Kim Maza
Site Vice President
Harris Nuclear Plant
5413 Shearon Harris Rd
New Hill, NC 27562-9300

984-229-2512

10 CFR 50.90

April 23, 2020
Serial: RA-20-0117

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Subject: Supplemental Information for License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits

Ladies and Gentlemen:

By letter dated March 6, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20066L112), Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment would modify Technical Specification (TS) 3/4.2.5, “DNB Parameters,” and TS 6.9.1.6, “Core Operating Limits Report,” in support of analysis development for HNP Cycle 24. HNP TS 3/4.2.5 would be revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 would reflect the incorporation of the Framatome, Inc. (Framatome) topical report (TR) EMF-2103(P)(A), Revision 3, “Realistic Large Break LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors,” and the removal of analytical methods no longer applicable for the determination of HNP core operating limits.

During the Nuclear Regulatory Commission (NRC) staff’s acceptance review of the requested license amendment, the NRC staff determined that supplemental information was needed to enable the NRC staff to make an independent assessment regarding the acceptability of the proposed amendment. In response to the request for supplemental information, Duke Energy is submitting the enclosed additional information to support the acceptance review of the proposed amendment.

The content of this supplemental correspondence does not change the No Significant Hazards Consideration provided in the original submittal.

No regulatory commitments are contained in this letter.

Please refer any questions regarding this submittal to Art Zaremba, Manager – Nuclear Fleet Licensing, at (980) 373-2062.

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

Executed on April 23, 2020.

Sincerely,

Kim E. Maza

Kim Maza
Site Vice President
Harris Nuclear Plant

Enclosure: Supplemental Information for License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits

cc: L. Dudes, USNRC Region II – Regional Administrator
J. Zeiler, USNRC Senior Resident Inspector – HNP
T. Hood, USNRC NRR Project Manager – HNP
W. L. Cox, III, Section Chief, NC DHSR (NC)

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

Executed on April 23, 2020.

Sincerely,

Kim Maza
Site Vice President
Harris Nuclear Plant

Enclosure: Supplemental Information for License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits

cc: L. Dudes, USNRC Region II – Regional Administrator
J. Zeiler, USNRC Senior Resident Inspector – HNP
T. Hood, USNRC NRR Project Manager – HNP
W. L. Cox, III, Section Chief, NC DHSR (NC)

ENCLOSURE

SUPPLEMENTAL INFORMATION FOR LICENSE AMENDMENT REQUEST TO REDUCE THE MINIMUM REQUIRED REACTOR COOLANT SYSTEM FLOW RATE AND UPDATE THE LIST OF ANALYTICAL METHODS USED IN THE DETERMINATION OF CORE OPERATING LIMITS

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400
RENEWED LICENSE NUMBER NPF-63

8 PAGES INCLUDING THE COVER

Introduction

By letter dated March 6, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20066L112), Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment would modify Technical Specification (TS) 3/4.2.5, "DNB Parameters," and TS 6.9.1.6, "Core Operating Limits Report," in support of analysis development for HNP Cycle 24. HNP TS 3/4.2.5 would be revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 would reflect the incorporation of the Framatome, Inc. (Framatome) topical report (TR) EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors," and the removal of analytical methods no longer applicable for the determination of HNP core operating limits.

During the Nuclear Regulatory Commission (NRC) staff's acceptance review of the requested license amendment, the NRC staff determined that supplemental information was needed to enable the NRC staff to make an independent assessment regarding the acceptability of the proposed amendment. In response to the request for supplemental information, Duke Energy is submitting the enclosed additional information to support the acceptance review of the proposed amendment.

NRC Request

1. Provide a disposition of the effects of the proposed GAIA fuel assembly introduction on the Harris UFSAR [Updated Final Safety Analysis Report] Chapter 15 safety analyses, excepting the small-break and a large-break LOCA analyses already provided.

Duke Energy Response

HNP is transitioning from the High Thermal Performance (HTP) fuel assembly design to the GAIA fuel assembly design beginning in Cycle 24. ANP-10342NP-A, "GAIA Fuel Assembly Mechanical Design" (ADAMS Accession No. ML19309D916), is a TR describing the GAIA fuel assembly mechanical design, as approved by the NRC in a Safety Evaluation (SE) dated September 24, 2019.

