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

984-229-2512

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390
UPON REMOVAL OF ATTACHMENTS 3 AND 6, THIS LETTER IS DECONTROLLED

10 CFR 50.90

November 24, 2020
Serial: RA-20-0335

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: Response to Request for Additional Information Regarding 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 application dated March 6, 2020, as supplemented by letters dated April 23 and June 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20066L112, ML20114E131, and ML20174A640, respectively), Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for 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 flow rate, whereas TS 6.9.1.6.2 would reflect the incorporation of the Framatome, Inc. topical report 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LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors," and the removal of analytical methods no longer applicable for the determination of HNP core operating limits.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the LAR and determined that additional information is needed to complete their review. Duke Energy received the request for additional information (RAI) from the NRC through electronic mail on October 13, 2020 (ADAMS Accession No. ML20297A307). The attachments to this letter provide Duke Energy's response to the RAI. Attachments 2 and 5 provide the affidavits from Framatome, Inc. and Duke Energy supporting the request for withholding of proprietary information in Attachments 3 and 6, respectively, from public disclosure.

This additional information does not change the No Significant Hazards Determination provided in the original submittal. No regulatory commitments are contained within 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 November 24, 2020.

Sincerely,

Kim E. Maza

Kim E. Maza
Site Vice President
Harris Nuclear Plant

Attachments:

1) Response to Request for Additional Information
2) Affidavit for Withholding of Proprietary Information (Framatome, Inc.)
3) ANP-3766Q1P, Revision 0, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design – NRC RAI Responses" (Proprietary)
4) ANP-3766Q1NP, Revision 0, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design – NRC RAI Responses" (Nonproprietary)
5) Affidavit for Withholding of Proprietary Information (Duke Energy)
6) Duke Energy Response to RAI 2 - Proprietary
7) Duke Energy Response to RAI 2 - Nonproprietary

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

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

Response to Request for Additional Information

By application dated March 6, 2020, as supplemented by letters dated April 23 and June 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20066L112, ML20114E131, and ML20174A640, respectively), Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for 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. topical report 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.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the LAR and determined that additional information is needed to complete their review. Duke Energy received the request for additional information (RAI) from the NRC through electronic mail dated October 13, 2020 (ADAMS Accession No. ML20297A307).

<u>RAI – 1</u>

Framatome Licensing Report ANP-3766P/NP, “Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design,” enclosed the March 6, 2020, letter, states, in Section 4.3:

> For Harris Nuclear Plant Unit 1, the condition for which all three RCPs [Reactor Coolant Pumps] are tripped is based on the RCS pressure with consideration of required operator action times specified in the plant Emergency Operating Procedure. A delayed RCP trip time of 5 minutes following the specified trip criteria being met was analyzed to demonstrate compliance to 10 CFR 50.46(b)(1-4) criteria…
>
> The spectrum of cold and hot leg breaks in this study includes break sizes from 1.00 to 8.70 inches. The results of the delayed RCP trip cases indicate that there is at least 5 minutes for operators to trip all three RCPs after the specified trip criteria being met with considerable margin to the 10 CFR 50.46(b)(1-4) criteria.

According to Title 10 of the Code of Federal Regulations (CFR), Section 50.46(a)(1)(i), emergency core cooling performance must be analyzed for a number of postulated loss-of-coolant accidents of break sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accident has been calculated.

Considering the limiting range of breaks for Harris and associated phenomenology, a 5-minute RCP trip delay is essentially the same as allowing the RCPs to run throughout the event. Meanwhile, it is not clear, based on the sequence of events for the limiting break size range, whether an operator would trip the RCPs sooner than the 5-minute trip delay time that was analyzed.

Please provide additional justification that a shorter RCP trip delay time would not lead to a higher Peak Cladding Temperature (PCT). Consider specifically, RCP trip delay times that may result in RCP trip prior to the time of PCT.

Duke Energy Response

HNP Operations personnel perform simulator validation for the Small Break LOCA (SBLOCA) manual RCP criteria as part of the current Time Critical Action (TCA) program. Performance of this action demonstrated that the operators would trip the RCPs within 2 minutes and 1 second relative to the start of the simulated SBLOCA event. This timing is the average of 3 operating crews. The individual timing for each of the crews is provided below and confirms that the average timing is representative of HNP Operations crew performance.

