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DUKE ENERGY®

ONS-2015-066 10 CFR 50.90

June 8, 2015

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed Operating License Nos. DPR-38, DPR-47, and DPR-55

Response to Request for Additional Information Regarding the License Amendment Request (LAR) to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection (HPI) Equipment is Inoperable
License Amendment Request No. 2013-03

References:
1. Letter from Scott L. Batson (Duke Energy) to Document Control Desk (USNRC), "License Amendment Request (LAR) to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection (HPI) Equipment is Inoperable, License Amendment Request No. 2013-03," dated June 30, 2014 (ML14184B384)
2. Letter from James R. Hall (USNRC) to Scott L. Batson (Duke Energy), "Oconee Nuclear Station, Units 1, 2, and 3 - Request for Additional Information Regarding License Amendment Request to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection Equipment is Inoperable (TAC Nos. MF4668, MF4669, and MF4670)," dated May 21, 2015 (ML15141A147)

In Reference 1, Duke Energy Carolinas, LLC (Duke Energy) submitted a license amendment request under 10 CFR 50.90 for Oconee Nuclear Station (ONS) Units 1, 2, and 3 to revise Technical Specification (TS) 3.5.2 by reducing the allowed maximum rated thermal power (RTP) at which the unit can operate when select HPI system equipment is inoperable.

Reference 2 transmitted Requests for Additional Information (RAIs) regarding the Reference 1 submittal. The responses to these requests are provided in the Attachment. This additional information does not impact the 10 CFR 50.92 evaluation of "No Significant Hazards Consideration" previously provided in Reference 1.

This letter makes no new commitments or changes to any existing commitments.

Should you have any questions regarding this request, please contact Sandra Severance at (864) 873-3466.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 8th day of June, 2015.

Sincerely,

Scott L. Batson
Vice President
Oconee Nuclear Station

Attachment: Response to Request for Additional Information Regarding the License Amendment Request (LAR) to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection (HPI) Equipment is Inoperable License Amendment Request No. 2013-03

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Attachment

Response to Request for Additional Information Regarding the License Amendment Request (LAR) to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection (HPI) Equipment is Inoperable
License Amendment Request No. 2013-03

Attachment

Response to Request for Additional Information Regarding the License Amendment Request (LAR) to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection (HPI) Equipment is Inoperable
License Amendment Request No. 2013-03

NRC Background

By letter dated June 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14184B384), Duke Energy (the licensee) submitted a license amendment request (LAR) for Oconee Nuclear Station (ONS), Unit Nos. 1, 2, and 3. The proposed amendment would revise Technical Specification 3.5.2 by reducing the allowed maximum Rated Thermal Power at which each unit can operate when select High Pressure Injection system equipment is inoperable. The U.S. Nuclear Regulatory Commission staff has reviewed the information submitted by the licensee, and has determined that additional information is needed to complete its review.

NRC Request #1

On November 26, 2014, AREVA, Inc. filed a report in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 21 "Reporting of Defects and Noncompliance," addressing a defect in the LOCA analysis for B&W plants (ADAMS Accession No. ML14337A095). Explain the impact of the fuel Thermal Conductivity Degradation identified by AREVA in that report on the Oconee Nuclear Station's small-break loss-of-coolant accident (LOCA) analysis of record.

Duke Energy Response

The 10 CFR 50.46 letter provided by Duke Energy to the NRC on December 17, 2014, Agencywide Documents Access and Management System (ADAMS) Accession No. ML14353A214 (Reference 1), described the impacts of thermal conductivity degradation (TCD) on the large-break loss-of-coolant accident (LBLOCA) analyses. It also included a statement that small-break loss-of-coolant accident (SBLOCA) analyses are not affected by the TCD issue and provided an estimate of 0 F change to the limiting SBLOCA peak cladding temperature (PCT). It is consistent with the AREVA response to SBLOCA RAI 4 provided in Volume 3 of BAW-10192P-A (Reference 2) that stated the effect on SBLOCA PCT is insensitive to initial fuel stored energy.

The RAI specifically asks for an explanation of TCD for the ONS SBLOCA analyses of record, which includes the full power, all Emergency Core Cooling System (ECCS) equipment initially available as well as the partial power 1 HPI pump out-of-service conditions that is the subject of this LAR. While the LAR specifically is for the partial power scenario, the similarities of the full power and partial power SBLOCA analyses allow them to be addressed in one response. The similarity is that in the analyses the hot bundle and hot pins have the same initial power in the entire fuel rod independent of core powers between 50 percent and 100 percent of full power. Therefore, the hot rod fuel pin powers and temperatures are similar. The average channel is lower at partial core powers, but other than an energy source to the problem, it is not limiting and results from full power are bounding of the partial power conditions.

