[ ] D[a,c]

[ ]$^{a,c}_{D}$

[ ] D[a,c]

[ ] D[a,c]

[ ]a,c

[ ]a,c

[ ] D[a,c]

[ ]D[a,c]

[ ]D[a,c]

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[ ]D[a,c]

[ ] D[a,c]

Miscellaneous RAI:

The NRC requested that the significant hazards consideration be updated to include statements to reflect that the analysis and results of over 3000 postulated break locations per unit actually represent an improvement in the safety of the plant due to the comprehensive nature of the analysis.

Miscellaneous RAI Response:

Enclosure 1 contains a revised version of the significant hazards consideration. It has been revised to contain more positive statements that the analysis and results of over 3000 postulated break locations per unit actually represent an improvement in the safety of the plant due to the comprehensive nature of the analysis. Changes have been marked by revision bars on the right.

References:

1. Letter from Duke Energy to NRC, dated August 28, 2019, “Duke Energy Carolinas, LLC Oconee Nuclear Station Renewed Facility Operating License Numbers DPR-38, DPR-47, and DPR-55 Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Breaks Outside of the Containment Building” (ADAMS Accession No. ML19240A925).
2. **[[**

   ]] D[a,c]
3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 386, 388, and 387, Implementation of the Protected Service Water System, dated August 13, 2014 (ADAMS Accession Number ML 14206A790).
4. DPC-NE-3000-PA, Oconee Nuclear Station, McGuire Nuclear Station, Catawba Nuclear Station, Thermal-Hydraulic Transient Analysis Methodology, Revision 5. (Safety Evaluations for Oconee Nuclear Station dated August 8, 1994 (ADAMS Accession Number ML16293A840); October 14, 1998 (ADAMS Accession Number 9810190223); September 24, 2003 (ADAMS Accession Number ML032670816); October 29, 2008 (ADAMS Accession Number ML082800408); and July 21, 2011 (ADAMS Accession Number ML11137A150).
5. DPC-NE-3005-PA, Oconee Nuclear Station, UFSAR Chapter 15 Transient Analysis Methodology, Revision 5. (Safety Evaluations dated October 1, 1998; May 25, 1999; September 24, 2003; October 29, 2008; July 21, 2011 (ADAMS Accession Number ML11137A150); and April 29, 2016 (ADAMS Accession Number ML16088A330)).
6. Letter dated July 11, 2012 from T. Preston Gillespie, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC to the Nuclear Regulatory Commission Document Control Desk, “Licensing Basis for Protected Service Water System – Responses to Request for Additional Information” (ADAMS Accession Number ML12195A325).

7. **[[** **]]** D^{a,c}
8. **[[** **]]** D^{a,c}
9. **[[** **]]** D^{a,c}
10. DPC-NE-3003-PA Duke Energy Methodology Report DPC-NE-3003-PA, Oconee Nuclear Station, Mass and Energy Release and Containment Response Methodology, Revision 1. (Safety Evaluations dated March 15, 1995; September 24, 2003 (ADAMS Accession Number ML032670816)).
11. **[[** **]]** D^{a,c}
12. **[[** **]]** D^{a,c}
13. **[[** **]]** D^{a,c}
14. **[[** **]]** D^{a,c}
15. **[[** **]]** D^{a,c}

ATTACHMENT 1

Duke Energy Affidavit

Attachment 1
Duke Energy Affidavit

AFFIDAVIT of Steve Snider

1. I am Vice President of Nuclear Engineering, Duke Energy Carolinas, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke Energy.
2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy's application for withholding which accompanies this affidavit.
3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential. I am familiar with the Duke Energy information contained in Enclosure 3 of the Oconee responses (correspondence no. RA-20-0170) to the Requests for Additional Information (RAI) issued by the Office of Nuclear Reactor Regulation for the Oconee License Amendment request which proposes to update the Updated Final Safety Analysis Report (UFSAR) regarding the High Energy Line Break (HELB) licensing basis (correspondence no. RA-19-0253).
4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke Energy and has been held in confidence by Duke Energy and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke Energy. Information is held in confidence if it falls in one or more of the following categories.

      (a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by a vendor or consultant, without a license from Duke Energy, would constitute a competitive economic advantage to that vendor or consultant.

      (b) The information requested to be withheld consist of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage for example by requiring the vendor or consultant to perform test measurements, and process and analyze the measured test data.

      (c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation assurance of quality or licensing of a similar product.

      (d) The information requested to be withheld reveals cost or price information, production capacities, budget levels or commercial strategies of Duke Energy or its customers or suppliers.

(e) The information requested to be withheld reveals aspects of the Duke Energy funded (either wholly or as part of a consortium) development plans or programs of commercial value to Duke Energy.

(f) The information requested to be withheld consists of patentable ideas.

The information in this submittal is held in confidence for the reasons set forth in paragraphs 4(ii)(a) and 4(ii)(c), above. Rationale for this declaration is the use of this information by Duke Energy provides a competitive advantage to Duke Energy over vendors and consultants, its public disclosure would diminish the information's marketability, and its use by a vendor or consultant would reduce their expenses to duplicate similar information. The information consists of analysis methodology details that provides a competitive advantage to Duke Energy.

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in the RAI response is that which is marked by brackets in the responses to the RAIs in Enclosure 3. These are responses to the RAIs for the Oconee License Amendment request which proposes to update the Updated Final Safety Analysis Report (UFSAR) regarding the High Energy Line Break (HELB) licensing basis. This information is consistent with marked proprietary information in the NRC-approved Duke Energy methodology reports DPC-NE-3000-PA, DPC-NE-3003-PA, and DPC-NE-3005-PA. This information enables Duke Energy to:

(a) Support license amendment requests for its Oconee reactors.

(b) Perform transient and accident analysis calculations for Oconee.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke Energy.

(a) Duke Energy uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke Energy can sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke Energy.

Attachment 1
Duke Energy Affidavit

5. Public disclosure of this information is likely to cause harm to Duke Energy because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke Energy to recoup a portion of its expenditures or benefit from the sale of the information.

Steve Snider affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

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

Executed on June 3, 2020.

____
Steve Snider