UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 8, 2017

Mr. Kelvin Henderson
Senior Vice President
Nuclear Corporate
Duke Energy Progress, LLC
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: DUKE ENERGY PROGRESS, LLC, FOR SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION TO ADOPT DPC-NE-3008-P, REVISION 0, "THERMAL-HYDRAULIC MODELS FOR TRANSIENT ANALYSIS," AND DPC-NE-3009-P, REVISION 0, "FSAR / UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY" (CAC NOS. MF8439 AND MF8440)

Dear Mr. Henderson:

By letter dated October 3, 2016, Duke Energy Progress, LLC (Duke Energy or the licensee) submitted a license amendment request (LAR) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP). The proposed amendment requested review, approval, and adoption into the HNP and RNP Technical Specifications DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," and DPC-NE-3009-P, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology." The LAR supersedes a November 19, 2015, submittal in its entirety, which had submitted DPC-NE-3008-P for review without DPC-NE-3009-P. These methodologies will be used to support the performance of (1) thermal-hydraulic calculations and (2) Final Safety Analysis Report (FSAR) and Updated FSAR Chapter 15 transient analysis as part of the reload design analysis for HNP and RNP. By letter dated October 24, 2016, the U.S. Nuclear Regulatory Commission (NRC) staff issued requests for additional information (RAIs) related to DPC-NE-3008-P. By letter dated November 10, 2016, Duke Energy responded. This letter is related to DPC-NE-3009-P.

The NRC staff has determined that additional information is needed to complete its review. The enclosed RAIs were e-mailed to the licensee in draft form on August 9, 2017. RAI clarification calls were held on August 17 and 28, 2017. During the August 28th call, the licensee agreed to provide responses to the RAIs by October 27, 2017. The NRC staff agreed with this date.

The NRC staff has determined that its documented RAIs (Enclosure 1) contain proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has prepared a redacted, nonproprietary version (Enclosure 2). However, the NRC will delay

The document transmitted herewith contains Sensitive Unclassified Non-Safeguards Information in Enclosure 1. When separated from Enclosure 1, this document is decontrolled.

placing the nonproprietary RAIs in the public document room for a period of 10-working days from the date of this letter to provide Duke Energy the opportunity to comment on any proprietary aspects. If you believe that any information in Enclosure 2 is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10-working days, the nonproprietary RAIs will be made publicly available.

If you have any questions, please contact me at (301) 415-6256 or Dennis.Galvin@nrc.gov.

Sincerely,

Dennis J Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-261 and 50-400

Enclosures:
1. RAI (Proprietary)
2. RAI (Nonproprietary)

cc w/enclosures:
Ms. Tanya M. Hamilton
Site Vice President
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road, M/C HNP01
New Hill, NC 27562-0165

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
3581 West Entrance Road, RNPA01
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cc w/o Enclosure 1: Listserv (10 working days after date of this letter)

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO ADOPT DPC-NE-3008-P, REVISION 0,

"THERMAL-HYDRAULIC MODELS FOR TRANSIENT ANALYSIS," AND DPC-NE-3009-P,

REVISION 0, "FSAR / UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY"

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NOS. 50-400 AND 50-261

Introduction and Regulatory Basis

By letter dated October 3, 2016 (Reference 1), Duke Energy Progress, LLC (Duke Energy or the licensee) (Reference 1) submitted a license amendment request for Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP). The proposed amendment requested review, approval, and adoption into the HNP and RNP Technical Specifications (TSs), DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," and DPC-NE-3009-P, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology." The license amendment request supersedes a November 19, 2015 submittal (Reference 2) in its entirety, which submitted DPC-NE-3008-P for review without DPC-NE-3009-P. These methodologies will be used to support the performance of (1) thermal-hydraulic calculations and (2) Final Safety Analysis Report (FSAR) and Updated FSAR (UFSAR) Chapter 15 transient analysis, as part of the reload design analysis for HNP and RNP. By letter dated October 24, 2016 (Reference 3), the U.S. Nuclear Regulatory Commission (NRC) staff issued requests for additional information related to DPC-NE-3008-P, which was provided as Attachments 4 (proprietary) and 5 (nonproprietary) of the October 3, 2016 (Reference 1), submittal. Duke Energy responded on November 10, 2016 (Reference 4). The section numbers in the questions below refer to DPC-NE-3009-P unless otherwise noted. DPC-NE-3009-P was provided as Attachments 6 (proprietary) and 7 (nonproprietary) of the October 3, 2016 (Reference 1), submittal.