Crew 1: SBLOCA RCP Trip TCA time of 2 minutes and 6 seconds
Crew 2: SBLOCA RCP Trip TCA time of 2 minutes and 14 seconds
Crew 3: SBLOCA RCP Trip TCA time of 1 minutes and 43 seconds

The use of a singular best-estimate, or average, RCP trip time value is consistent with other licensee responses to similar RAIs.

- Palo Verde Nuclear Generating Station Units 1, 2, and 3 – Letter dated May 17, 2019, “Response to NRC Staff Request for Additional Information from Reactor Assessment and Human Performance Branch Regarding License Amendment and Exemption Requests Related to the Implementation of Framatome High Thermal Performance Fuel [ADAMS Accession No. ML19137A118]
- North Anna & Surry Power Stations Units 1 and 2 – Letter dated May 28, 2020, “Proposed License Amendment Requests Addition of Analytical Methodology to the Core Operating Limits Report for a Small Break Loss of Coolant Accident (SBLOCA) Response to Request for Additional Information and Analysis Error Correction” [ADAMS Accession No. ML20149K694]

The simulated SBLOCA break size used in the HNP RCP trip validation was a 2.6-inch break on the cold leg RCS piping. For this break size, the RCP trip criteria are met in approximately 1 minute after event initiation. The operators must perform the Immediate Action steps in HNP Emergency Operating Procedure (EOP) EOP-E-0, “Reactor Trip or Safety Injection,” before the RCP trip criteria provided in the procedure are applicable. The larger break sizes evaluated in ANP-3766P which have limiting PCTs would reach the RCP trip criteria sooner, but these breaks will have no significant impact on the timing to stop all of the RCPs. The steps of EOP-E-0 will remain the same for larger break sizes, and the 3-way communications required by the Operations personnel will remain the same. It is the procedure implementation and the communications that will continue to establish the manual RCP trip time.

Framatome performed SBLOCA break spectrum sensitivity studies using the manual RCP trip time of 121 seconds following event initiation. Those sensitivity studies are documented in Framatome Report ANP-3766Q1P (see Attachment 3; nonproprietary version provided in Attachment 4). The limiting PCT from the break spectrum analysis and the 5-minute delayed RCP trip cases documented in ANP-3766P bounds the limiting PCT from the best estimate RCP trip timing cases shown in ANP-3766Q1P.

The HNP GAIA fuel SBLOCA break spectrum analysis, 5-minute RCP trip study, and the sensitivity studies presented in this RAI response (Attachments 3 and 4) confirm that the 10 CFR 50.46 acceptance criteria continue to be met under the HNP SBLOCA application of the methodology specified in EMF-2328(P)(A), “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” as supplemented in EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0. RCP trip behavior based on the best-estimate operator action time is shown to not produce more severe consequences. The licensing basis analyses described in ANP-3766P supporting GAIA fuel implementation for HNP Cycle 24 are those that assume RCP trip coincident with reactor trip.

RAI – 2

In response to NRC Request No. 3, in the April 23, 2020, supplement letter, the licensee provided information addressing the limitations and conditions associated with the implementation of the Framatome GAIA fuel assembly topical report, ANP-10342NP-A, “GAIA Fuel Assembly Mechanical Design,” (ADAMS Accession No. ML19309D916). Among other things, the licensee noted that the rod cluster control assembly (RCCA) ejection accident would be subject to analysis or evaluation to support loading of the GAIA fuel assembly design, in accordance with the reload safety analysis methods in use at Harris. This information was provided to address Limitation and Condition 5 of the ANP-10342NP-A safety evaluation.

However, Limitation and Condition 5 requires demonstration that the acceptance criteria are satisfied. The information referenced in the previous RAI response was based on a different fuel system design, meaning it is unclear how the analysis would apply to GAIA fuel.

Please provide information to demonstrate that implementing the GAIA fuel design at Harris will satisfy the applicable acceptance criteria, thus ensuring applicability of the submitted loss-of-coolant accident analyses.

