Tanya M. Hamilton
Vice President
Harris Nuclear Plant
5413 Shearon Harris Road
New Hill, NC 27562

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390
UPON REMOVAL OF ATTACHMENTS 2 AND 6, THIS LETTER IS DECONTROLLED

10 CFR 50.90

April 10, 2019
RA-19-0017

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: License Amendment Request to Modify the Departure from Nucleate Boiling Ratio Safety Limit to Address Transition to New Fuel Type

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), is submitting a request to the Nuclear Regulatory Commission (NRC) for an amendment to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment modifies TS 2.1.1.a to add the departure from nucleate boiling ratio safety limit associated with the transition to the GAIA fuel design from the current HTP fuel. In addition, TS 6.9.1.6.2 will be revised to include the topical report for the NRC-approved correlation associated with the fuel design transition. Duke Energy also requests the review and approval of Revision 6 of DPC-NE-2005-P, “Thermal-Hydraulic Statistical Core Design Methodology,” for the addition of Appendix J addressing the application of the ORFEO-GAIA critical heat flux correlation.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been concluded that the proposed changes involve no significant hazards consideration. Attachment 1 of this license amendment request provides the affidavits from Duke Energy and Framatome, Inc. (Framatome), supporting the request for withholding information in Attachments 2 and 6 from public disclosure. Attachments 2 and 3 (proprietary and non-proprietary, respectively) provide Duke Energy’s evaluation of the proposed changes. Attachment 4 provides a copy of the proposed TS changes. Attachment 5 provides a copy of the TS Bases markup based on the proposed changes (for information only). Attachments 6 and 7 provide the proprietary and non-proprietary versions of the proposed Appendix J to Duke Energy methodology report DPC-NE-2005-P-A associated with the application of the ORFEO-GAIA correlation to the GAIA fuel design.

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390
UPON REMOVAL OF ATTACHMENTS 2 AND 6, THIS LETTER IS DECONTROLLED

Approval of the proposed license amendment is requested within twelve months of acceptance. The amendment shall be implemented prior to the startup of Cycle 24.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated North Carolina State Official.

This document contains no new Regulatory Commitments.

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

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

Executed on April 10, 2019.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Attachments:
1. Affidavits for Withholding of Proprietary Information (Duke Energy and Framatome)
2. Evaluation of the Proposed Change [Proprietary]
3. Evaluation of the Proposed Change [Non-Proprietary]
4. Proposed Technical Specification Changes
5. Proposed Technical Specification Bases Changes (For Information Only)
6. Application of the ORFEO-GAIA Correlation to the GAIA Fuel Design [Proprietary]
7. Application of the ORFEO-GAIA Correlation to the GAIA Fuel Design [Non-Proprietary]

cc: J. Zeiler, NRC Senior Resident Inspector, HNP
W. L. Cox, III, Section Chief N.C. DHSR
M. Barillas, NRC Project Manager, HNP
C. Haney, NRC Regional Administrator, Region II

Approval of the proposed license amendment is requested within twelve months of acceptance. The amendment shall be implemented prior to the startup of Cycle 24.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated North Carolina State Official.

This document contains no new Regulatory Commitments.

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

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

Executed on April , 2019.

Sincerely,

Tanya M. Hamilton

Attachments:
1. Affidavits for Withholding of Proprietary Information (Duke Energy and Framatome)
2. Evaluation of the Proposed Change [Proprietary]
3. Evaluation of the Proposed Change [Non-Proprietary]
4. Proposed Technical Specification Changes
5. Proposed Technical Specification Bases Changes (For Information Only)
6. Application of the ORFEO-GAIA Correlation to the GAIA Fuel Design [Proprietary]
7. Application of the ORFEO-GAIA Correlation to the GAIA Fuel Design [Non-Proprietary]

cc: J. Zeiler, NRC Senior Resident Inspector, HNP
W. L. Cox, III, Section Chief N.C. DHSR
M. Barillas, NRC Project Manager, HNP
C. Haney, NRC Regional Administrator, Region II

RA-19-0017

ATTACHMENT 1

AFFIDAVITS FOR WITHHOLDING OF PROPRIETARY INFORMATION
(DUKE ENERGY AND FRAMATOME)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

6 PAGES PLUS COVER

AFFIDAVIT of Steve Snider

1. I am Vice President of Nuclear Engineering, Duke Energy Corporation, 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 Attachments 2 and 6 to Duke Energy license amendment request letter RA-19-0017 regarding application to revise technical specifications and the corresponding impact to report DPC-NE-2005-P-A, *Thermal-Hydraulic Statistical Core Design Methodology*.

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, analysis results, supporting data, and aspects of development programs, relative to a method of analysis 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 is that which is marked in Attachments 2 and 6 to Duke Energy letter RA-19-0017 regarding application to revise technical specifications and the corresponding impact to report DPC-NE-2005-P-A, *Thermal-Hydraulic Statistical Core Design Methodology*. This information enables Duke Energy to:

(a) Support license amendment requests for its Harris reactor.

(b) Support reload design calculations for Harris reactor cores.

(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.

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 April 4, 2019.

Steve Snider

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Nathan E. Hottle. I am Manager, Product Licensing, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the following document: Duke Energy Letter RA-19-0017, "License Amendment Request to Modify the Departure from Nucleate Boiling Ratio Safety Limit to Address Transition to New Fuel Type," Attachments 2 and 6, referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(a) and 6(d) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 21

day of February, 2019.

Heidi Hamilton Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH OF VIRGINIA
REGISTRATION NO. 7777873
MY COMM. EXPIRES: 12/31/2022
NOTARY PUBLIC

RA-19-0017

ATTACHMENT 3

EVALUATION OF THE PROPOSED CHANGE [NON-PROPRIETARY]

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

18 PAGES PLUS COVER

Evaluation of the Proposed Change

Subject: License Amendment Request to Modify the Departure from Nucleate Boiling Ratio Safety Limit to Address Transition to New Fuel Type

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), is submitting a request to the Nuclear Regulatory Commission (NRC) for an amendment to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment modifies TS 2.1.1.a to add the departure from nucleate boiling ratio safety limit associated with the transition to the GAIA fuel design from the current HTP fuel. In addition, TS 6.9.1.6.2 will be revised to include the topical report for the NRC-approved correlation associated with the fuel design transition. Duke Energy also requests the review and approval of Revision 6 of DPC-NE-2005-P, "Thermal-Hydraulic Statistical Core Design Methodology," for the addition of Appendix J addressing the application of the ORFEO-GAIA critical heat flux correlation.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

As documented in the HNP TS Bases:

> The restrictions of the reactor core safety limits prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.
>
> Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in the heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and reactor coolant temperature and pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB...
>
> The restrictions of this safety limit also prevent fuel centerline melting. Fuel centerline melting occurs when the local linear heat rate (LHR), or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The fuel centerline temperature limit is a function of weight percent of Gadolinia and pin burnup...
>
> The safety limit figure provided in the Core Operating Limits Report (COLR) shows the loci of points of Fraction of Rated Thermal Power, Reactor Coolant System (RCS) Pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit,

that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, and that the exit quality is within the limits defined by the DNBR correlation. The reactor core safety limits are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criteria) that the hot fuel rod in the core does not experience DNB; and
b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core safety limits are used to define the various Reactor Protection System (RPS) functions such that the above criteria are satisfied during steady state operation and Condition I and II events. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Over Temperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the thermal power, RCS pressure, RCS average temperature, RCS flow rate, and ΔI (flux difference) that the reactor core safety limits will be satisfied during steady state operation and Condition I and II events.