From a UFSAR Chapter 15 Non-LOCA Analysis standpoint, the main changes from the HTP fuel design to the GAIA fuel design pertain to the fuel assembly spacer grids and loss coefficients. Refer to ANP-10342NP-A for a detailed description of the GAIA fuel assembly.

The differences between the HTP and GAIA fuel assembly designs are expected to have no significant impact on the Safety Analysis Physics Parameters (SAPPs) important to the UFSAR Chapter 15 Non-LOCA Analyses. According to the NRC-approved Duke Energy reload evaluation methodology DPC-NE-3009, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," (ADAMS Accession No. ML16278A080, Section 4.4), any SAPP violations calculated to occur for a given reload cycle will be resolved by analysis, evaluation, redesign, or a combination of these.

The differences between the HTP and GAIA fuel assembly designs were judged to have no significant impact on the SIMULATE-3K neutronic and thermal-hydraulic portion of the UFSAR Chapter 15.4.8 Spectrum of Rod Cluster Control Assembly (RCCA) Ejection Accidents ("REA")

Analysis. Accordingly, the SIMULATE-3K forcing functions calculated for a reference core with HTP fuel may be retained for partial or full GAIA reload cores. Key physics parameters important to the transient response (e.g., Doppler temperature coefficient, effective delayed neutron fraction, ejected rod worth, and moderator temperature coefficient) were adjusted to bound expected reload values. The analysis values are confirmed to remain bounding for each reload core. If the reload value of a key parameter exceeds the analysis value, then the approach described above will be used to resolve the difference.

The differences between the HTP and GAIA fuel assembly designs were judged to have no significant impact on the RETRAN-3D system thermal-hydraulic portions of the UFSAR Chapter 15 Non-LOCA Analyses. Accordingly, the RETRAN-3D boundary conditions calculated in an analysis with HTP fuel may be retained for partial or full GAIA reload cores. The RETRAN-3D analyses are performed using bounding values of key physics parameters. These values are confirmed to remain bounding for each reload core. If the reload value of a key parameter exceeds the analysis value, then the approach described above will be used to resolve the difference.

Some of the differences between the HTP and GAIA fuel assembly designs do impact the VIPRE-01 core thermal-hydraulic portions of the UFSAR Chapter 15 Non-LOCA Analyses. These impacts will be evaluated by reanalyzing selected VIPRE-01 calculations to produce departure-from-nucleate-boiling ratio (DNBR) limits applicable to the HTP and GAIA fuel assembly designs, accounting for mixed-core effects as needed. The resulting limits will be used in cycle-specific reload calculations to confirm that the analysis acceptance criteria are met. The selection of VIPRE-01 calculations for reanalysis is described below.

The differences between the HTP and GAIA fuel assembly designs are expected to have no significant impact on the Radiological Dose portions of the UFSAR Chapter 15 Non-LOCA Analyses. The differences will be evaluated or analyzed to ensure that the radiological dose consequences meet the acceptance criteria.

Based on the preceding discussion, the UFSAR Chapter 15 Non-LOCA Analyses were divided into two categories for disposition. For simplicity, the events were categorized based only on the need for VIPRE-01 analysis or reanalysis. Radiological dose analyses or evaluations will also be completed as noted above, but were not used to categorize the events.

<u>Category 1 – No VIPRE-01 Analysis or Reanalysis Required</u>

Table 1.1 lists the events and evaluation bases for Category 1. Some of the Category 1 events had non-limiting VIPRE-01 calculations that did not require cycle-specific monitoring. The remaining Category 1 events do not require VIPRE-01 calculations.

<u>Category 2 – VIPRE-01 Analysis or Reanalysis Required</u>

Table 1.2 lists the events and evaluation bases for Category 2. Relative to HTP fuel, the mixed core is expected to cause a relatively minor decrease in minimum DNBR (MDNBR) results and Maximum Radial Peaking Factor (MARP) limits, and the GAIA fuel is expected to provide an increase in MDNBR values and MARP limits. Section 3.1.3 of the original LAR identified three UFSAR Chapter 15 Non-LOCA Analyses as being reanalyzed using Duke Energy methods and

accounting for the proposed decrease in the TS minimum RCS flow rate. These events are included in Category 2.