Duke Energy Response

See Attachments 6 and 7 for the respective proprietary and nonproprietary responses to RAI 2.

ATTACHMENT 2

AFFIDAVIT FOR WITHHOLDING OF PROPRIETARY INFORMATION (FRAMATOME, INC.)

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

3 PAGES PLUS THE COVER

A F F I D A V I T

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs 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 Licensing Report ANP-3766Q1P, Revision 0, entitled, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design - NRC RAI responses," dated November 2020 and 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 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(d) and 6(e) 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.

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

Executed on: November 6, 2020

____

Gayle Elliott

ATTACHMENT 4

ANP-3766Q1NP, REVISION 0, "HARRIS NUCLEAR PLANT UNIT 1 SMALL BREAK LOCA ANALYSIS WITH GAIA FUEL DESIGN – NRC RAI RESPONSES" (NONPROPRIETARY)

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

12 PAGES PLUS THE COVER

framatome

Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design - NRC RAI responses

ANP-3766Q1NP
Revision 0

Licensing Report

November 2020

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

Nomenclature

Acronym	Definition
CFR	Code of Federal Regulations
CWO	Core-Wide Oxidation
EM	Evaluation Model
LOCA	Loss of Coolant Accident
MLO	Maximum Local Oxidation
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SBLOCA	Small Break Loss-of-Coolant Accident

1.0 SUMMARY

This report contains the Framatome response to RAI 1 submitted by the NRC for the HNP Unit 1 SBLOCA Analysis supporting the GAIA fuel design (Reference 1).

2.0 RAI 1

Request:

Framatome Licensing Report ANP-3766P/NP, "Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design," enclosed the March 6, 2020, letter, states, in Section 4.3:

> *For Harris Nuclear Plant Unit 1, the condition for which all three RCPs [Reactor Coolant Pumps] are tripped is based on the RCS pressure with consideration of required operator action times specified in the plant Emergency Operating Procedure. A delayed RCP trip time of 5 minutes following the specified trip criteria being met was analyzed to demonstrate compliance to 10 CFR 50.46(b)(1-4) criteria.*
>
> *The spectrum of cold and hot leg breaks in this study includes break sizes from 1.00 to 8.70 inches. The results of the delayed RCP trip cases indicate that there is at least 5 minutes for operators to trip all three RCPs after the specified trip criteria being met with considerable margin to the 10 CFR 50.46(b)(1-4) criteria.*

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Considering the limiting range of breaks for Harris and associated phenomenology, a 5-minute RCP trip delay is essentially the same as allowing the RCPs to run throughout the event. Meanwhile, it is not clear, based on the sequence of events for the limiting break size range, whether an operator would trip the RCPs sooner than the 5-minute trip delay time that was analyzed.

Please provide additional justification that a shorter RCP trip delay time would not lead to a higher Peak Cladding Temperature (PCT). Consider specifically, RCP trip delay times that may result in RCP trip prior to the time of PCT.

Response:

RAI 1 Supporting Material:

RCP trip time delay sensitivity studies were performed for Harris. Both cold and hot leg SBLOCA studies were conducted for breaks between 1.0 and 8.7 inches [ ] The RCP trip delay studies of Reference 1 utilized a “delayed RCP trip time of 5 minutes following the specified trip criteria being met”. In response to RAI 1, additional studies were performed, tripping the RCPs 2 minutes and 1 second following break initiation. The shorter delay was based on Harris simulator validation of Time Critical Operator Action regarding post-SBLOCA reactor coolant pump trip when loss of offsite power is not considered. The new studies used a best-estimate operator action time, but retained all of the conservatisms prescribed in the EM for the break spectrum analysis (i.e. Appendix K requirements were adapted). The limiting break from the break spectrum results (PCT of 1832°F for a break 7.5 inches in diameter, Reference 1) was also included.

The results for both cold and hot leg breaks supporting Reference 1 (5 minute RCP trip delay) are presented in Table 2-1 and Table 2-2. Results of the updated studies (2 minute, 1 second RCP trip delay) are presented in Table 2-3 and Table 2-4. Like the 5 minute delay studies reported in Reference 1, the studies performed for the RAI 1 response demonstrate that with a 2 minute and 1 second manual RCP trip delay, the 10 CFR 50.46(b)(1-4) criteria continue to be met.