2.2 Current Technical Specification Requirements

HNP TS 2.1.1.a currently addresses the DNBR safety limit associated with the HTP DNB correlation, ensuring a value of greater than or equal to 1.141 is maintained.

HNP TS 6.9.1.6.2.j documents the current correlation methodology associated with HTP fuel.

The current Duke Energy Statistical Core Design methodology report DPC-NE-2005-P-A, Revision 5, contains NRC-approved appendices which currently apply the methodology to McGuire, Catawba, and Oconee nuclear stations, and Robinson and Harris nuclear plants. The most recent Appendices H and I were approved by the NRC in Reference 6.2.

2.3 Reason for the Proposed Change

HNP is transitioning from the 17x17 HTP fuel design to the GAIA fuel design starting in Cycle 24 and continuing to full core GAIA fuel in Cycle 26. The changes necessary to implement GAIA fuel into Duke Energy's approved methods include:

- Validation of the ORFEO-GAIA critical heat flux correlation used with the VIPRE-01 thermal hydraulic code,
- Calculation of a safety limit for Duke Energy's implementation of the ORFEO-GAIA correlation and inclusion in TS 2.1.1.a,
- Calculation of a statistical core design limit for GAIA fuel using the ORFEO-GAIA correlation,
- Development of a mixed core penalty to apply to analysis results for cores containing both 17x17 HTP and GAIA fuel.

Additionally, relevant sections of Request for Additional Information (RAI) #22 from the NRC's review of Duke Energy methodology DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis" (Reference 6.3), concerning mixed core applications are addressed in this technical justification.

2.4 Description of the Proposed Change

The proposed change requests NRC approval to modify the HNP TS 2.1.1.a to incorporate the DNBR limit for the GAIA fuel utilizing the ORFEO-GAIA DNB correlation. Specifically, HNP TS 2.1.1.a would be revised to read as follows:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation for HTP fuel and ≥ 1.12 for the ORFEO-GAIA DNB correlation for GAIA fuel.

In addition, the proposed change requests review and approval of Appendix J to DPC-NE-2005-P, to be incorporated as Revision 6 of the methodology report. This appendix addresses the extended applicability of the methodology to the GAIA fuel design at HNP.

HNP TS 6.9.1.6.2 will be updated to reflect the addition of the ORFEO-GAIA correlation. Because the current HNP TSs are consistent with TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR," inclusion of revision dates for the NRC-approved topical report in the TS is not required, which is also consistent with NUREG-1431, Revision 4, "Standard Technical Specifications – Westinghouse Plants" (ADAMS Accession No. ML12100A222). The proposed change also removes excess unused space on HNP TS Page 6-24a that is editorial in nature and does not impact the content currently provided under TS 6.9.1.6.2. As a result of this change, there will not be the need to create a new TS page to accommodate the addition of the ORFEO-GAIA methodology to TS 6.9.1.6.2.

HNP Tech Spec Bases will be updated to reflect the addition of the DNBR limit for the GAIA fuel utilizing the ORFEO-GAIA DNB correlation.

3.0 TECHNICAL EVALUATION

3.1 GAIA Description

3.1.1. GAIA Specifications and Characteristics

The Framatome, Inc. (Framatome) GAIA fuel assembly design intended for use at HNP is a 17x17 rod design using M5 cladding and Q12 instrument and guide tubes. The assembly contains GAIA mixing grids and mid-span-mixing grids referred to as intermediate GAIA mixers (IGM), both made of M5. The top and bottom non-mixing grids are made of Ni Alloy. Table 3.1 shows the GAIA fuel specifications along with the values for the current 17x17 HTP fuel assembly.

Table 3.1 Fuel Specifications for 17x17 HTP and First Batch GAIA (nominal)

	GAIA	17x17 HTP
Fuel Assembly:		
Array (W15x15, CE16x16, BW15x15, etc.)	W 17x17	
Assembly Pitch (in.)	8.466	
Rod Pitch (in.)	0.496	
Incore Entry Direction (from Top, Bottom)	Bottom	
Fuel Rod:		
Quantity per Assembly	264	
Fuel Column Length (in.)	144	
Clad Material	M5	
End Cap Material	Zr-4	
Clad Outside Diameter (in.)	0.374	0.376
Clad Inside Diameter (in.)	0.329	0.328
Pellet Diameter (in.)	0.3225	0.3215
Pellet Material (Doped, BLEU, MOX, etc.)	UO_2 & UO_2+Gd_2O_3	
Burnable Poison Material (Gad, B_4C, etc.)	Gd_2O_3	
Pellet Density (UO_2) (% Theo. Density)	[ ]F	96
Plenum Location (upper and/or lower)	Upper	
Structure or Cage:		
Type (Welded or Floating Grid)	Welded	
Guide Tube:		
Quantity per Assembly	24	
Type (swaged, MONOBLOC, non-dashpot, etc.)	MONOBLOC	
Material	Q12	Zr-4
Top Outside Diameter (in.)	[ ]F	0.482
Top Inside Diameter (in.)	0.451	0.450
Bottom (Dashpot) Outside Diameter (in.)	[ ]F	0.482
Bottom (Dashpot) Inside Diameter (in.)	0.397	
Length from bottom of guide tube end fitting to start of dashpot transition (in.)	[ ]F	[ ]F
Instrument Tube:		
Quantity per Assembly	1	
Type (constant diameter, MONOBLOC, dimpled)	Constant diameter	
Material	Q12	Zr-4
Top Outside Diameter (in.)	[ ]F	0.482
Top Inside Diameter (in.)	0.451	0.450
Bottom Outside Diameter (in.)	N/A	
Bottom Inside Diameter (in.)	N/A	

Grids:		
Quantity per Assembly	11	
Top End Grid Type	Relaxed HMP	HTP
Quantity per Assembly	1	1
Material	Nickel Alloy	Zr-4
Intermediate (Spacer) Grid Type	GAIA Mixing Grid	HTP
Quantity per Assembly	6	6
Material	M5	Zr-4
Intermediate Mixing Grid Type	IGM	IFM
Quantity per Assembly	3	3
Material	M5	Zr-4
Bottom End Grid Type	HMP	
Quantity per Assembly	1	
Material	Nickel Alloy	
Bottom Nozzle	GRIP	FUELGUARD
Quantity per Assembly	1	1
Material	Stainless Steel	

3.1.2. Lead Test Assembly (LTA) Program

A lead test assembly (LTA, or lead fuel assembly, LFA) program is ongoing at HNP, having started in cycle 20. Eight GAIA LTAs were introduced in the cycle 20 core in non-limiting locations. These 8 assemblies were included in the cycle 21 design, and 4 of them remained in the cycle 22 design. No problems have been encountered in terms of mechanical, thermal-hydraulic, or neutronic compatibility with the other HTP assemblies.