The spectrum of potentially limiting SBLOCA break sizes that makes up the ONS analyses of record encompasses break areas between 0.01 ft^2 and the maximum reactor coolant system (RCS) piping break size for which there is no cladding departure from nucleate boiling (DNB) during the first few seconds after break opening. This is consistent with the methods in the NRC-approved LOCA Evaluation Model (EM) contained in BAW-10192P-A Revision 0. Breaks that do cause DNB are considered large break LOCAs. The location of the break, fuel design, core peaking, total core power, RCS initial flows, and off-site power availability are all important in determining the break size that initially undergoes DNB. A break in the cold leg pump discharge (CLPD) region results in a decrease in the core flow due to break discharge. Larger SBLOCAs result in faster RCS depressurization with additional flashing and core boiling that increases the quality in the core and reduces the critical heat flux (CHF) rate. A high total peaking at full power with a core exit skewed axial peak also reduces the maximum break size that does not exceed DNB. For the 102% full power analysis of record (AOR), DNB is predicted for the Mark-B-HTP fuel for a CLPD break with cross- sectional area slightly larger than 0.5 ft^2.

The limiting SBLOCA break sizes from a PCT perspective generally progress through five phases: (1) subcooled depressurization, (2) reactor coolant pump and loop flow coastdown and natural circulation, (3) loop draining, (4) boiling pot, and (5) refill and long-term cooling. If there is core uncovering, the PCT will occur during the later portion of the boiling pot phase and beginning of the refill phase before the top of the heated core is covered by a two phase mixture level. The larger SBLOCAs are not limiting from a PCT perspective, but they do have the earliest PCT time. Any impact from fuel stored energy changes related to TCD changes are magnified by the shortest time interval between break opening and the PCT time as they have the least time to remove the fuel pin stored energy. However, as will be shown, the bulk of the fuel stored energy due to fission is removed well before the core uncovers and the cladding temperatures begin to heat up to the maximum cladding temperature for these larger break sizes.

For a SBLOCA, the highest power fuel pin regions do not go through DNB during the initial flow and pressure decrease. The RCS pressure decreases below the low pressure reactor trip setpoint and the subsequent control rod insertion shuts down the core fission power production. Thereafter, the energy removal of decay heat and stored energy in the lower portion of the core is by the forced convection process, while the upper portions of the core are cooled by subcooled or saturated nucleate boiling. These heat transfer regimes rapidly remove the stored energy from the fuel pin following shutdown. The larger break flows degrade the core flow and this, in combination with the RCS depressurization, results in local voiding from flashing and boiling. The fuel pin surface heat transfer degrades faster for larger break sizes so these are the primary focus when considering DNB. The Oconee 102% full power 0.5-ft^2 CLPD break with LOOP was used to assess the TCD impacts for SBLOCA. While the LAR is specifically focused on the partial power SBLOCA results, the hot bundle power is identical for both the 52% and 102% full power cases. Therefore, the 102% full power case can be used as a generic response that is slightly conservative as it has additional decay heat and stored energy in the average core that must be removed as well.

For both the 102% full power and the 52% full power steady state conditions at the time of break opening, the core power peaking is set at a maximum nuclear source linear heat rate (LHR) of 17.3 kW/ft based on an axial peak at 11 ft elevation from the bottom of the heated core. At the onset of the SBLOCA transient, the volume-average fuel temperature (VAFT) is set to a 95/95 bounding beginning of life (BOL) hot rod peak value of approximately 2420 F. At steady-state, the hot rod, hot spot temperature produces the highest fuel centerline and VAFT as it has the highest fission power. The hot spot temperature distribution across the fuel pin is shown in Figure 1 (taken from the 0.0001 second edit of the 0.5-ft^2 AOR loss-of-offsite power case). This distribution is established by the local power, the fuel pin thermal conductivity, and the fuel pellet radial power distribution. This peak power location has nearly a 2900 F temperature across the fuel pellet at the BOL steady-state condition. The fuel performance codes show that for the hot spot, the VAFT decreases with burnup as the pellet to cladding gap closes at a constant LHR. Once the gap closes, the VAFT would begin to rise due to the fuel pellet TCD when LHR is held at a constant. Near a rod average burnup of 34 GWd/mtU, the UO_2 VAFT uncertainty might need to be approximately 11.5 percent higher to account for TCD, but the LHR limit was reduced by 2 kW/ft or ~11.6 percent (2 kW/ft / 17.3 kW/ft) to keep the VAFT similar to support LBLOCA applications. In addition, the LHR limit is reduced by a larger percentage at higher burnups to ensure that the LBLOCA PCTs between middle of life (MOL) to end of life (EOL) burnups cannot become limiting.