Table 2-1
Harris Cold Leg Break Transient Results - 5 Minute RCP Trip Delay

[ ]

Table 2-2
Harris Hot Leg Break Transient Results - 5 Minute RCP Trip Delay

[ ]

Table 2-3
Harris Cold Leg Break Transient Results - 2 Minute, 1 Second RCP Trip Delay

[ ]

Table 2-4
Harris Hot Leg Break Transient Results - 2 Minute, 1 Second RCP Trip Delay

[ ]

3.0 REFERENCES

[1] ANP-3766P, Revision 0, Harris Nuclear Plant Unit 1 Small Break LOCA Analysis with GAIA Fuel Design.

ATTACHMENT 5

AFFIDAVIT FOR WITHHOLDING OF PROPRIETARY INFORMATION
(DUKE ENERGY)

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

3 PAGES PLUS THE 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 Attachment 6 to Duke Energy letter RA-20-0335 regarding the response to the request for additional information pertaining to the application 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.

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 consists 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 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, 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 Attachment 6 to Duke Energy letter RA-20-0335 regarding the response to the request for additional information pertaining to the application 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. 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 November 17, 2020

Steve Snider

ATTACHMENT 7

DUKE ENERGY RESPONSE TO RAI 2 - NONPROPRIETARY

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

6 PAGES PLUS THE COVER

Note: Text that is within brackets with an “a,c” superscript is proprietary to Duke Energy and has been removed.

1.0 Background

The transition from the HTP fuel design to the GAIA fuel design is being pursued to take advantage of an increase in thermal performance, along with features to protect against fuel failures from debris and grid-to-rod fretting. Corrosion resistance is maintained at a level consistent with the HTP fuel design through the continued use of the M5 fuel cladding material. The most significant change associated with the GAIA fuel assembly design relative to the HTP fuel design pertains to the introduction of a new spacer grid design which increases departure from nucleate boiling (DNB) performance. Fuel rod changes include a small increase in the pellet diameter, compensated for by an increase in clad inner diameter to maintain the same pellet-to-clad gap as the HTP design. These changes also preserve fission gas volume. The fuel rod outer diameter is also decreased by 2 mils. Small diametrical changes are also made to the guide tubes and instrument tubes to increase the structural integrity of the fuel assembly. Last, the fuel pellet density is increased, which increases the uranium loading and effective fissile content of the fuel while reducing burnup for a given time of operation. The cumulative effect of these changes is an increase in thermal peaking margins, which provides additional fuel management flexibility. Important fuel rod design characteristics for the HTP and GAIA fuel assembly designs are summarized in Table 1. In addition, GAIA instrument and guide tube outer diameters are slightly increased along with the implementation of new spacer grid design.

Table 1
Advanced HTP and GAIA Fuel Rod Design Characteristics

Parameter	Advanced HTP	GAIA
Fuel Rod Pitch	0.496 in.	0.496 in.
Active Fuel Height	144 in.	144 in.
Fuel Pellet Outer Diameter	0.3215 in.	0.3225 in.
Clad Outer Diameter	0.376 in.	0.374 in.
Clad Inner Diameter	0.328 in.	0.329 in.
Clad Thickness	0.024 in.	0.0225 in.
Pellet-Clad Gap Width	0.00325 in.	0.00325 in.

The small fuel rod changes between the GAIA and HTP fuel rod designs will not have a significant impact on the neutronic response of the reactor core during a control rod ejection accident. The important parameters that drive this accident are the ejected rod worth, beta-effective, and Doppler temperature coefficient, and to a much lesser extent, the moderator temperature coefficient. The small changes in fuel rod and assembly design do not significantly impact the magnitude of these reactivity parameters. Additionally, the control rod ejection accident is evaluated using bounding values for ejected rod worth, Doppler and moderator temperature coefficient, and beta-effective. The values assumed in the safety analysis are confirmed on a cycle-specific basis to ensure the safety analysis remains bounding. The nodal simulator used in the calculation of key physics parameters and power distributions models each unique fuel type in the reactor core. Consequently, cycle-specific calculations intrinsically account for the fuel management and fuel assembly design differences, in the confirmation of safety analysis assumptions and in the power distributions used to verify the acceptability of energy deposition and DNB ratio (DNBR) thermal limits.