An analysis was performed prior to the beginning of the LTA program to determine if any hydraulic dissimilarities existed between GAIA and HTP fuel that would adversely affect the performance of the reactor core in full core (GAIA or HTP) or LTA mixed core configurations. Table 3.2 summarizes the thermal hydraulic compatibility of GAIA and HTP fuels in full and LTA mixed core configurations from this analysis and concludes that there are no expected issues that would cause adverse effects to the operation of the plant.

Table 3.2 - GAIA / 17x17 HTP Thermal Hydraulic Compatibility

Core Pressure Drop	[ ]^{F} psid for HTP versus [ ]^{F} psid for GAIA
Control Rod Drop Time	No adverse effect due to similar geometries
DNB Performance	Assembly inlet flow diverts to GAIA fuel due to the lower assembly pressure drop, resulting in lower DNBR for HTP fuel
Guide Tube Flow and Heating	No adverse effect due to similar geometries

Reactor Coolant System Loop Flow	The RCS loop flow is expected to [ ]F once a full core of GAIA fuel is loaded. This will have no adverse effect on Technical Specifications.
Total Bypass Flows	There is no change in bypass flow as a result of 8 GAIA LTAs
Hydraulic Instability	No adverse effects due to open-lattice fuel assemblies

The GAIA fuel assembly for full batch implementation differs minimally from the LTA GAIA assemblies evaluated previously. The changes include minor dimensional changes associated with industrialization and the [ ]F. These changes negligibly impact the core pressure drop but improve sub-channel mixing performances for subsequent DNB analyses. These changes are captured in updates to the [ ]D VIPRE-01 model used to generate the statistical design limit (SDL). Therefore, the conclusions tabulated above remain valid for full batch implementation.

3.2 ORFEO-GAIA Critical Heat Flux (CHF) Correlation

3.2.1. Correlation Description

Reference 6.1 details the ORFEO-GAIA CHF correlation developed by Framatome for use with fuel equipped with GAIA mixing vane grids. Tests were performed at Framatome's KATHY test loop in Karlstein, Germany. The qualification of KATHY as an acceptable source of CHF test data for CHF correlation development was provided with the ACH-2 CHF correlation.

3.2.2. NRC Review and Approval of Vendor Topical Report

The NRC reviewed and approved topical report ANP-10341P-A, "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations," (Reference 6.1), which contains the ORFEO-GAIA and ORFEO-NMGRID correlations on September 24, 2018. This approval was given to Framatome and is subject to the following conditions and limitations (Table and Figure numbers refer to Reference 6.1):

1. The inlet subcooling must be greater than 0 degrees. This is to ensure that the burnout length is limited to the fuel region.
2. For ORFEO-NMGRID, Framatome should confirm that the reload calculation performed for set points, anticipated operational occurrences, and accidents are far removed from the [ ]F subregion. If the calculations are not far removed from this region, then Framatome must quantify the additional uncertainty of the region and apply that increased uncertainty in the analysis.
3. While both ORFEO-GAIA and ORFEO-NMGRID are approved over their entire application domain, this approval is given under the assumption that their use in the low quality region (i.e. equilibrium qualities below -0.1) has minimal impact on the limiting minimum DNBR values. Limiting minimum DNBR is defined as the scenario in which the event is approaching the design limit. Application of the ORFEO-GAIA and ORFEO-NMGRID CHF correlations for events in which the limiting DNBR is sufficiently far from the design limit is

not subject to this condition regardless of the local quality. Should this assumption no longer be true and should the low quality domain become a limiting domain, Framatome would need to provide additional analysis in quantifying the uncertainty in this domain.

4. ORFEO-GAIA is approved for use in predicting the CHF downstream of GAIA mixing vane spacers in GAIA fuel. This prediction must be made in the subchannel code COBRA-FLX with the modeling option as specified in Table 5.1 of the topical report with a design limit of 1.12 over the application domain specified in Table 2-2 of the initial submittal of the topical report. The approved design limit contains a bias of 0.01 which the NRC staff believed was necessary to account for variations between the tested fuel assembly and the production fuel assembly which will be used in the reactor.
5. ORFEO-NMGRID is approved for use in predicting the CHF downstream of W 17x17 HMP non-mixing grids and GAIA mixing vane spacers in GAIA fuel. This prediction must be made in the subchannel code COBRA-FLX with the modeling option as specified in Table 5.1 of the topical report with a design limit of 1.15 over the application domain specified in Table 2-5 of the initial submittal of the topical report.

The same limits and conditions will apply to Duke Energy's use of the above correlations with VIPRE-01 as follows:

1. Duke Energy will adhere to this limitation.
2. Duke Energy intends to use the ORFEO-NMGRID correlation in the non-mixing span only and the ORFEO-GAIA correlation for the remainder of the assembly, similar to how the BWU-N CHF correlation is used in the non-mixing span with the HTP CHF correlation. DNB has not been observed in the non-mixing span region of the fuel assembly and is not expected in future applications.
3. Duke Energy will adhere to this limitation.
4. Duke Energy will use ORFEO-GAIA with VIPRE-01 as proposed in this amendment request.
5. Duke Energy intends to use the ORFEO-NMGRID correlation in the non-mixing span only and the ORFEO-GAIA correlation for the remainder of the assembly, similar to how the BWU-N CHF correlation is used in the non-mixing span with the HTP CHF correlation. DNB has not been observed in the non-mixing span region of the fuel assembly and is not expected in future applications.

The modeling options used in VIPRE-01 are inherent in the correlation validation described in Section 3.2.3.

3.2.3. ORFEO-GAIA Validation with VIPRE-01

The same approach used in Reference 6.1 was followed by Duke Energy in implementing and validating the ORFEO-GAIA correlation with VIPRE-01. VIPRE-01 is a thermal hydraulic subchannel analysis code that Duke Energy uses in its current approved methodologies (References 6.2 and 6.4). The same test data set consisting of [ ]^{F} data points contained in Reference 6.1 was utilized with models representing the test geometry for each subset. The proposed Appendix J of DPC-NE-2005-P, included as Attachment 6 (proprietary) and

Attachment 7 (non-proprietary) of this submittal, contains details and results of Duke Energy's evaluation of the ORFEO-GAIA correlation with VIPRE-01. The conclusion is that a safety limit of 1.12, which is the higher limit of the VIPRE-01 and COBRA-FLX limits, with a standard deviation of [ ]^{D} is applicable.

None of the [ ]^{F} data points predicted DNB in the non-mixing span (below the first GAIA grid). The ORFEO-GAIA implementation in VIPRE-01 follows the application provided in Reference 6.1 and uses the ORFEO-NMGRID correlation in the non-mixing span similar to how the BWU-N CHF correlation is used in the non-mixing span with the HTP CHF correlation. Therefore, Duke Energy's use of ORFEO-NMGRID in this span is considered acceptable.

3.3 Statistical Design Limit (SDL) for GAIA Fuel

3.3.1. Statistical Core Design Methodology

The Duke Energy methodology for statistical core design contained in the methodology report DPC-NE-2005-P-A, as approved for use by HNP by the NRC per Reference 6.2, is utilized to calculate a SDL which accounts for uncertainties on key parameters (pressure, temperature, flow, thermal power). The full details of the results of the GAIA SDL calculation are contained in the proposed Appendix J of DPC-NE-2005-P, included in this submittal as Attachment 6 (proprietary) and Attachment 7 (non-proprietary). Pending NRC approval, Appendix J will be incorporated in Revision 6 of the methodology report.