The LHR decrease needed to compensate for TCD during an LBLOCA could also be credited in the SBLOCA analyses; however, although the reduction occurs, it does not have to be credited to demonstrate that TCD effects are not significant for SBLOCA. At steady-state, the LBLOCA MOL VAFT uncertainty had to be increased by roughly 11.5% based on the VAFT in Celsius. Assuming the bounding SBLOCA BOL VAFT of 2420 F is maintained, it would need to be increased by roughly 280 F to obtain an 11.5% increase. It is noted that the 95/95 VAFT increase determined for LBLOCA was the value that was needed to force the TACO3 (Reference 3) steady-state fuel pin code VAFT without TCD modeled to match or exceed the COPERNIC2 (Reference 4) steady-state fuel pin code 95/95 VAFT with TCD explicitly modeled. For the LBLOCA TCD analyses, an increase to the TACO3 VAFT of 228 F bounds the prediction from COPERNIC2 at MOL. It is noted that this value is less than the 280 F approximated for the SBLOCA TCD stored energy increase. The LBLOCA VAFT increase was less than 280 F because the fuel temperature initially decreases between BOL and MOL due to clad creep down and fuel pellet swelling that was not explicitly analyzed or credited for this example.

The temperature rise across the pellet at BOL was 2900 F just prior to break opening. If the temperature rise needs to increase by 280 F at MOL, then the effective thermal conductivity change needed across the pellet is effectively ~10% lower to account for the TCD. After reactor trip for an SBLOCA, the cladding remains in pre-CHF heat transfer regimes and this heat removal, combined with reactor trip to shut down the fission power, allows the fuel pellet temperatures to decrease rapidly. The fuel pellet VAFT drops to 840 F by 20 seconds into the transient as the temperature distribution equilibrates with the decay heat power production rate. The VAFT continues to drop as the decay heat decreases and it is down to 650 F at 70 seconds into the event. The fuel pin temperature distributions at 20 and 70 seconds are also shown in Figure 1.

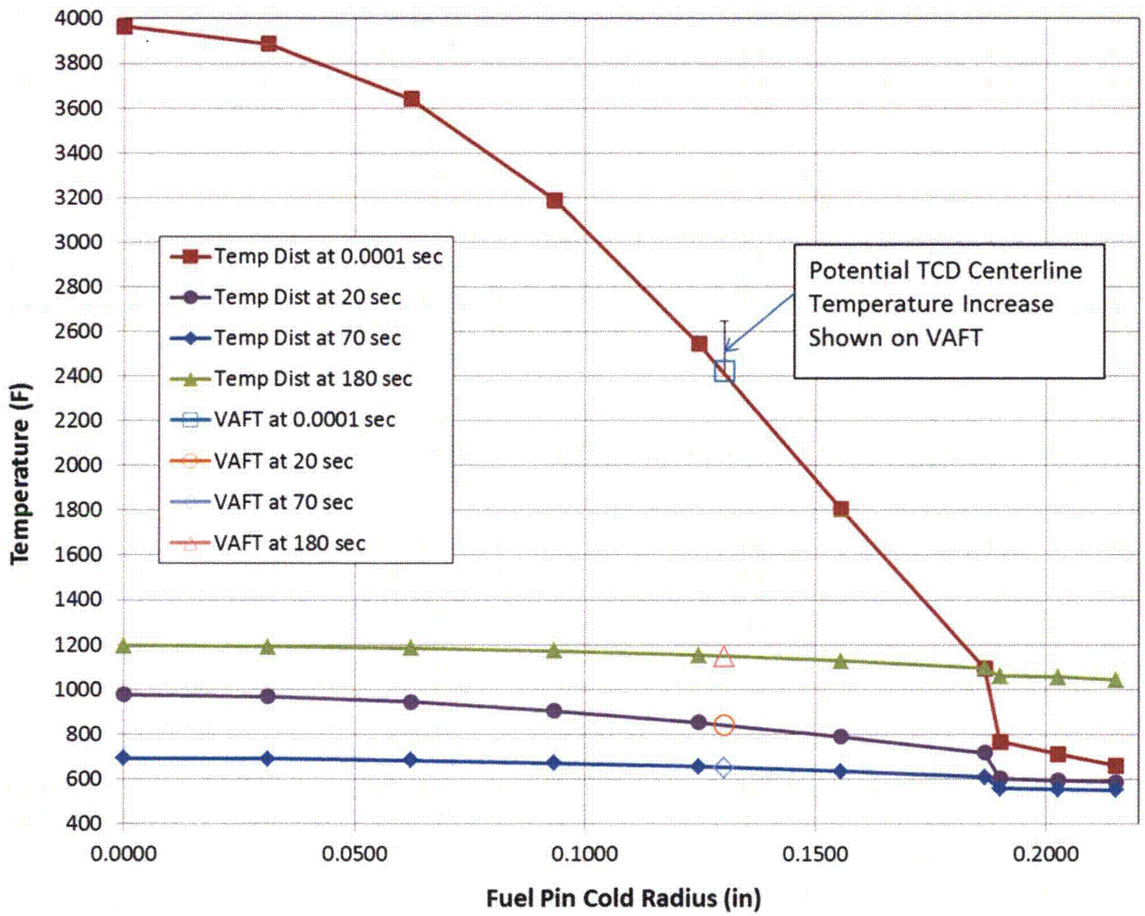
The figure shows that the temperature change across the pellet is proportional to the power produced. The temperature distribution is established as a quasi-steady balance between the fuel power production and the thermal conduction that transmits it to the surface of the pellet,