The proposed analysis methodologies are covered by a number of sections of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition" (SRP), including SRP Sections 15.0, "Introduction - Transient and Accident Analyses," 15.0.2, "Review of Transient and Accident Analysis Methods" (References 5 and 6 respectively) and most of the other Chapter 15 sections that relate to specific safety analyses. Related topics in SRP Sections 4.2, "Fuel System Design," 4.3, "Nuclear Design," and 4.4, "Thermal and Hydraulic Design" (References 7, 8, and 9, respectively), are also pertinent. The review guidance in these sections provides the basis for these questions. Generally, the questions below fall into one of several categories:

- Methodology documentation, which is discussed in SRP 15.0.2, Section II.1.

- The physical modeling in the codes used to evaluate the transients, which is discussed in SRP 15.0.2, Section III.2.B.
- Analytical methods for calculating core physics parameters, discussed in SRP 4.3 Sections III.2, III.3, and III.7.
- Analytical methods for calculating core thermal hydraulic parameters, discussed in SRP 4.4 Section III.4, III.5, and III.8.
- The values of parameters assumed in the analytical model, especially core physics parameters and initial/boundary conditions, which is discussed throughout the SRP Chapter 15 sections that correspond to each transient analysis section in the proposed methodology.
- NRC acceptance criteria for reactivity insertion accidents, discussed in Appendix B to SRP 4.2 and Draft Regulatory Guide (DG)-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents" (Reference 10).

The applicable SRP Section or applicable topical report limitation and condition are identified for each question in parentheses.

Requests for Additional Information

1. Section 3.2, "RETRAN-3D," discusses two RETRAN-3D nodalizations. (SRP 15.0.2)
   a. One of the nodalizations divides the vessel into two azimuthal regions. Given that the prediction of the **[[ ]]**[1] is one of the most important parameters affecting the transient response, how does Duke Energy ensure appropriate mixing between these parallel flow paths in the RETRAN-3D model?
   b. The other nodalization combines the steam generator secondary side volume into a single volume, **[[ ]]**.
      i. How does Duke Energy ensure that steam flow modeling is adequate in this model?
      ii. How does Duke Energy ensure that heat transfer between the primary and secondary sides of the steam generator is adequate in this model?

2. Duke Energy proposed in Section 5.0, "Transient Analysis Methods," to use the RETRAN-3D inter-region heat transfer model. As was noted in DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology" (Reference 11), the interphase heat transfer is a user input. Condition 18 on the NRC staff's use of RETRAN-3D (see Reference 12) requires such user-supplied parameters for use in the pressurizer model to be justified. While DPC-NE-3000 discussed a method for determining the heat transfer coefficient based on comparisons to plant data at various pressurizer insurge rates, such a method was not proposed for HNP and RNP. The NRC staff is uncertain of how Duke Energy proposes to ensure that this model is properly biased in the models presented in DPC-NE-3009, and thus how RETRAN-3D Condition 18 is met. Please provide the requested justification for the selection of the inter-region heat transfer. (Conformance with limitation and condition on the use of RETRAN-3D)

3. Duke Energy proposed in DPC-NE-3008 that "[l]icensing applications of the RETRAN-3D models for HNP and RNP may incorporate other uses of non-conducting heat exchangers to

[1] The text between bolded brackets **[[ ]]** contains proprietary information.

model, for example, ambient heat losses." While Section 5.0 of DPC-NE-3009 proposed potential applications for this model, these uses were not described in sufficient detail for the NRC staff to review them. The issue is not that the model is suspect - indeed, being able to add heat to a fluid volume without using a heat conductor is a standard feature of thermal-hydraulic systems analysis codes. However, the future uses were not specified in sufficient detail for the NRC staff to determine whether or not they are acceptable. Please provide additional detail for the NRC staff to complete the review. (SRP 15.0.2)

4. Duke Energy proposed in Section 5.0 to use the RETRAN-3D local conditions heat transfer model. As noted in Condition 28 of the NRC staff's safety evaluation on RETRAN-3D, the local conditions heat transfer model assumes that saturated fluid conditions are present in the fluid volume. However, Duke Energy is interested in applying the model to the **[[** **]]** where, it is noted in Section 3.2.6.5, "Non-Equilibrium Pressurizer," of DPC-NE-3000, **[[** **]]**. It is unclear to the NRC staff how the local conditions heat transfer model is compatible with these phenomena. Please provide additional justification for the use of this model in this scenario. (SRP 15.0.2)

5. Duke Energy proposed in Section 5.0 to use the RETRAN-3D decay heat model. In specifying how this model will be used, Duke Energy did not specify which transients would use low, nominal, or high decay heat assumptions. Please clarify the decay heat assumptions for the transients considered in DPC-NE-3009. (SRP 15.0.2)