Summary

The differences between the HTP and GAIA fuel assembly designs are expected to have no significant impact or provide a beneficial impact on most aspects of the UFSAR Chapter 15 Non-LOCA Analyses. The mixed core is expected to cause a relatively minor penalty in DNB results for HTP fuel assemblies. Confirmation of the expected effects and demonstration of positive analytical margin will be documented as part of the reload design process.

Table 1.1 – Category 1 Events – No VIPRE-01 Analysis or Reanalysis Required

Section	Title	Evaluation
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	A
15.1.3	Excessive Increase in Secondary Steam Flow	B
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	A
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	D
15.2.2	Loss of External Electrical Load	A
15.2.3	Turbine Trip	B
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	A
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	A
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	A
15.2.8	Feedwater System Pipe Break	B
15.3.1	Partial Loss of Forced Reactor Coolant Flow	A
15.3.4	Reactor Coolant Pump Shaft Break	A
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	F
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	D
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	C
15.4.9	Spectrum of Rod Drop Accidents in a BWR	D
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	C
15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	A
15.5.3	A Number of BWR Transients	D
15.6.2	Break in Instrument Line or Other Line From Reactor Coolant Pressure Boundary that Penetrate Containment	E
15.6.4	Spectrum of BWR Steam System Piping Failures Outside Containment	D
15.7.1	Radioactive Waste Gas System Leak or Failure	E
15.7.2	Liquid Waste System Leak or Failure	E
15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failure	E
15.7.4	Design Basis Fuel Handling Accidents	E
15.7.5	Spent Fuel Cask Drop Accidents	E
15.8	Anticipated Transients Without Scram	C

Table 1.1 – Category 1 Events – No VIPRE-01 Analysis or Reanalysis Required (Continued)

Evaluations

A. This event is bounded by another event as described in the LAR.
B. This event had non-limiting VIPRE-01 results that did not require cycle-specific monitoring.
C. This event does not require VIPRE-01 calculations.
D. This event does not apply to HNP.
E. This event does not involve a Nuclear Steam Supply System (NSSS) transient.
F. This event is not credible during power operation; no analysis necessary at zero power operation.

Table 1.2 – Category 2 Events – VIPRE-01 Analysis or Reanalysis Required

Section	Title	Evaluation
15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	G
15.1.5	Steam System Piping Failure	H
15.2.7	Loss of Normal Feedwater Flow	G
15.3.2	Complete Loss of Forced Reactor Coolant Flow	I
15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	I
15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	I
15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	I
15.4.3.1	Dropped Full Length RCCA or RCCA Bank	I
15.4.3.2	Withdrawal of a Single Full Length RCCA	I
15.4.3.3	Statically Misaligned RCCA or RCCA Bank	J
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	J
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	K
15.6.1	Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve	I
15.6.3	Steam Generator Tube Rupture	I

Evaluations

G. These events include new analyses with non-limiting VIPRE-01 calculations that will be evaluated for the mixed core and GAIA.
H. This event includes a reanalysis (HZP) and a new analysis or evaluation (HFP) for DNB response.
I. These events include reanalysis or evaluation of the event-specific MARP limits.
J. These events include reanalysis of the Nominal MARP limits.
K. This event includes reanalysis or evaluation of the event-specific MARP limits and the limiting Fuel Temperature and Enthalpy calculation.

NRC Request

2. Provide information justifying the implementation of ANP-10342NP-A for batch loading of the GAIA fuel assembly design.

Duke Energy Response

The Safety Evaluation (SE) for TR ANP-10342NP-A concludes in Section 5.0: "Therefore, on the basis of the above review and justification, the NRC staff concludes that the GAIA fuel assembly design is acceptable for use in Westinghouse three-loop and four-loop design reactors which use a 17 x 17 fuel rod array with LEU [low enrichment uranium] fuel subject to the L&Cs [limitations and conditions] included in this SE." HNP is a Westinghouse three-loop design reactor utilizing 17 x 17 fuel rod arrays with LEU fuel. The means in which HNP is subject to the L&Cs included in TR ANP-10342NP-A is addressed in the response to NRC Request #3 below.