2.0 Acceptance Criteria

The acceptance criteria for the control rod ejection accident are specified in the response to RAI 43 for the DPC-NE-3009-P-A submittal (Reference 1) in letter dated October 30, 2017 (ADAMS Accession Nos. ML17303B209, ML17303B205). These limits ensure core coolability and offsite doses are within regulatory limits. Limits on peak fuel average enthalpy, prompt enthalpy rise and fuel melt were specified and are summarized below. DNB failure limits (i.e. number of fuel rods in DNB) are site specific and are based on offsite dose consequences satisfying regulatory limits. A peak primary pressure limit is established to maintain reactor coolant system integrity.

2.1 Core Coolability Limits

Core coolability limits may not be exceeded.

- Peak radial average fuel enthalpy must be less than 230 cal/gm.
- Peak fuel temperature must be less than the fuel melt temperature.

2.2 Fuel Cladding Failure Limits

The number of fuel rods exceeding each failure limit must be less than the number of fuel pins assumed in the dose analysis. Fuel rods which fail more than one criterion are not double counted. Failure mechanisms evaluated include high temperature cladding (total enthalpy), DNBR and pellet clad mechanical interaction (PCMI) type failures. Radial average fuel enthalpies are calculated as described in Section 5.4.8.2 of Reference 1. Calculation of radial average enthalpy limits and cycle-specific confirmation of these limits are described in the response to RAI 48 of Reference 1. The methodology for determining DNBR limits and the method used to calculate the number of fuel rods in DNB are described in Section 5.4.8.3 of Reference 1, while the PCMI methodology is described in the response to RAI 43 of this same reference.

High Temperature Cladding Failure Limits:
High temperature cladding failure limits are applicable for control rod ejections initiated from the HZP initial condition. The limits are dependent upon fuel rod internal pressure and are specified in terms of total fuel enthalpy. For fuel rods with an internal pressure at or less than system pressure, the limit is 170 cal/gm. A limit of 150 cal/gm is applicable for fuel rods with an internal pressure greater than system pressure.

DNBR Limits:
Fuel cladding failure from DNB is assumed to occur if the surface heat flux exceeds the DNBR limit. DNBR limits are calculated for each fuel type resident in the reactor core using NRC-approved critical heat flux (CHF) correlations. Fuel type dependent limits or a composite limit which bounds all fuel types may be used. Assessment of this failure criterion is applicable for MODE 1 reactor operation (> 5% rated thermal power).

PCMI limits:
PCMI limits are from Figure B-1 of NUREG-0800, “Standard Review Plan,” Section 4.2, “Fuel System Design,” Revision 3 (Reference 2) and are specified in terms of prompt fuel enthalpy rise as a function of oxide-to-wall thickness. These limits are applicable for all MODES of operation.

2.3 Peak Primary Pressure

The peak primary pressure from the power excursion must not exceed Service Limit C as defined in the American Society of Mechanical Engineers (ASME) code (120% of the design pressure).

3.0 Impact of GAIA Fuel Transition

The transition from the HTP to GAIA fuel assembly design has the potential to impact the following control rod ejection accident calculations.

- SIMULATE-3K transient power response
- Fuel enthalpy (total and enthalpy rise) and temperature
- DNB
- PCMI
- Peak Primary Pressure
- Source Term

The impact of the transition from the HTP to the GAIA fuel design for each of these calculations is discussed next.

3.1 SIMULATE-3K Transient Power Response

SIMULATE-3K is used to calculate the transient core power response and transient core power distributions resulting from a rapid positive reactivity insertion from a control rod ejection. The resulting temporal core power excursion and power distributions are used in VIPRE-01 calculations to calculate fuel temperature and enthalpy, and the allowable power peaking to avoid exceeding the DNBR limit. As previously discussed, the transition from the HTP to the GAIA fuel design has negligible impact on the neutronic characteristics of the fuel and, therefore, the transient power response of the event and power distribution. The reference analysis assumes bounding values of important physics parameters which are expected to bound future reload cores and produce a conservative power response. Cycle-specific confirmation of these parameters assures the acceptability of the reference analysis for future core designs containing a combination of HTP and GAIA fuel or a full complement of GAIA fuel.