6. Section 3.3, "VIPRE-01," proposes the use of the "Modified Barnett" correlation for critical heat flux evaluations of off-rated conditions in VIPRE-01. The cited basis for approval of this correlation is Section 15.3.1.2.5 of DPC-NE-3005, "UFSAR Chapter 15 Transient Analysis Methodology" (Reference 13). The NRC staff found no mention of the Modified Barnett correlation in the version of the referenced topical report it reviewed (Revision 2). However, Section 8.0, "References," lists DPC-NE-3005, Revision 5, which has not been submitted to the NRC. Please provide additional justification for the use of the Modified Barnett correlation or provide evidence that it has been reviewed and approved by the NRC staff for similar applications. (SRP 4.4)

7. Section 3.3 proposes the use of the VIPRE-01 **[[** **]]**. Please provide additional justification for the use of the **[[** **]]** model in these circumstances. (SRP 4.4)

8. In Section 4.0, "Safety Analysis Physics Parameters," Duke Energy proposed that the bounding nature of the safety analysis can be determined by comparing key physics parameter values from previous, "similar" core reload designs to the safety analysis values. How does the proposed method determine that one core reload design is "similar" to

another core reload design? Additionally, how does such an approach demonstrate the amount of margin available between the proposed core design and the safety analysis? (SRP 15.0.2)

9. In Section 4.1, "Generic Parameters," Duke Energy proposed the use of a "conservative relationship between rod position (percent inserted) and normalized reactivity worth." Please identify how this relationship is determined. (SRP 4.3)

10. In Section 4.1, Duke Energy proposed that "perturbed power distributions allowed by the AFD [Axial Flux Difference] and rod insertion limits are considered" for transients in which the initial power distribution significantly impacts the course of the event. It is unclear to the NRC staff what these "perturbed power distributions" are and how they are different from the "core power distributions permitted by AFD and rod insertion limits" discussed earlier in the same paragraph. Please clarify. (SRP 15.0.2, values of core physics parameters in Chapter 15 sections)

11. In Section 4.1, Duke Energy stated that the prompt neutron lifetime is not a key parameter and typical time-in-life values would be used for it. However, in the NRC-approved methodology described in DPC-NE-3001, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology" (Reference 14), this parameter is biased such that the ratio of beta-effective to the prompt neutron lifetime is minimized, which increases the neutron power spike in the event of prompt criticality. Why is this not necessary in DPC-NE-3009? (SRP 15.0.2, values of core physics parameters in Chapter 15 sections)

12. The methods used to determine several key physics parameters in Section 4.2, "Control Rod Worth Calculations" (maximum differential rod worth at subcritical, maximum differential rod worth at power, ejected rod worth) state that adverse power distributions are considered. Please specify the range of adverse power distributions considered in the evaluation of these parameters and how they are generated. (SRP 15.0.2, values of core physics parameters in Chapter 15 sections)

13. The specification of ejected rod worth in Section 4.2 indicates that only certain limiting locations are considered in the analysis. The previously-approved DPC-NE-3001 methodology states that "all possible rods" are analyzed. Without considering all locations, a potentially limiting rod might be missed. Clarify how limiting locations are determined so that a potentially limiting rod is not missed. (SRP 15.0.2, values of core physics parameters in Chapter 15 sections)

14. For some transients discussed in Chapter 5 of the methodology report, Duke Energy states that an aspect of the transient (usually primary and/or secondary pressure response) is bounded by other transients and is, thus, not analyzed. The SRP sections in Chapter 15 for non-loss-of-coolant accident transients state that, if the event is said to be bounded by other transients, the NRC staff should review the justification that the other transient bounds the one under review. Without justification for which transients are bounding (especially with respect to primary and secondary pressure response), the NRC staff is not capable of performing this review. Please provide a discussion of the transients that are known or expected to be bounding, especially with regard to primary and secondary system pressure response, and justification of why those transients bound the other transients in the methodology. (SRP 15.0: "The reviewer evaluates licensees' claims that individual AOOs [anticipated operational occurrences] and postulated accidents are limiting or nonlimiting, or

bounded by other AOOs and postulated accidents, with particular attention to the bases used for comparison.")

15. In Section 5.0, Duke Energy proposed to use the statistical core design (SCD) methodology described in the NRC-approved DPC-NE-2005 (Reference 15) methodology for evaluation of departure from nucleate boiling (DNB). This methodology relies on subchannel analyses performed at a number of statepoints to determine the statistical core design limit, and as such this limit is only valid when the analysis is performed at or near these statepoints. If the analysis statepoint falls outside of the range of the statistical design limit, a non-SCD analysis is performed. The NRC staff has several questions about both the SCD and non-SCD analyses.
    a. Duke Energy proposed that the SCD conditions could be updated to bound statepoints that would otherwise require non-SCD calculations. The NRC staff is unclear on how this is consistent with Condition 1 on the approval of DPC-NE-2005, which states that "[Duke] committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate." How will such justification be provided in the event that the SCD statepoints are adjusted? (Conformance with limitation and condition on DPC-NE-2005)
    b. Duke Energy stated that "the final selection of SCD or non-SCD methodology for a given transient will depend on the actual analysis results and may differ from the expected selection as presented here." However, insufficient information has been provided to the NRC staff to determine the acceptability of the parameter biasing for transients that are currently envisioned as SCD transients to be performed as non-SCD analyses. This is because the tables in Section 5 of the methodology report that provide the parameter biasing simply say "SCD" for key parameters in the DNB ratio (DNBR) evaluation. If the flexibility to switch between SCD and non-SCD analyses is required, additional information must be provided to justify the modeling in either scenario for each transient. (SRP 15.0.2)