across the gap to the cladding and finally into the coolant. At 180 seconds post trip (slightly after the PCT time for the 0.5-ft^2 CLPD break from either 102% or 52% full power), the decay heat contribution is approximately 4% full power (FP). Therefore, the initial 2900 F temperature rise across the pellet at 102% full power decreases by the power ratio to an estimated 114 F {2900 F * (4% * Po) / (102% * Po)}, where Po is the steady-state full power value. (Note: This simple estimate of fuel pellet temperature distribution is confirmed by the fuel pin temperatures taken from the 0.5-ft^2 CLPD break at 102% full power from the 180 second edit as shown in Figure 1.) If the 10 percent fuel temperature change due to TCD was applied to this temperature rise, it would be estimated that the centerline temperature could be roughly 12 F higher. Since the fuel thermal conductivity is lower, the VAFT could increase by roughly half of the centerline temperature, while keeping the same temperature change across the fuel pin gap that produced the previous PCT without TCD. Since the fuel temperature could increase, there would be less energy reaching the cladding and the PCT might be a few degrees lower. Another way to look at it is, if the VAFTs were the same, then half of the temperature change across the pellet could allow the PCT to be a few degrees lower. Regardless, the PCT with TCD stored energy and lower conductivity in the transient would be expected to be within +/- 6 F of the value calculated with BOL stored energy and the BOL thermal conductivity for the largest SBLOCA break size. It should be reiterated that the 0.5-ft^2 CLPD break does not produce the limiting PCT results and the LHR limit reduction used for LBLOCA TCD offset was not required or credited for SBLOCA.

For smaller break sizes, the time of PCT is later so the decay heat is lower. Thus, at the SBLOCA time of PCT, the ratio of the decay heat generation rate to full power is lower, so the impact of TCD on the PCT change would be proportionally lower as well. Given the small potential that TCD has to change the SBLOCA fuel pin temperature at the time of PCT, potentially either up or down, the PCT change for the limiting SBLOCA was estimated to be zero.

The effects of fuel pellet TCD on SBLOCA results has been considered for the spectrum of break sizes. The larger, non-limiting, SBLOCAs have the potential to be the most impacted; however, the potential change is very small and it could be slightly positive or negative based on quantitative values. Therefore, use of a value of zero is reasonable for all SBLOCAs that predict core uncovering and cladding heatup.

Figure 1: Representative Temperature Distribution over the Fuel Pin Radius at Several Times during a 0.5-ft^2 CLPD SBLOCA



NRC Request #2

10 CFR 50.46(a)(1)(i) requires that performance of the emergency core cooling system should be analyzed using a comprehensive selection of assumed breaks. The analysis presented in the license amendment request did not consider reactor coolant system (RCS) breaks in the hot-leg and the reactor coolant pump suction. Provide the rationale for not considering all the possible spectrum of break locations in support of the LAR.