3.2 Fuel Temperature and Enthalpy

Rod ejection accident fuel enthalpy and temperature calculations are performed using the transient VIPRE-01 fuel rod conduction model with input from SIMULATE-3K. Section 5.4.8.2 of Reference 1 describes additional details about the methodology. **[**

][a,c]

For fuel rods that may approach DNB, the increase in the thermal performance produced from improvements in the GAIA mixing vane grid design **[**

][a,c] Collectively, the changes in the GAIA fuel rod and assembly design increase the peaking margin to fuel temperature and enthalpy limits for the rod ejection transient.

3.3 DNB

VIPRE-01 is used to calculate pin radial peaking limits for both the advanced HTP and GAIA fuel designs. These limits are subsequently used to determine the number of fuel pins in DNB as described in Section 5.4.8.3 of Reference 1. **[**

][a,c]

[

][a,c]

Because the pressure drop for the HTP fuel design is greater than that for the GAIA fuel design, flow is diverted in a mixed core from the HTP to GAIA fuel. This results in a DNBR penalty for the HTP fuel. Accordingly, the radial peaking DNBR limits for HTP fuel are penalized for mixed core conditions based on the method approved in Reference 3. No credit is taken for the flow increase in the calculation of GAIA radial pin DNBR peaking limits. **[**

][a,c]

3.4 PCMI

The PCMI acceptance criterion is based on the amount of oxide present on the fuel and is indexed to the oxide-to-wall thickness ratio (Figure B-1 of Reference 2). **[**

][a,c]

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][a,c]

In summary, **[**

][a,c]

3.5 Peak Primary Pressure

The peak primary pressure resulting from the rod ejection power excursion must be less than or equal to 120% of the ASME Service Limit C design pressure. Analysis of the peak primary pressure for this event is not performed because it is bounded by the Uncontrolled Bank Withdrawal from Subcritical or Low Power accident. The cycle-specific confirmation of the key assumptions made in the rod ejection accident analysis ensures the transient power response assumed in the reference control rod ejection analysis remains bounding, thus confirming the peak primary pressure acceptance criterion.

3.6 Source Term

The change in the GAIA fuel rod design results in a small increase in uranium loading and source term. This change is reflected in the calculation of dose consequences and the failed fuel fraction limits established to satisfy regulatory dose requirements. Cycle-specific calculations are performed to demonstrate that these limits are satisfied for each reload core.

4.0 Summary

Fuel rod and assembly design changes associated with the transition from the HTP to GAIA fuel assembly design do not have a significant impact on the neutronic response of the reactor core during a control rod ejection accident. **[**

][a,c]

The acceptability of the GAIA fuel design is confirmed for each reload core through the performance of cycle-specific calculations which model each fuel type present in the reactor core. These calculations are performed in accordance with the methodology described in Reference 1 to ensure the reference SIMULATE-3K control rod ejection analysis remains bounding and all acceptance criteria are satisfied. In the event that a design limit is challenged, margin exists to the bounding key physics parameters assumed in the analysis of the Harris rod ejection accident. If necessary, the accident could be reanalyzed with less limiting parameters or, alternately, the reload core could be redesigned to recapture margin to the design limit being challenged.

5.0 References

1. DPC-NE-3009-P-A, Rev. 0, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology”, as approved by the NRC per letter dated April 10, 2018 (ADAMS Accession No. ML18060A401)

2. U.S. Nuclear Regulatory Commission, NUREG-0800, “Standard Review Plan,” Section 4.2, “Fuel System Design,” Revision 3, March 2007 (ADAMS Accession No. ML070740002)

3. Letter from U.S. NRC (Tanya Hood) to Duke Energy Progress (Kim Maza) dated September 29, 2020, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment No. 179 Regarding Departure from Nucleate Boiling Ratio Safety Limit to Address Transition to New Fuel Type (EPID L-2019-LLA-0076)" (ADAMS Accession No. ML20212L594)