16. Section 5.0 discussed key parameter selection and biasing, on a generic basis, for all of the transients. The NRC staff have the following questions about this biasing. (SRP 15.0.2, values of initial and boundary conditions in Chapter 15 sections)
    a. Transients discussed as non-SCD are stated to have uncertainties included in the biasing of initial and boundary conditions. What uncertainties are included and how are they represented in the analysis?
    b. Does the maximized or minimized reactor coolant system (RCS) flow rate include instrument and other uncertainties?
    c. How is the best estimate of core bypass flow determined?
    d. Why is steam generator tube plugging not just assumed to be zero when it is biased "low"? What is the basis for plugging values assumed that are less than 1%?

17. For the increase in feedwater flow transient discussed in Section 5.1.1, "Increase in Feedwater Flow," Duke Energy stated that "where applicable, a coincident step decrease in main feedwater temperature is also assumed to occur." It is not clear to the NRC staff how applicability is determined, or how the step change in feedwater temperature is determined. Please clarify/justify. (SRP 15.0.2)

18. The analysis in Section 5.1.3, "Inadvertent Opening of a Steam Generator Relief or Safety Valve," is said to use the same method as that described in Section 5.1.4, "Steam System Piping Failure," with an adjusted break area. What is the basis for the area assumed in the analysis? Are break sizes assumed consistent with the sizes of the steam dumps, steam generator power operated relief valves, or steam generator secondary safety valves? If only one valve is analyzed, how is it determined that the chosen valve is limiting? (SRP 15.0.2, values of initial and boundary conditions in Chapter 15 sections)

19. The steam line break analysis described in Section 5.1.4 of the methodology is nominally performed at hot zero power (HZP) conditions, with an "evaluation" at hot full power (HFP) conditions to assure that HZP is limiting. The NRC staff are unclear on what is involved in this evaluation – is a full analysis performed at HFP conditions? Please clarify, and provide justification for performing anything less than a full analysis with a HFP initial condition. Please also discuss why an intermediate power is not potentially limiting as an initial condition. (SRP 15.1.5.II (Reference 16), SRP Acceptance Criteria: "The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident.")

20. Section 5.1.4.4, "Core Power Distributions and Reactivity Feedback," defined the limiting RETRAN-3D statepoint as the "time of maximum core average surface heat flux, with corresponding values of core inlet mass flow rate, core inlet temperature, and core exit pressure." The conditions at this statepoint are passed to VIPRE-01 for further thermal-hydraulic analysis. Given the complexity of the downstream calculations and the number of interrelated core parameters, please justify why the RETRAN-3D statepoint corresponding to the maximum core average surface heat flux is known to be limiting for the DNBR or centerline fuel melt (CFM) evaluations. (SRP 4.4)

21. Section 5.1.4.1, "RETRAN-3D Models and Options," proposed the use of the RETRAN-3D general transport model to track boric acid injected by the emergency core cooling system (ECCS). Section 4.3, "Reactivity Coefficients," indicates that differential boron worth is considered to be a function of reactor coolant temperature and other variables. Given that **[[**
]], the NRC staff was unsure of how the differential boron worth was determined for this application. Please clarify. (SRP 15.0.2, values of core physics parameters in Chapter 15 sections)

22. Section 5.1.4.2, "Primary Systems and Components," proposed to model cold leg accumulators with the built-in RETRAN-3D accumulator model. However, the NRC staff was unclear on the initial conditions of the accumulators (including boric acid concentration, initial level, cover gas pressure) and whether the potential for incursion of noncondensible gases into the RCS is considered in the steam line break analysis. Please provide additional information on the modeling and initial/boundary conditions of the accumulators. (SRP 15.0.2, values of initial and boundary conditions in Chapter 15 sections)

23. In the steam line break analysis of Section 5.1.4.1 in the methodology report, liquid carryout is suppressed **[[**
]]. In the NRC-approved DPC-NE-3001 steam line break model, an additional step of **[[**
]], which appears to be applicable to HNP and RNP. Why is this extra stop not needed in the DPC-NE-3009 model for HNP and RNP? (SRP 15.0.2)