Duke Energy Response

AREVA Topical Report BAW-10192P-A, Volume II, Section 4.3.2.2 states that for hot leg breaks, all of the Emergency Core Cooling System (ECCS) injection flow is available for core cooling. Section 4.3.2.4 of BAW-10192P-A, Volume II indicates that the bottom of the CLPD piping is the limiting location for SBLOCAs, and that CLPD break locations, as well as core flood tank line and high pressure injection line breaks are included in the SBLOCA break spectrum.

Section 3.14 of the NRC's Safety Evaluation for BAW-10192P, Rev. 0 indicates that analysis of a double-ended guillotine break in the pump suction piping resulted in a lower peak clad temperature prediction than the case of a double-ended guillotine break in the pump discharge piping. This is attributed to retention of a significant amount of ECCS water during blowdown which shortened the adiabatic heatup period. Pump suction piping break flow rates are also lower than breaks in the pump discharge piping because of increased pump resistance due to the changed break location.

Therefore, hot leg breaks and Reactor Coolant Pump (RCP) suction line breaks are not included in the SBLOCA break spectrum per the approved SBLOCA methodology described in BAW-10192P-A. The Oconee partial-power SBLOCA analyses performed to support the proposed change to ONS TS 3.5.2 are performed in accordance with the NRC-approved LOCA evaluation model described in BAW-10192P-A with respect to SBLOCA break spectrum locations.

NRC Request #3

The technical specifications (TSs), as defined in 10 CFR 50.36, are required to be representative of the design and plant specific operating conditions. Provide the rationale that the planned changes to the Oconee TSs and emergency operating procedures (EOPs) are representative of all 3 Oconee units.

Duke Energy Response

All of the Oconee units are of the Lowered Loop B&W plant design, which are applicable plant designs for LOCA analyses performed in accordance with AREVA Topical Report BAW-10192P-A (See Table 1-1 of BAW-10192P-A).

The replacement once-through steam generators (ROTSGs) at all Oconee units are identical in design, and a common steam generator tube plugging assumption is used in the ONS partial-power SBLOCA analyses supporting the proposed change to Oconee TS 3.5.2. The core flood tanks at each Oconee unit are controlled to the same operating bands for water volume and nitrogen cover pressure via Oconee TS 3.5.1.

The RCPs installed in the Oconee Unit 1 plant are the Westinghouse type, while the Oconee Units 2 and 3 plants utilize the Bingham pump type. For a bounding analysis, the Westinghouse pump type is chosen with RELAP5/MOD2-B&W two-phase fully degraded head difference curves and the M3-modified two-phase void dependent degradation multiplier. For smaller SBLOCAs, the RCPs are modeled to trip on loss of offsite power (LOOP) coincident with reactor trip. For those SBLOCA cases that do not postulate LOOP, an operator action to manually trip the RCPs is credited at 2 minutes following loss of subcooling margin.

In Reference 5, the NRC concluded that the M3-modified curve is applicable to RCPs in operation at the B&W plants to ensure operation in compliance with 10 CFR 50.46, and that use of the M3 RCP degradation curve is acceptable for all B&W plants when predicting time available for operator action to trip RCPs following loss of subcooling margin.

NRC Request #4

10 CFR 50, Appendix B, Criterion V, requires that procedures affecting quality shall be appropriate to the circumstances and should include acceptance criteria for determining that important activities have been satisfactorily accomplished. Describe how the EOPs will be modified to accomplish these requirements so as to inhibit the intrusion of nitrogen into the RCS by the core flood tanks during a LOCA event without compromising any of the many other safety analysis requirements.

Duke Energy Response

For partial-power SBLOCA scenarios where enhanced primary-to-secondary heat transfer via the steam generators and depressurization via the atmospheric dump valve (ADV) flow path is credited, the Oconee Emergency Operating Procedures (EOPs) are planned to be modified to modulate Main Steam pressure in a range of 250 -275 psig that will preclude the Core Flood Tanks (CFTs) from fully emptying their water inventory. Guidance is planned to be placed in the Loss of Subcooling Margin section of the Oconee EOPs for the case where a condition of Inadequate High Pressure Injection (HPI) flow is met as determined by the operators.

For the limiting partial-power SBLOCA scenarios analyzed for the proposed change to Oconee TS 3.5.2, the timing of this action to modulate Main Steam pressure would occur at time greater than 1500 seconds from the event initiation.