3.3.2. VIPRE-01 [ ]^{D} Model

The model used to evaluate the SDL is a [ ]^{D} GAIA fuel model based on the [ ]^{D} HTP model contained in Appendix I of DPC-NE-2005-P-A. The dimensions were updated per the specifications contained in Table 3.1. Fuel specific inputs for CHF correlations, mixing coefficients, and pressure loss coefficients were updated as well. The flow and heat transfer correlations (except CHF correlation) were kept the same. The model is further detailed in the proposed Appendix J to the methodology report, included in this submittal as Attachment 6 (proprietary) and Attachment 7 (non-proprietary).

3.3.3. Range of Applicability and Uncertainties

The calculated SDL is applicable for the range of conditions spanned by the state points analyzed. The uncertainties in plant parameters that are used for the GAIA SDL calculation are the same or larger than what were used for the HTP analysis. Larger uncertainties are more conservative and allow for more margin between actual and calculated values. The uncertainties and applicability range are given in Table J-5 and J-7, respectively, in Attachment 6 (proprietary) and Attachment 7 (non-proprietary).

3.4 Mixed Core Penalty (HTP and GAIA Fuel)

3.4.1. Penalty Derivation

To ensure adequate margin in the Final Safety Analysis Report Chapter 15 transients and accidents, maximum allowed peaking limits to preclude DNBR are calculated and applied to cycle specific core designs using Duke Energy's approved methodology (Attachment 9 of Reference 6.5, as approved by the NRC for reference by HNP in Reference 6.6). Peaking limits are generated for various transients as described in Duke Energy methodology report DPC-NE-

3009-P-A, “FSAR/UFSAR Chapter 15 Transient Analysis Methodology” (Reference 6.4). Peaking limits have been calculated for a full core of HTP fuel. Limits will also be calculated for a full core of GAIA fuel following the same method (Attachment 9 of Reference 6.5) used to calculate the full core HTP limits. As indicated in Table 3.2 above, coolant flow is diverted from an HTP assembly into a GAIA assembly in a mixed core configuration thereby incurring a DNBR penalty for HTP fuel. For the application of peaking limits to HTP fuel assemblies in cores consisting of mixed HTP and GAIA fuel, a penalty can be derived using either a [ ]^{D} mixed core model or a [ ]^{D} mixed core VIPRE-01 model (see Section 3.4.2 below). The peaking limits for a range of transients were recalculated with the mixed core model and compared to the full core HTP peaking limits, and a bounding set of penalties were derived based on the differences. There is no mixed core penalty for GAIA fuel during the transition from a full core of HTP fuel to a full core of GAIA fuel due to the increased flow through the GAIA assembly. Consequently, in a mixed core configuration with co-resident HTP fuel, cycle specific peaking in GAIA fuel assemblies will be compared to the maximum allowed peaking limits applicable to a full core of GAIA fuel.

3.4.2. Mixed Core VIPRE-01 Models

A [ ]^{D} mixed core VIPRE-01 model was created in order to determine the mixed core penalty. This model is the same as the previously discussed HTP and GAIA [ ]^{D} models, except the hot assembly is assumed to be HTP fuel and the surrounding lumped channel ([ ]^{D}) is GAIA fuel. Due to the assembly grid form loss coefficients for each fuel type, this results in a conservative flow arrangement by providing less flow to the hot assembly channels.

A [ ]^{D} mixed core VIPRE-01 model has also been constructed. The model assumes the assembly modeled in subchannel detail is an HTP assembly and the assemblies immediately surrounding it are a checkerboard of HTP and GAIA assemblies as would be found in the first transition core. The model is used to compare the results of the mixed core [ ]^{D} model and address applicable portions of RAI #22 associated with DPC-NE-3008-P-A (see Section 3.4.4 below) in anticipation of using the [ ]^{D} model for cycle specific analyses, with the possibility of being used to calculate the mixed core penalty. The comparison to the [ ]^{D} model results confirmed that the [ ]^{D} mixed core model calculates a conservative maximum allowable peaking limit relative to the [ ]^{D} mixed core model. A full core [ ]^{D} GAIA VIPRE-01 model has also been created similar to that described in Section 5 of DPC-NE-3008-P-A and will be used for GAIA fuel in either a full core or a mixed core application with co-resident HTP fuel.

3.4.3. Mixed Core SDL

The analysis verified that the SDL used for HTP fuel is the same for mixed core conditions. The same process was used to calculate the SDL except the [ ]^{D} model representing an HTP fuel assembly surrounded by GAIA fuel was used to verify that, for the same conditions, the SDL remains the same or is bounded by the full core HTP value. The analysis concluded that the SDL remains the same, making the mixed core penalty derived following the method described in Section 3.4.1 the only necessary conservatism for applying HTP peaking limits to HTP assemblies in a mixed core configuration.

3.4.4. DPC-NE-3008-P-A RAI #22

During review of DPC-NE-3008-P-A, the NRC asked for additional information concerning the use of the expanded VIPRE-01 models. In its response, Duke Energy specified that the application of the models to mixed core cases would be submitted separately for NRC review and approval prior to implementation. This section details the content related to the parts of RAI #22 concerning mixed core applications.

RAI-22a

Question 22a asked how the expanded VIPRE-01 model will be used by Duke Energy. The expanded model can be used to calculate the mixed core penalty following the method described in Section 3.4.1. The expanded model is also used to validate that the DNBR predictions using the mixed core [ ]^{D} model are conservative as discussed in Section 3.4.2 above. It will also be used, if needed, for cycle specific analyses as described in the original response and in Section 3.4.2 above following NRC approval of the response to RAI 22e below.

RAI-22e

Question 22e pertains to how Duke Energy determined and validated crossflow and turbulent mixing coefficients for fuel assemblies from different fuel vendors. Turbulent mixing is a subchannel phenomenon and is conservatively assumed not applicable to fuel assembly boundaries or lumped channels. The VIPRE-01 energy and momentum equations contain terms that describe the exchange of energy and momentum between adjacent subchannels due to turbulent mixing. These terms are the turbulent momentum factor (FTM) and the turbulent mixing coefficient (ABETA). FTM specifies how efficiently the turbulent crossflow mixes momentum. It is specified on a scale from 0.0 to 1.0, where 0.0 specifies that the turbulent crossflow mixes enthalpy only and not momentum and 1.0 specifies that momentum and enthalpy are mixed with the same strength. Duke Energy conservatively assumes a value of 0.0 for DNBR analysis. The ABETA coefficient is a fuel type specific value provided by the fuel vendor. Higher values result in more smoothing of the hot channel flow conditions, i.e. a decrease in enthalpy and increase in flow rate and DNBR. Lower values are therefore more conservative. For the subchannels in the expanded VIPRE-01 model that are located on the boundary between GAIA and HTP fuel assemblies, the turbulent mixing coefficient specified is the average of the GAIA and HTP values. Since the mixing coefficients for GAIA and HTP are similar and from the same fuel vendor, there is virtually no difference in the results when using one fuel assembly's mixing coefficient or the average of the coefficients. This is true for the subchannels on the assembly border and the more limiting subchannels located towards the center of the hot assembly.