24. In the steam line break analysis of Section 5.1.4.3, "Secondary Systems and Components," in the methodology report, it is stated that auxiliary feedwater (AFW) is modeled conservatively and is assumed to be terminated by operator action. (SRP 15.1.5.II, SRP Acceptance Criteria: "The analyses should take account of the effect that loss of offsite power has ... the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents"; SRP 15.1.5.III: "the availability of the auxiliary feedwater system to supply adequate auxiliary feedwater flow to the intact steam generators during the accident and the subsequent shutdown condition is evaluated.")
    a. In the plant analyses of record as presented in the HNP and RNP (U)FSARs, AFW flow starts at the time of break initiation. Please discuss why it is acceptable to eliminate this conservatism.
    b. No details were provided on the operator action needed to terminate AFW or how (or even if) it is incorporated into the analysis. Please provide additional details for the NRC staff's evaluation.

25. For the cycle specific evaluation of the steam line break presented in Section 5.1.4.7, "Cycle-Specific Evaluation," of the methodology report, Duke Energy stated that **[[**

    ]]. Please provide additional details on this analysis. (SRP 15.0.2)

26. Section 5.1.4.2 describes the safety injection system model for the steam line break analysis methodology; however, this section does not describe how the safety injection pumps are modeled. Considering the effect of boron injection from the ECCS is important for modeling a main steam line break, Duke Energy should provide the following information to supplement the methodology: (SRP 15.1.5.III: "Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements: ... Emergency core cooling system (ECCS)")
    a. A discussion of the safety injection pump modeling.
    b. A discussion of the assumptions used for safety injection boric acid concentration and temperature.

27. In the HNP steam line break demonstration analysis presented in Section 6.1, "Steam System Piping Failure (HNP)," (as well as the RNP demonstration analysis presented in Section 6.2, "Steam System Piping Failure (RNP)"), there is a power spike as the reactor returns to criticality. This is in contrast to the HNP analysis of record in the FSAR, where the power increases quickly but smoothly. Do the HNP and RNP demonstration analyses go prompt critical? If so, is the case that results in prompt criticality expected to be limiting or did Duke Energy evaluate other conditions to determine the limiting scenario? (SRP 15.1.5.II, SRP Acceptance Criteria: "The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.")

28. Section 5.2.3, "Loss of Non-Emergency AC Power to the Station Auxiliaries," presents the analysis methodology for the loss of non-emergency alternating current power to the station auxiliaries transient. NRC staff are asked in SRP 15.2.1-15.2.5 Section III.3 (Reference 17) to review the effects of single active system or component failures that may affect the course of the transient. From the licensee's submittal, it is clear that the limiting failure is assumed

to occur in the AFW. However, it is unclear exactly what the limiting failure is or how the AFW is modeled for the transient. The NRC staff has the following questions to clarify this issue. (SRP 15.2.1-15.2.5)

a. The licensee's submittal states that "Auxiliary feedwater actuates on either low steam generator level or loss of non-emergency AC power with a conservative delay." Does this mean that there is a delay only when AFW actuates on loss of non-emergency AC power, or is there also one when it actuates on low steam generator level?
b. The licensee's submittal states that "Minimum flow from at least one auxiliary feedwater pump is assumed." It is unclear what is implied about the state of the AFW pumps from this statement, since "at least one auxiliary feedwater pump" having minimum flow says nothing about the other pumps. Please clarify.
c. What is considered to be the limiting failure within the system? Are a variety of failures investigated in the analysis to determine what is most limiting?

29. In the feedwater system pipe break analysis methodology presented in Section 5.2.5, "Feedwater System Pipe Break," the short term core cooling and peak primary pressure cases assume core physics parameters consistent with beginning of cycle (BOC) conditions. The initial phases of the feedwater line break result in an overcooling event until the faulted steam generator is depleted. Because of this, it is not clear that BOC core physics parameters would provide the most limiting response, especially for the short-term core cooling case, where a more negative moderator temperature coefficient (MTC) could provide for a worse core power response. Please justify the choice of core physics parameters for the short-term core cooling and peak primary pressure cases. (SRP 15.2.1-15.2.5.II.3.C, "The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution."; SRP 15.2.1-15.2.5.III.5, "The values of system parameters and initial core and system conditions as input to the model are reviewed by the organization responsible for reactor systems. Of particular importance are (A) the values of reactivity coefficients and control rod worths in the applicant's analysis and (B) the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates the applicant's justification showing that the core burn-up selected yields the minimum safety margins.")

30. In the feedwater system pipe break analysis methodology presented in Section 5.2.5, the licensee credits operator actions to terminate pumped safety injection and AFW. However, since no details are provided, it is unclear whether or not these actions are appropriate, and how they are modeled in the analysis. Please provide additional information to clarify the modeling of these operator actions. (SRP 15.2.1-15.2.5.III.1.E: "The SAR [safety analysis report] (or DCD [design control document]) description of these transients is reviewed for the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed for: ... The extent to which operator actions are required.")