Per the Applicable Safety Analysis section of the Oconee Technical Specification Bases (ONS TSB 3.5.1), the CFTs are credited in both the large and small break LOCA analyses. In addition to LOCA analyses, the CFTs have been assumed to operate to provide borated water for reactivity control for severe overcooling events such as a main steam line break (MSLB).

For the Large Break LOCA (LBLOCA) event, the CFTs discharge to the reactor coolant system (RCS) as soon as RCS pressure decreases below CFT pressure. Since the CFTs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46. For the Oconee LBLOCA analyses, RCS pressures decrease very rapidly, and the CFTs are discharged in the first 50 seconds of the LBLOCA event. Therefore, the proposed EOP changes will have no impact to the LBLOCA safety analyses described in Oconee Updated Safety Analysis Report (UFSAR) Section 15.14.

For SBLOCA events initiated at full-power conditions, the limiting peak cladding temperature occurs at approximately 640 seconds per Oconee UFSAR Figure 15-231, when adequate HPI flow is being provided to the RCS. For this case, the main steam pressures remain well above the 250 – 275 psig range to be used in the EOPs, as shown in Oconee UFSAR Figure 15-228. For the limiting SBLOCA event from full-power conditions, the analysis is terminated at approximately 1300 seconds, per Oconee UFSAR Figure 15-231. The ADV flow path is not used to depressurize the steam generators for the full-power SBLOCA analyses; therefore, the proposed EOP changes do not affect full-power SBLOCA analyses described in Oconee UFSAR Section 15.14.

For the Oconee Large Main Steam Line Break without LOOP as described in Oconee UFSAR Section 15.13.2, the overcooling event is terminated at 10 minutes, since operator action is credited at that point to terminate feeding of the faulted steam generator, which stops the uncontrolled RCS cooldown. Per Oconee UFSAR Figure 15-43, the core does not experience a return to power during the event, with a margin to criticality of approximately 2.5 dollars. The CFTs do inject into the RCS for this Large Steam Line Break event, however the CFT injection ceases at approximately 500 seconds into the event as the CFT and RCS pressures equilibrate. After this point in time, the boron concentration in the RCS continues to increase due to continued HPI flow with suction from the Borated Water Storage Tank.

Therefore, the proposed EOP changes will have no impact to the Large Main Steam Line Break without LOOP safety analyses described in Oconee UFSAR Section 15.13.2, since the timing of these operator actions would occur well after the point in time when the safety analysis acceptance criteria have been demonstrated.

For the Oconee Large Main Steam Line Break with LOOP as described in ONS UFSAR Section 15.13.3, the event is only modeled for the first five seconds after the break, since the point of minimum Departure from Nucleate Boiling Ratio (DNBR) occurs at approximately 1.5 seconds, per Oconee UFSAR Figure 15-167. The RCS pressure remains above 1800 psig during the event, which is well above the CFT pressure, per Oconee UFSAR Figure 15-166. Therefore the proposed EOP changes have no impact to the Large Main Steam Line Break with LOOP safety analyses described in Oconee UFSAR Section 15.13.3.

For Small Steam Line Breaks, the main steam pressures remain greater than 450 psig for the duration of the transient, per Oconee UFSAR Figure 15-169. Additionally, the RCS pressure remains greater than 1750 psig per Oconee UFSAR Figure 15-173; therefore, the CFTs would not discharge any fluid into the RCS during the Small Steam Line Break transient, and the proposed EOP changes will have no impact to the Small Steam Line Break safety analyses described in Oconee UFSAR Section 15.17.

References

1. Letter from Duke Energy (Ernest J. Kapopoulos, Jr.) to USNRC (Document Control Desk), "Duke Energy Carolina, LLC (Duke Energy): 10 CFR 50.46 – 30-day Report for Oconee Nuclear Station, Units 1, 2, and 3; Estimated Impacts to Peak Cladding Temperature due to Fuel Pellet Thermal Conductivity Degradation," dated December 17, 2014 (ML14353A214).

2. AREVA Topical Report BAW-10192P-A, Revision 0, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.

3. AREVA Topical Report BAW-10162P-A, Revision 0, "TACO3 – Fuel Pin Thermal Analysis Code," October 1989.

4. AREVA Topical Report BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.

5. Letter from R. A. Gramm (NRC) to J. S. Holm (Framatome ANP), Subject: Request for Amendment of Safety Evaluation for "Report of Preliminary Safety Concern (PSC) 2-00 Related to Core Flood Line Break with 2-Minute Operator Action Time," January 10, 2005. (TAC No. MA9973) (Accession Number ML043550355).