RAI-22f

In the original response to RAI-22f, Duke Energy provided the results of benchmarking the HTP expanded [ ]^{D} model to the [ ]^{D} model. These results confirmed that the [ ]^{D} model is conservative relative to the [ ]^{D} model. A similar process was repeated using the mixed core models. A select number of maximum allowable peaking limits developed using the [ ]^{D} mixed core VIPRE-01 model are compared to the equivalent [ ]^{D} mixed core VIPRE-01 model. The results are presented in Table 3.3 and show comparable and consistent performance of the expanded model for mixed core conditions.

Table 3.3 – Maximum Allowable Radial Peaking (MARP) Comparison Between [ ]D and [ ]D Mixed Core VIPRE-01 Model (HTP Surrounded by GAIA Fuel)

Accident Description	Axial Peak and Location		MARP Value	
	F_z	x/l	[ ]D	[ ]D
100% Power Loss of Flow Accident	1.05	0.01		
	1.05	0.9		
	1.5	0.5		
	1.9	0.01		
	1.9	0.9		
Single Uncontrolled Rod Withdrawal Accident	1.1	0.01		
	1.1	0.9		
	1.5	0.5		
	1.9	0.01		
	1.9	0.9		
100% Uncontrolled Bank Withdrawal Accident at BOC	1.3	0.1		
	1.3	0.8		
	1.7	0.1		
	1.7	0.5		
	1.7	0.8		
	2.1	0.1		
	2.1	0.8		
100% Uncontrolled Bank Withdrawal Accident at EOC	1.3	0.1		
	1.3	0.8		
	1.7	0.1		
	1.7	0.5		
	1.7	0.8		
	2.1	0.1		
	2.1	0.8		

The flow effects of the expanded mixed core model are shown in Figures 3.1 through 3.3 and confirm that the expected distribution of flow favoring the GAIA assemblies occurs. The [ ]D models used for this purpose set all the pin powers to 1.0 in order to isolate any observed differences to be due to hydraulic effects only. The 3-D plots illustrate the calculated channel exit flow in units of Mlbm/hr-ft^2 for the hot assembly in the middle and the three subchannels in the surrounding assemblies. The mixed core model shown in Figure 3.3 contains an HTP assembly in the middle with the GAIA fuel face adjacent. The channel exit flow profile in the full core HTP (Figure 3.1) and GAIA (Figure 3.2) models remains relatively flat across the hot assembly and neighboring assemblies since the fuel type is identical. The mixed core model results (Figure 3.3) show that the flow favors the surrounding GAIA assemblies and is depressed in the central HTP hot assembly. The mixed core model used to generate the results in Figure 3.3 assumes an ABETA term (see response to RAI-22e) in the assembly gap equal to the average ABETA for the two fuel types.

[ ]D

Figure 3.1 – [ ]D Full Core HTP Model Channel Exit Mass Flux

D

Figure 3.2 – [ ]D Full Core GAIA Model Channel Exit Mass Flux

[ ]D

Figure 3.3 – [ ]D Mixed Core Model Channel Exit Mass Flux

3.5 Conclusion

As discussed in the sections above, there is sufficient technical justification to ensure that the transition from HTP to GAIA fuel at HNP will not have any adverse effects on plant operation or the health and safety of the public.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.36, Technical Specifications, defines a safety limit as a limit upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.

General Design Criteria (GDC) 10, Reactor Design, requires that specified fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur.

The inclusion of the bounding Departure from Nucleate Boiling Ratio (DNBR) limit for GAIA fuel using the ORFEO-GAIA correlation continues to meet the requirements for safety limits as defined by 10 CFR 50.36 and maintains the GDC 10 requirements.

4.2 Precedent

The NRC previously approved a change to Callaway Plant, Unit 1, Technical Specifications via letter dated February 29, 2016 (ADAMS Accession No. ML16020A516), that allowed for providing the DNBR in a form that reduced the need for cycle-specific license amendments and the addition of an NRC-approved methodology for determining core operating limits.

The NRC also approved a change to Arkansas Nuclear One, Unit No. 2, Technical Specifications via letter dated September 18, 2009 (ADAMS Accession No. ML092460118), that updated the DNBR safety limit to reflect the transition to the Combustion Engineering 16x16 Next Generation Fuel design and the associated DNB correlations.

Additionally, per Reference 6.2, the NRC approved the use of Duke Energy methodology report DPC-NE-2005-P-A, Revision 5, by HNP, and the addition of Appendix I to the report, in which the use of the HTP correlation in VIPRE-01 was determined to be acceptable.

4.3 No Significant Hazards Consideration Determination

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment modifies TS 2.1.1.a to include the departure from nucleate boiling ratio safety limit associated with the transition to the GAIA fuel design from the

current HTP fuel. In addition, TS 6.9.1.6.2 will be revised to include the topical report for the NRC-approved correlation associated with the fuel design transition along with minor editorial adjustments to the formatting of the impacted pages. Duke Energy also requests the review and approval of Revision 6 of DPC-NE-2005-P, “Thermal-Hydraulic Statistical Core Design Methodology,” for the addition of Appendix J addressing the application of the ORFEO-GAIA critical heat flux correlation.

Duke Energy has evaluated whether or not 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed change incorporates into HNP TS a limit on the departure from nucleate boiling ratio (DNBR) safety limit for the GAIA fuel design that is based on a NRC reviewed and approved correlation, and does not require a physical change to plant systems, structures or components. Plant operations and analysis will continue to be in accordance with the HNP licensing basis. This change does not impact any of the accident initiators. The departure from nucleate boiling ratio is the basis for protecting the fuel and is consistent with the safety analysis. The DNBR setpoint continues to ensure automatic protective action is initiated to prevent exceeding the proposed DNBR safety limit.

The proposed safety limit ensures that fuel integrity will be maintained during normal operations and anticipated operational transients. The core operating limits report will be developed in accordance with the approved methodology. The proposed safety limit value and proposed Appendix J to the DPC-NE-2005-P methodology report do not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

The proposed change also adds a topical report for a NRC reviewed and approved critical heat flux (CHF) correlation to the list of topical reports in the HNP TS, which is administrative in nature and has no impact on a plant configuration or system performance relied upon to mitigate the consequences of an accident. The list of topical reports in the TS used to develop the core operating limits does not impact either the initiation of an accident or the mitigation of its consequences. Additionally, the editorial change to remove the excess spacing between sub-items in TS 6.9.1.6.2 is strictly administrative and has no impact on the technical content.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not require a physical change to plant systems, structures or components. The proposed change extends the use of DPC-NE-2005-P-A by HNP for GAIA fuel and adds a limit on DNBR for GAIA fuel to the HNP TS that is based on the NRC reviewed

and approved ORFEO-GAIA CHF correlation, ensuring that the fuel design limits are met. Operations and analyses will continue to be in compliance with NRC regulations. The addition of a new DNBR limit does not affect any accident initiators that would create a new accident.

The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The CHF correlation is not an accident/event initiator. No new initiating events or transients result from the use of the ORFEO-GAIA CHF correlation or the related safety limit change.