31. In the partial or complete loss of forced reactor coolant flow analysis methodology described in Section 5.3.1, "Partial or Complete Loss of Forced Reactor Coolant Flow," it is mentioned that the peak pressure evaluation assumes that offsite power is maintained but no such discussion is provided about the core cooling evaluation. Please provide a discussion of whether or not it is conservative from a DNBR standpoint to model a loss of offsite power for the loss of flow transient. (SRP 15.3.1-15.3.2.III.2 (Reference 18): "For new applications,

LOOP [loss of offsite power] should not be considered a single failure; each loss of flow transient should be analyzed with and without a LOOP in combination with a single active failure.")

32. In the complete loss of flow demonstration analysis presented in Section 6.3, "Complete Loss of Forced Reactor Coolant Flow (HNP)," of the report, it appears that the reactor coolant pump coastdown curve is much less conservative than the one assumed in the analysis of record based on the core flow rate plots provided (the flow reaches a minimum value of 60% in 10 seconds, versus 25% in the analysis of record). Please provide comparisons between plant measurements of the pump coastdown curve and the curve assumed in the analysis so that the NRC staff may verify that appropriately conservative boundary conditions are used. (SRP 15.3.1-15.3.2.III.5: "Time-related variations of the following parameters should be reviewed for consistency: ...core and recirculation loop coolant flow rates")

33. In the locked rotor analysis methodology presented in Section 5.3.2, "Reactor Coolant Pump Shaft Seizure (Locked Rotor) or Shaft Break," the use of the VIPRE-01 fuel pin heat conduction model was proposed for DNBR evaluations. However, no details were provided on how this model is initialized or what modeling options are used. Is the model used in the same manner as it is for the rod ejection analysis? More detail is needed for the NRC staff to appropriately review this application of the model. (SRP 15.0.2)

34. In the HNP locked rotor demonstration analysis presented in Section 6.4, "Reactor Coolant Pump Locked Rotor (HNP)," it is stated that "the selection of the affected loop has a negligible effect on the transient results." However, the NRC staff did not see a basis provided for this claim. Duke Energy should provide additional justification, preferably in the form of sensitivity studies, for the statement that the choice of the affected loop has a negligible effect. (SRP 15.3.1-15.3.2.III.2: "The SAR or DCD must present a quantitative analysis of the most limiting loss of flow transient.")

35. Duke Energy proposed to analyze minimum DNBR using the SCD methodology for the uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal from subcritical or low power transient, as described in Section 5.4.1, "Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition." Considering the power starts at zero or a very low level for this transient, please justify that the analysis will fall into the range of applicability of the SCD methodology. What action would be taken if a particular case did not fall within the SCD methodology range? (SRP 4.4, conformance with the DPC-NE-2005 methodology)

36. In many sections of the methodology, centerline fuel melt evaluation cases are analyzed with the same initial and boundary conditions as minimum DNBR evaluation cases. Please justify why this is appropriate, considering that high fuel temperature is frequently at odds with assumptions for minimizing the DNBR. (SRP 15.0.2, values of initial and boundary conditions in Chapter 15 sections)

37. The primary difference between the withdrawal of a single full-length RCCA event presented in Section 5.4.4, "Withdrawal of a Single Full-Length RCCA," and the uncontrolled RCCA bank withdrawal at power event presented in Section 5.4.2, "Uncontrolled RCCA Bank Withdrawal at Power," is that the former provides a much more localized effect than the latter, resulting in severe radial power peaking. In the RNP demonstration analysis for this

transient, it is unclear whether these effects are considered. For example, the analysis of record in the RNP UFSAR predicts DNB in the immediate vicinity of the withdrawn rod due to highly localized peaking, while the demonstration analysis predicts a minimum DNBR of 1.571. Therefore, please discuss in more detail how it is ensured that the core power distribution is appropriately calculated using the methodology discussed in Section 5.4.4, particularly with regard to local power distributions in the immediate vicinity of the withdrawn control rod. Are pin-level effects considered? (SRP 15.0.2, SRP 15.4.3.III.2 (Reference 19))

38. What reactivity insertion rate was assumed in the withdrawal of a single full-length RCCA demonstration analysis presented in Section 6.6, "Withdrawal of a Single Full-Length RCCA (RNP)," of the methodology report? (SRP 15.4.3.III.1)

39. Duke Energy stated that the peaking resulting from statically misaligned rods (discussed in Section 5.4.5, "Static Misalignment of a Single Full-Length RCCA"), is analyzed to confirm that DNB or CFM will not occur. No transient systems analysis is performed. The licensee stated that the peaking analysis is a steady-state, three-dimensional power peaking analysis, which leads the staff to believe that the analysis is performed using SIMULATE-3; however, this was not specified by the licensee. Please specify the name of the code used for the peaking analysis? (SRP 15.0.2)