NRC Request

3. Provide information to indicate how the licensee has satisfied the limitations and conditions associated with the SE approving ANP-10342NP-A, including particularly a rod control cluster assembly ejection accident that is consistent with Limitation 5.[2]

   [2] The NRC staff notes that Section 3.5.21.1 of the SE approving Amendment No. 164 to the Harris Operating License (ADAMS Accession No. ML18060A401) discusses rod control cluster assembly ejection accident acceptance criteria that are consistent with those envisioned in limitations and conditions 5 of ANP-10342NP-A.

Duke Energy Response

The referenced L&Cs are provided in Section 4.0 of the SE for ANP-10342NP-A, and are as follows:

1) This GAIA fuel assembly design is approved for use with low enrichment uranium (LEU) fuel, which has been enriched to less than or equal to 5 percent.
2) The GAIA fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 Megawatt-days/metric ton of Uranium.
3) The final LTA [lead test assembly] program PIE [post-irradiation examination] report shall be submitted to NRC staff prior to any reload batch of GAIA assemblies reaching the third cycle of operation.
4) (Removed)
5) As part of the plant-specific LAR implementing GAIA, the licensee must demonstrate acceptable performance of GAIA under RIA [reactivity initiated accident] conditions, including fuel damage, coolable geometry, and radiological consequences, using approved methods. Current guidance and analytical limits are found in SRP 4.2 Appendix B. Newer guidance is expected soon (e.g., DG-1327). The licensee should consider the most up-to-date guidance and analytical limits at the time of submittal. Alternative means to demonstrate compliance will be considered on a case-by-case basis.

As it relates to L&C #1, HNP TS 5.3.1 states in part: "Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235." As such, HNP TS 5.3.1 ensures that HNP meets L&C #1.

As it relates to L&C #2, the NRC approved HNP's application of the COPERNIC computer code per license amendment No. 171 (ADAMS Accession No. ML19288A139). Per the SE for the COPERNIC code (ADAMS Accession No. ML021360461), it "is acceptable for fuel licensing applications up to a rod average burnup of 62 Gwd/MTU." As discussed in Section 3.1 of the SE for HNP's application of the code, "the 62 GWd/MTU burnup limit will be verified as part of the normal reload design process." Per the approved methodology for the site, HNP meets L&C #2.

L&C #3 is related to Framatome sending in the final LTA program PIE report to NRC staff, prior to any reload batch of GAIA assemblies reaching the third cycle of operation. With introduction of reload batches in Spring 2021, this report would need to be sent to the NRC prior to Spring 2024. This L&C does not preclude the ability of HNP to utilize GAIA fuel at this time.

L&C #4 was removed from the final SE associated with ANP-10342NP-A.

As it relates to L&C #5, the NRC approved Duke Energy methodology report DPC-NE-3009, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," for application specific to HNP per letter dated April 10, 2018 (ADAMS Accession No. ML18060A401), allowing for the performance of reactor safety analyses as part of the core reload design process. DPC-NE-3009 performs the RIA analysis per the criteria in SRP 4.2 Appendix B. At the time of NRC approval of DPC-NE-3009, Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," had not yet been approved. However, the NRC asked in request for additional information (RAI) 43 of the DPC-NE-3009 submittal to specify the acceptance criteria being used and describe how the evaluation method was applicable to the selected acceptance criteria for rod control cluster assembly (RCCA) ejection accidents. Duke Energy responded per letter dated October 30, 2017 (ADAMS Accession No. ML17303B209), to which the NRC found the response to be acceptable as documented in Section 3.5.21.1 of the SE for DPC-NE-3009. The reanalysis/evaluation approach described in the response to NRC Request #1 above for the UFSAR Chapter 15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents event will use the NRC-approved methodology described in DPC-NE-3009. Therefore, Duke Energy complies with L&C #5.