The proposed change also adds a topical report for a NRC reviewed and approved CHF correlation to the list of topical reports in the HNP TS, which is administrative in nature and has no impact on a plant configuration or system performance. The proposed change updates the list of NRC-approved topical reports used to develop the core operating limits. There is no change in the parameters within which the plant is normally operated. Additionally, the editorial change to remove the excess spacing between sub-items in TS 6.9.1.6.2 is strictly administrative and has no impact on the technical content. The possibility of a new of different kind of accident is not created.

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 change involve a significant reduction in a margin of safety?*

The proposed safety limit value has been established in accordance with the methodology for the ORFEO-GAIA CHF correlation to ensure that the applicable margin of safety is maintained (i.e., there is at least 95% probability at a 95% confidence level that the hot fuel rod does not experience DNB). The other reactor core safety limits will continue to be met by analyzing the reload using NRC-approved methods and incorporation of resultant operating limits into the Core Operating Limits Report (COLR). Consistent with the existing methodology, the use of the proposed Appendix J to the methodology will continue to ensure that all applicable design and safety limits are satisfied such that the fission product barriers will continue to perform their design functions.

Therefore, the proposed change does not involve a significant reduction in a 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, and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

Duke Energy has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Framatome Inc., ANP-10341P-A, Revision 0, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations,” September 2018 (ADAMS Accession No. ML18236A371 (Final Safety Evaluation Package)).
2. Barillas, M., U.S. Nuclear Regulatory Commission, letter to J.M. Frisco, Duke Energy Corporation, “Issuance of Amendments Revising Technical Specifications for Methodology Report DPC-NE-2005-P, Revision 5, “Thermal-Hydraulic Statistical Core Design Methodology” (CAC Nos. MF5872 and MF5873),” March 8, 2016 (ADAMS Accession No. ML16049A630).
3. Elnitsky, J., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, “Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3008, Revision 0,” November 10, 2016 (ADAMS Accession Number ML16315A286).
4. Galvin, D., U.S. Nuclear Regulatory Commission, letter to S. Capps, Duke Energy Corporation, “Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-3008-P, Revision 0, “Thermal-Hydraulic Models for Transient Analysis,” and DPC-NE-3009-P-A, Revision 0, “FSAR/UFSAR Chapter 15 Transient Analysis Methodology” (CAC Nos. MF8439 and MF8440; EPID L-2016-LLA-0012),” April 10, 2018 (ADAMS Accession Nos. Package - ML18060A404, Proprietary AMD - ML18060A318, Non-proprietary AMD - ML18060A401).
5. Letter from Duke Energy (John Elnitsky) to U.S. NRC dated May 4, 2016, “Supplemental Information for License Amendment Request Regarding Methodology Report DPC-NE-1008-P,” (ADAMS Accession No. ML16125A420).
6. Letter from M. Barillas (NRC) to K. Henderson (Duke Energy) dated May 18, 2017, “Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 - Issuance Of Amendments Revising Technical Specifications For Methodology Reports DPC-NE-1008-P Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 For Westinghouse Reactors," DPC-NF-2010 Revision 3, "Nuclear Physics Methodology For Reload Design," and DPC-NE-2011-P Revision 2, "Nuclear Design Methodology Report For Core Operating Limits Of Westinghouse Reactors" (CAC NOS. MF6648/MF6649 AND MF7693/MF7694),” (ADAMS Accession Nos. ML17102A911 (Proprietary AMD) and ML17102A923 (Non-proprietary AMD)).

RA-19-0017

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

7 PAGES PLUS COVER

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

a. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation for HTP fuel and ≥ 1.12 for the ORFEO-GAIA DNB correlation for GAIA fuel.

2.1.1 Th operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

~~a. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation.~~

b. The peak centerline temperature shall be maintained < [(2790 – 17.9 x P – 3.2 x B) x 1.8 + 32] ˚F where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

APPLICABILITY: MODES 1 and 2.

ACTION:

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig except during hydrostatic testing.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

No changes to this page

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

f. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), and V(Z) for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

i. Reactor Core Safety Limits Figure for Specification 2.1.1.

j. Overtemperature ΔT and Overpower ΔT setpoint parameters and time constant values for Specification 2.2.1.

k. Reactor Coolant System pressure, temperature, and flow Departure from Nucleate Boiling (DNB) limits for Specification 3/4.2.5.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

b. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

(Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

c. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

f. ANP-3011(P), “Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis,” Revision 1, as approved by NRC Safety Evaluation dated May 30, 2012.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

Reformatting page to remove excess unused space. Items from page 6-24b through 6-24d will be moved up to fill in the excess space. There will be no change to content of the method documentation.

No changes to the existing content provided on this page. Reformatting may result in relocation to a different page.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

k. BAW-10240(P)(A), “Incorporation of M5 Properties in Framatome ANP Approved Methods.”

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.2 – SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 – DNB Parameters, and 3.9.1 – Boron Concentration).

l. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

n. EMF-2310(P)(A), “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors”, approved version as specified in the COLR.

No changes to the existing content provided on this page. Reformatting may result in relocation to a different page.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

p. DPC-NE-2005-P-A, “Thermal-Hydraulic Statistical Core Design Methodology,” approved version as specified in the COLR.

(Methodology for Specification 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor)

q. DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” as approved by NRC Safety Evaluation dated May 18, 2017.

(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 – Boron Concentration).

r. DPC-NF-2010-A, “Nuclear Physics Methodology for Reload Design,” as approved by NRC Safety Evaluation dated May 18, 2017.

(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, and 3.9.1 – Boron Concentration).

s. DPC-NE-2011-P-A, “Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors” as approved by NRC Safety Evaluation dated May 18, 2017.

(Methodology for Specification 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

Reformatting may result in relocation to a different page.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

t. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated April 10, 2018.

(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

ADD: <INSERT>

u. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated April 10, 2018.

(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)

<INSERT>

v. ANP-10341P-A, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations,” approved version as specified in the COLR.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

RA-19-0017

ATTACHMENT 5

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(FOR INFORMATION ONLY)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

2 PAGES PLUS COVER

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB. The DNBR safety limit for high thermal performance fuel is 1.141 for the Siemens HTP correlation (Reference 1).

The restrictions of this safety limit also prevent fuel centerline melting. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The fuel centerline temperature limit is a function of weight percent of Gadolinia and pin burnup as presented in Reference 2 and approved for use at HNP per Reference 3.

ADD:
The DNBR safety limit for GAIA fuel is 1.12 for the ORFEO-GAIA correlation (Reference 4).

Undervoltage and Underfrequency - Reactor Coolant Pump Buses (Continued)

at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine inlet pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

References

1. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."
2. XN-NF-79-56(P)(A) Revision 1, "Gadolinia Fuel Properties for LWR Safety Evaluation."
3. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results."

ADD:
4. ANP-10341P-A, Revision 0, "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations", September 2018.

RA-19-0017

ATTACHMENT 7

APPLICATION OF THE ORFEO-GAIA CORRELATION TO THE GAIA FUEL DESIGN [NON-PROPRIETARY]

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

22 PAGES PLUS COVER

DPC-NE-2005

Duke Energy
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX J

Harris Plant Specific Data

GAIA Fuel

Application of the ORFEO-GAIA Correlation to the GAIA Fuel Design

February 2019

Note: Bracketed text, tables and figures are "D" (Duke) and/or "F" (Framatome Inc.) proprietary information.