40. For the dilution transient discussed in Section 5.4.6, "CVCS [Chemical and Volume Control System] Malfunction that Decreases the Boron Concentration of the Reactor Coolant," it was not clear to the staff whether the acceptance criteria for RNP are based on the time of the alarm or the time of the dilution initiation. Please provide additional detail on the proposed method and how it relates to the current licensing basis for the dilution event for RNP. (SRP 15.0.2)

41. For the dilution transient discussed in Section 5.4.6, it is unclear to the NRC staff how the dilution analysis is actually performed in DPC-NE-3009. Please provide a discussion describing how the time to reach criticality during a dilution event is calculated, including any formulas used to determine the actual time to criticality as well as any codes or methods used to determine the critical boron concentration in various modes of operation. Please also discuss how the calculation accounts for the different acceptance criteria based on alarms or dilution initiation. (SRP 15.0.2)

42. The fuel assembly misloading analysis methodology described in Section 5.4.7, "Inadvertent Loading of a Fuel Assembly in an Improper Position," has insufficient detail for the NRC staff to review it. Though it is apparent from the discussion that part of the evaluation hinges on whether or not a misloading would be detected during refueling and power ascension operations and testing, the acceptance criteria used to detect a misloading was not clearly defined. Furthermore, the method for evaluating DNBR and CFM for misloadings that were not detectable was not discussed in the report. Please provide the specific criteria used to detect a misloaded assembly and the method used when misloadings are not detected. (SRP 15.0.2)

43. In Section 5.4.8, "Spectrum of RCCA Ejection Accidents," Duke Energy stated that the rod ejection analysis "must demonstrate that the radially averaged fuel pellet enthalpy does not exceed the NRC acceptance criteria at any location." What NRC acceptance criteria will be used in the analysis? The acceptance criteria provided in DG-1327 are a function of

cladding differential temperature or excess hydrogen, and the ones provided in SRP 4.2 Appendix B are a function of oxide thickness. In addition, it is not apparent to the NRC staff that the method presented for evaluation of the rod ejection accident is capable of predicting these parameters. Please specify the acceptance criteria being used and describe how the evaluation method is applicable to the selected acceptance criteria. (SRP 4.2 Appendix B, DG-1327)

44. Duke Energy's proposed analysis methodology for the rod ejection accident considers combinations of BOC or end of cycle core physics conditions with HZP or HFP initial conditions. DG-1327 specifically states that intermediate times in life and power levels should be considered in rod ejection analyses. Please justify how the chosen conditions encompass the set of limiting initial and boundary conditions for all of the quantities of interest (e.g., minimum DNBR, enthalpy deposition, etc.) for a rod ejection accident. (SRP 4.2 Appendix B, DG-1327)

45. In Section 5.4.8.1, "Nuclear Analysis," Duke Energy states that during the trip following a rod ejection accident the control rods fall into the core "at a speed that satisfies the maximum rod drop time in the technical specifications." Since TSs are derived from the accident analysis, please specify if the time used in the analysis is consistent with the time specified in the TSs or is the rod drop time shorter? (SRP 15.4.8.III.1.D (Reference 20))

46. In Section 5.4.8.1, Duke Energy stated that the SIMULATE-3K heat conduction model is used to calculate the temperature distribution within the pin as well as the transport of heat from the fuel, through the gap and cladding, and into the coolant. The model is also used to **[[** **]]**. It is unclear to the NRC staff whether and how this fuel temperature calculation feeds back to the neutronics solution. Are individual pin radial temperature distributions determined and used to calculate Doppler reactivity effects? If average fuel temperatures are used to evaluate Doppler reactivity effects, please justify why this is appropriate. (SRP 15.0.2)

47. In Section 5.4.8.2, "Fuel Temperature and Enthalpy Calculations," Duke Energy proposed the use of **[[** **]]**. Is the **[[** **]]** incorporated into VIPRE-01? If so, how? If not, please provide additional justification of how **[[** **]]**. (SRP 4.2, SRP 15.4.8.III.1.E, SRP 15.4.8.III.2.B)

48. It is unclear to the NRC staff how the proposed methodology in Section 5.4.8.2 for rod ejection accident fuel temperature calculations counts the number of fuel pins that fail to meet the enthalpy deposition or fuel centerline melting criteria. What happens in the analysis methodology when peak pins fail to meet acceptance criteria? How are pins that fail to meet the enthalpy deposition or fuel centerline melting criteria tallied? Please provide additional detail. (SRP 15.0.2, SRP 15.4.8.III.1.E, SRP 4.2 Appendix B, DG-1327)

49. In Section 5.4.8.4, "RETRAN-3D System Thermal-Hydraulic Calculations," Duke Energy states that the failure of the control rod drive mechanism housing that causes the rod ejection is also assumed to create a hole in the reactor vessel head the size of the control rod drive shaft. However, the boundary conditions of the hole or how it is modeled are not

specified. Please provide additional information describing how the hole is modeled and the boundary conditions used. (SRP 15.4.8.III.1.E)

50. In Section 6.7, "Spectrum of RCCA Ejection Accidents (HNP)," for the rod ejection accident demonstration analysis at HNP, Duke Energy stated for the BOC/HZP case that "the high boron concentration necessary to achieve a positive MTC was found to suppress the power excursion," so the case was analyzed with an MTC of zero percent millirho per degree Fahrenheit. To appropriately justify and demonstrate the conservatism of the approach used in the demonstration analysis, please provide a summary, including a timeline of events and plots, of the BOC/HZP RCCA ejection analysis that achieved a positive MTC. (SRP 15.4.8.III.1.C)

51. In Section 6.8, the rod ejection accident demonstration analysis at RNP, the BOC/HFP case does not trip on a flux-based reactor trip and the transient is simulated for a longer duration with RETRAN-3D, per the method discussed in Section 5.4.8.4 of the methodology report. Please provide plots of additional results from the reactor coolant system calculation for the NRC staff to understand how the transient proceeds in the RETRAN-3D calculation, including RCS temperatures, pressurizer pressure, and pressurizer level. (SRP 15.4.8.II)

52. Section 6.8, "Spectrum of RCCA Ejection Accidents (RNP)," also provides a set of analyses to justify the use of a fuel pin gas gap conductance model **[[**

]]. Therefore, please provide additional justification for why the use of **[[** **]]** is more appropriate than **[[** **]]**. (SRP 15.4.8.III.1.E)

53. In the methodology presented for analysis of the inadvertent operation of the ECCS in Section 5.5.1, "Inadvertent Operation of the Emergency Core Cooling System," of the report, it is unclear to the NRC staff whether boron injection is modeled during the transient. The HNP analysis of record for this event explicitly considers the effects of boric acid on reactivity. If boron injection is modeled in the proposed methodology, please discuss the assumptions used related to boron concentrations, boron worth, and boron transport. If not, please discuss how the proposed method appropriately accounts for the effects of inadvertent operation of the ECCS on the core power level. (Consistency with HNP current licensing basis (CLB))

54. Section 5.6.1, "Inadvertent Opening of a Pressurizer Relief or Safety Valve," of the methodology presents the proposed methodology for analyzing the inadvertent opening of a pressurizer relief or safety valve. It is unclear to the staff whether both power-operated relief valves and safety valves are considered as potential failures, or if the limiting valve is selected in advance. It is also unclear how the initiating valve failure is modeled. Please clarify. (SRP 15.6.1 (Reference 21))

55. The SCD methodology is used to evaluate the core cooling capability for the inadvertent opening of a pressurizer relief or safety valve transient in Section 5.6.1. Given that this event has the potential for core uncovery, please justify why it is appropriate to evaluate the event from nominal, rather than conservative initial conditions. Is a separate core uncovery analysis performed? (SRP 15.0.2, SRP 15.6.1.I.1.B)

56. For the steam generator tube rupture, the licensing basis analysis presented in the HNP FSAR focuses almost exclusively on the margin to steam generator overfill, which is not presented as an acceptance criterion in Section 5.6.2, "Steam Generator Tube Rupture," of the methodology report. Please discuss whether steam generator overfill will be analyzed with the proposed method, with a justification for why the analysis is or is not needed. This discussion should consider single failure assumptions for the margin to overfill analysis, consistent with the HNP FSAR. (Consistency with HNP CLB)

57. The current licensing basis for HNP relies on several operator actions to recover from a steam generator tube rupture. Please discuss whether operator actions are modeled to terminate the steam generator tube rupture in the DPC-NE-3009 methodology. If operator actions are required, please describe and justify how each operator action is modeled. (There is no SRP section specifically for the review of steam generator tube ruptures, but the requirement to identify operator actions credited in transient and accident analyses is in SRP 15.0.I.6)

58. Please clarify whether any new operator actions are being credited as part of the license amendment request, and if so, include for each action how operators will be trained on the action, how the action will be incorporated into the procedures, and how the time allowed for the action was derived.

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9. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan," Section 4.4, "Thermal and Hydraulic Design," Revision 2, March 2007 (ADAMS Accession No. ML070550060).

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Subject: DUKE ENERGY PROGRESS, LLC, FOR SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION TO ADOPT DPC-NE-3008-P, REVISION 0, "THERMAL-HYDRAULIC MODELS FOR TRANSIENT ANALYSIS," AND DPC-NE-3009-P, REVISION 0, "FSAR / UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY" (CAC NOS. MF8439 AND MF8440) DATED SEPTEMBER 8, 2017

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