This Appendix contains plant specific data and limits for the Harris Nuclear Plant with GAIA fuel using the ORFEO-GAIA critical heat flux correlation. The thermal hydraulic statistical core design analysis process was performed as described in the main body of this method report.

Plant Specific Data

The Harris Nuclear Plant is a three loop Westinghouse PWR. This analysis models the 0.374 inch fuel rod outer diameter GAIA fuel assembly design as described in Reference J-2.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference J-3 is used in this analysis. A [ ]^{D} model, based on the Harris Nuclear Plant 17x17 HTP fuel design in Appendix I of Reference J-1, was developed for the Harris Nuclear Plant GAIA fuel design. Due to the fuel assembly design differences, some specific data supplementary to Appendix I are updated. This data is listed in Table J-1 and the model adjustments are shown in Figure J-1. Table J-1 includes fuel rod, control rod, and instrument guide tube outer diameters, the number and design of the grids, and the fuel rod length.

The Harris Nuclear Plant 17x17 HTP fuel design [ ]^{D} VIPRE-01 model approved in Appendix I of Reference J-1 is used to analyze Harris Nuclear Plant GAIA fuel with the following modifications:

1) The GAIA fuel assembly geometry information and model layout as described in Table J-1 and Figure J-1.

2) The model orientation is rotated 90° around the assembly center instrument tube. This is done in order to put the guide/instrument tube type 1 channel (central channel 5, and also channels 1 and 10) in the center of the model since it has a higher flow loss coefficient and thus a lower minimum DNBR value compared to the guide/instrument tube type 0 channel (channels 2, 4, and 11).

3) The mixing coefficient between sub-channels is changed to the GAIA vendor supplied value of **[ ]** [F].

Critical Heat Flux Correlation

The NRC approved ORFEO-GAIA critical heat flux correlation form in Reference J-4 is used for the GAIA analyses in VIPRE-01. This correlation was developed by Framatome for application to the GAIA fuel design with the COBRA-FLX code, Reference J-4.

The ORFEO-GAIA correlation was added to the VIPRE-01 thermal-hydraulic computer code along with the ORFEO-NMGRID correlation (for use in the non-mixing grid region of the fuel) by Duke Energy, and the CHF test data base was analyzed in its entirety. The results of this analysis are shown in Table J-2. The bounding values for standard deviation and correlation limit are derived by examining the statistics of the data arranged by different sub-regions of pressure, mass flux, and thermodynamic quality. This process helps to account for the presence of any non-conservative sub-regions that exist in the correlation applicability range. The more conservative bounding results are also included in Table J-2. In the NRC's Safety Evaluation Report for Framatome's implementation of the ORFEO-GAIA correlation (Reference J-4), a bias of 0.01 is applied to Framatome's calculated correlation limit to account for differences in the tested fuel assembly and the production fuel assembly. This bias is included in the final correlation limit values in Table J-2. The resulting VIPRE-01 average M/P is [ ],[D] the bounding standard deviation [ ],[D] and the correlation DNBR limit is lower than the value of COBRA-FLX (Reference J-4, shown on Table J-2 under COBRA-FLX column). Figures J-3 through J-6

graphically shows the results of this evaluation. Figure J-3 shows there is [

]^{D} Figures J-4 through J-6 show there is [

]^{D}

Based on the results shown in Table J-2 and Figures J-3 through J-6, the ORFEO-GAIA CHF correlation can be used in DNBR calculations with VIPRE-01 for GAIA fuel. Table J-3 shows the correlation allowable parameter range and design limit with VIPRE-01. Note that the higher correlation limit will be used for VIPRE-01 analyses of the GAIA fuel design.

Statistical Core Design Analysis

Statepoints

The state point conditions evaluated in this analysis are listed in Table J-4. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied. The range of key parameter values analyzed is listed on Table J-7.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table J-5. The uncertainties were selected to bound the values calculated for each parameter at the Harris Nuclear Plant with the GAIA fuel design.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed on Table J-6. Section 1 of Table J-6 contains the 500 case runs and Section 2 contains the 10,000 case runs. All of the DNBR distributions are judged to be normally distributed. The maximum statistical DNBR value in Table J-6 for 10,000 case runs is [ ] D. Therefore, the statistical design limit, using the ORFEO-GAIA CHF correlation in VIPRE-01 for GAIA fuel at Harris, is conservatively selected to be 1.28.

References

J-1 DPC-NE-2005-PA, Revision 5, Thermal-Hydraulic Statistical Core Design Methodology, March 2016.

J-2 ANP-10342P, Revision 0, GAIA Fuel Assembly Mechanical Design Topical Report, December 2016.

J-3 EPRI NP-2511-CCM-A, Revision 4.6, VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, Vol. 1-4, Battelle Pacific Northwest Laboratories, February 2017.

J-4 ANP-10341P-A, Revision 0, The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations, September 2018.

FIGURE J-1

GAIA FUEL DESIGN

VIPRE-01 [ ]^{D} MODEL

D

xx Fuel Rod

xx Instrument Guide Tube

xx Control rod Guide Tube

xx = Rod Index

yy Flow Channel

yy = Channel Index

FIGURE J-2

GAIA FUEL DESIGN

VIPRE-01 [ ]D MODEL RADIAL POWER DISTRIBUTION

D

Fuel Rod

Instrument Guide Tube

Control Rod Guide Tube

FIGURE J-3

VIPRE-01 MEASURED CHF VERSUS PREDICTED CHF

ORFEO-GAIA DATABASE

VIPRE-01 Measured vs Predicted DNB Heat Flux

D

FIGURE J-4

VIPRE-01 MEASURED TO PREDICTED CHF VERSUS MASS FLUX

ORFEO-GAIA DATABASE

VIPRE-01 DNB Heat Flux as a Function of Local Mass Flux

D

FIGURE J-5

VIPRE-01 MEASURED TO PREDICTED CHF VERSUS PRESSURE

ORFEO-GAIA DATABASE

VIPRE-01 DNB Heat Flux as a Function of Pressure

D

FIGURE J-6

VIPRE-01 MEASURED TO PREDICTED CHF VERSUS QUALITY

ORFEO-GAIA DATABASE

VIPRE-01 DNB Heat Flux as a Function of Quality at CHF

D

TABLE J-1
GAIA FUEL ASSEMBLY DATA
(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod outer diameter, inches (nominal):	0.374
Thimble tube diameter, inches (nominal):	[ ]^{F}
Instrument guide tube diameter, inches (nominal):	[ ]^{F}
Fuel rod pitch, in (nominal):	0.496
Fuel assembly pitch, inches (nominal):	8.466
Active Fuel Length, inches (nominal):	144.0
Fuel rod length, inches (nominal):	151.89

GENERAL FUEL CHARACTERISTICS

Component	Material	Quantity	Position	Type
Grid	Nickel Alloy-718	1	Lower	HMP, Non-Mixing
	M5	6	Intermediate	GAIA
	M5	3	Intermediate	IGM
	Nickel Alloy 718	1	Upper	HMP, Non-Mixing
Fuel Rod	M5	264		
CRGT	Q12	24		
IGT	Q12	1		

CRGT = Control Rod Guide Tube
IGT = Instrument Guide Tube

TABLE J-2

VIPRE-01 ORFEO-GAIA CORRELATION VERIFICATION

VIPRE-01 / COBRA-FLX STATISTICAL RESULTS

	VIPRE-01	COBRA-FLX
n, # of Data	[ ] D	[ ] F
M/P, Average Measured to Predicted CHF	[ ] D	[ ] F
σ (M/P), Calculated	[ ] D	[ ] F
DNBR Correlation Limit, Calculated	[ ]D	[ ]F
σ (M/P), Bounding	[ ] D	[ ] F
DNBR Correlation Limit, Calculated	1.10	1.11
DNBR Correlation Limit, including 0.01 bias	1.11	1.12

TABLE J-3

CHF TEST DATABASE ANALYSIS RESULTS

PARAMETER RANGES

Pressure, psia	1446 to 2530.9
Mass Flux, Mlbm/hr-ft^2	0.5012 to 3.1878
Thermodynamic Quality at CHF	less than 0.7992

Thermal-Hydraulic Computer Code	*VIPRE-01*
Spacer Grid	*GAIA*
DNBR Correlation Limit	*1.12*

TABLE J-4
HARRIS NUCLEAR PLANT SCD STATEPOINTS

Statepoint #	Core Exit Pressure (psia)	Core Inlet Temperature (°F)	Core Inlet Flow (1) (%)	Core Power (2) (% FP)	Axial Peak Magnitude Location (Fz)	Axial Peak Magnitude Location Z	Radial Peak (GAIA) (FΔH)
1							
2							
3							
4							
5							
6							
7							
8							
9							
10							
11							
12							
13							
14							
15							
16							
17							
18							
19							
20							
21							
22							
23							
24							
25							
26							
27							
28							
29							
30							
31							
32							
33							
34							
35							
36							
37							
38							

D

TABLE J-4
HARRIS NUCLEAR PLANT SCD STATEPOINTS
(CONTINUED)

Statepoint #	Core Exit Pressure (psia)	Core Inlet Temperature (°F)	Core Inlet Flow (1) (%)	Core Power (2) (% FP)	Axial Peak Magnitude Location (Fz)	Axial Peak Magnitude Location Z	Radial Peak (GAIA) (FΔH)
39							
40							
41							
42							
43							
44							
45							
46							
47							
48							
49							
50							

D

1) 100% RCS Flow = 293,540 gpm
2) 100% RTP = 2,948 MWt

TABLE J-5

HARRIS STATISTICALLY TREATED UNCERTAINTIES

PARAMETER	UNCERTAINTY / STANDARD DEVIATION	DISTRIBUTION
Core Power:*	± 0.34% / 0.21%	Normal
Coolant Flow		
Measurement:	± 2.2% / 1.34%	Normal
Bypass Flow:	± 1.5%	Uniform
Core Exit Pressure:	± 50.0 psia	Uniform
Core Inlet Temperature:	± 5.0 degrees F	Uniform
Radial Power Distribution		
$F^N_{\Delta H}$ (measurement):	± 4.0% / 2.43%	Normal
$F^E_{\Delta H}$ (engineering):	± 5.4% / 3.28%	Normal
Axial Power Distribution		
F_Z:	± 4.5% / 2.74%	Normal
Z:	± 3 inches	Uniform
DNBR		
Correlation:	[ ] F,D	Normal
Code/Model:	[ ] D	Normal

* Percentage of 100% RTP (2948 MWth) wherever applied.

TABLE J-5 (continued)

HARRIS STATISTICALLY TREATED UNCERTAINTIES

PARAMETER	JUSTIFICATION
Core Power	The core power uncertainty is calculated by combining various component uncertainties associated with the measurement of core power. Since the component uncertainties are random and are normally distributed, the combination of these uncertainties using the sum of the squares (SRSS) methodology results in a core power uncertainty that is also normally distributed.
Coolant Flow	
Measurement:	Same approach as Core Power uncertainty.
Bypass Flow:	The core bypass flow is the parallel core flow paths in the reactor vessel non-fuel regions and is dependent on the driving pressure drop. The bypass flow uncertainty is explicitly applied in the calculation of core inlet flow rate for each state point condition. This uncertainty was conservatively applied with a uniform distribution
Core Pressure	The reactor coolant pressure uncertainty is calculated by statistically combining various component uncertainties associated with the measurement of pressure. This uncertainty is conservatively applied as a uniform distribution.
Core Temperature	Same approach as Pressure uncertainty.
Radial Power Distribution	
$F^N_{\Delta H}$ (measurement):	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly and pin power, and the measurement of the assembly power. This uncertainty is applied as a normal distribution.
$F^E_{\Delta H}$ (engineering):	This uncertainty accounts for the effect on peaking due to manufacturing variations in the variables affecting the heat generation rate along the flow channel and for the effect on peaking due to reduced hot channel flow area. The uncertainty is determined by statistically combining all the manufacturing tolerances. The uncertainty is normally distributed and is conservatively applied as one-sided to assure the MDNBR channel location is consistent for all cases.

TABLE J-5 (continued)

HARRIS STATISTICALLY TREATED UNCERTAINTIES

PARAMETER	JUSTIFICATION
Axial Power Distribution	
F_Z:	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. This uncertainty is applied as a normal distribution.
Z:	This uncertainty accounts for the possible error in interpolating on axial peak location in the Maneuvering Analysis. The uncertainty is one half of the physics code's axial node length. The uncertainty distribution is conservatively applied as uniform.
DNBR	
Correlation:	This uncertainty accounts for the CHF correlation's ability to predict DNB. This uncertainty is applied as a normal distribution.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between various model sizes. This uncertainty is applied as a normal distribution.

TABLE J-6

HARRIS STATEPOINT STATISTICAL RESULTS

SECTION 1

GAIA FUEL
ORFEO-GAIA CRITICAL HEAT FLUX CORRELATION
(500 CASE RUNS)

State Point #	Mean	Standard Deviation	Coefficient of Variation	Statistical DNBR Limit
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				
26				
27				
28				
29				
30				
31				
32				
33				
34				

D

TABLE J-6 (continued)

HARRIS STATEPOINT STATISTICAL RESULTS

SECTION 1
(CONTINUED)

State Point #	Mean	Standard Deviation	Coefficient of Variation	Statistical DNBR Limit
35				
36				
37				
38				
39				
40				
41				
42				
43				
44				
45				
46				
47				
48				
49				
50				

D

TABLE J-6 (continued)

HARRIS STATEPOINT STATISTICAL RESULTS

SECTION 2

GAIA FUEL
ORFEO-GAIA CRITICAL HEAT FLUX CORRELATION
(10,000 CASE RUNS)

State Point #	Mean	Standard Deviation	Coefficient of Variation	Statistical DNBR Limit
1				
15				
31				
43				
49				

D

TABLE J-7

HARRIS KEY PARAMETER RANGES

Key Parameter	Maximum	Minimum
Core Power (% RTP)		
Coolant Flow (GPM)		
Core Exit Pressure (psia)		
Core Inlet Temperature (°F)		
$F_{\Delta H}$, F_z, Z		

D

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.