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CNS-14-081

June 27, 2014

10 CFR 50.90

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

Subject: Duke Energy Carolinas, LLC ("Duke Energy" or "Duke")
McGuire Nuclear Station, Units 1 and 2; Docket Nos. 50-369, 50-370
Catawba Nuclear Station, Units 1 and 2; Docket Nos. 50-413 and 50-414
License Amendment Request for Methodology Report DPC-NE-3001-P, Revision 1, *Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology* (Proprietary)
Response to NRC Request for Additional Information (RAI)
(TAC Nos. MF3119, MF3120, MF3121, and MF3122)

References:

1. Letter from Duke Energy to NRC, License Amendment Request for Methodology Report DPC-NE-3001-P, Revision 1, *Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology* (Proprietary), dated November 14, 2013

2. Letter from NRC to Duke Energy, Catawba Nuclear Station, Units 1 and 2 and McGuire Nuclear Station, Units 1 and 2 - Request for Additional Information Regarding License Amendment Request Re: Methodology Report DPC-NE-3001-P, Revision 1, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology (TAC Nos. MF3119, MF3120, MF3121, and MF3122), dated May 22, 2014

The Reference 1 letter requested NRC review and approval of proposed changes to the Facility Operating Licenses (FOLs) based on DPC-NE-3001-P, *Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology*. The Reference 2 letter formally provided RAIs which were discussed in a telephone conference call between Duke Energy and the NRC on May 6, 2014. The purpose of this letter is to provide Duke Energy's docketed responses to these RAIs. Attachments A and B to this letter provide the proprietary and non-proprietary versions, respectively, of the responses to these RAIs. The format of Attachments A and B is to restate each RAI, followed by its response.

As DPC-NE-3001-P contains information that is proprietary to Duke Energy, in accordance with 10 CFR 2.390, Duke Energy requests that this information be withheld from public disclosure. An affidavit is included (Attachment C to this letter) from Duke Energy attesting to the proprietary nature of the information in this RAI response submittal. The specific information that is proprietary to Duke Energy is identified in the submittal.

Attachment A to this letter contains proprietary information.
Withhold from public disclosure under 10 CFR 2.390.
Upon removal of Attachment A, this letter is uncontrolled.

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NRR

This RAI response submittal does not affect the conclusions of the Regulatory Evaluation or the Environmental Consideration which were provided in the Reference 1 letter.

Pursuant to 10 CFR 50.91, a copy of this RAI response submittal has been forwarded to the appropriate State of North Carolina and State of South Carolina officials.

There are no regulatory commitments contained in this letter or its attachments.

If you have any questions or need additional information on this matter, please contact L.J. Rudy at (803) 701-3084.

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

Executed on June 27, 2014.

Very truly yours,

Kelvin Henderson
Vice President, Catawba Nuclear Station

LJR/s

Attachments

xc (with all attachments):

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Attachment B
Duke Non-Proprietary Responses to NRC Request for Additional Information for Revision 1 to DPC-NE-3001-P

Duke Responses to DPC-NE-3001, Rev. 1 NRC RAI

1. The main change made to Section 4.2.1.2, described in Change 4-1, proposes the use of Version 2 of SIMULATE-3K. The new models listed on page A-11, especially including burnup-dependent gap conductance, fuel thermal conductivity, and radial power profile. Were benchmarks of Version 2 of SIMULATE-3K performed relative to Version 1. If so, please provide a summary of the benchmarks. If not, please justify why benchmarking is not needed.

   Response:

   Duke has performed rod ejection accident benchmark calculations at both HFP and HZP end-of-cycle (EOC) conditions between SIMULATE-3K versions 1 and 2 to assess the differences in the transient core power response produced by each version of the code. Key input parameters assumed in the SIMULATE-3K version 1 and 2 rod ejection transient analyses are presented in Table RAI 1-1. The input values selected are expected to bound core reload values.

Table RAI 1-1
EOC HFP and HZP Key REA Input Parameter

Parameter	EOC HFP	EOC HZP
Ejected Rod Worth (pcm)	200	900
Beta-effective	0.0040	0.0040
Doppler Temp. Coeff. (pcm/°F)	-1.20	-1.20
Moderator Temp. Coeff. (pcm/°F)	-10.0	-10.0
Shutdown Margin (pcm)	< 210	< 210

   Figures RAI 1-1 and RAI 1-2 show a comparison between the version 1 and 2 core transient power response for the HFP and HZP cases, respectively. Good agreement between the transient power response of the two code versions is observed. Table RAI 1-2 shows maximum power level, time of maximum power, and peak FΔH and Fq peaking factors prior to control rod insertion. The deviation in the time of peak power between the two code versions is [

   ] Overall, the agreement between the two SIMULATE-3K code versions is very good.

   Benchmark results for version 2 of the code were not originally presented because SIMULATE-3K version 2 was previously approved in DPC-NE-3005-PA, "Oconee Nuclear Station UFSAR Chapter 15 Accident Analysis Methodology", Rev. 4a, and because there has been extensive benchmarking of the neutronics and thermal-hydraulic models performed by Studsvik Scandpower (the code vendor) for each version of the code as described in section 4.2.1.3, "SIMULATE-3K Code Verification".

Table RAI 1-2
EOC Rod Ejection Key Parameter Summary

2. The changes to Section 4.2.2.2, described in Change 4-3, allows for the use of [ ] Since the [ ] model is described in detail in DPC-NE-3000-PA, please provide additional details about the [ ] model.

   Response:

   [

   ]

Figure: RAI 2-1

3. The technical justification for Change 4-13 states "The power level uncertainty has been reduced from 2% to approximately 0.3% as part of the measurement uncertainty recapture (MUR) uprate at McGuire and Catawba." While McGuire has received approval for an MUR uprate, Catawba has not. Please clarify this statement as appropriate.

   Response:

   The statement that the power level uncertainty has been reduced from 2% to approximately 0.3% as part of the measurement uncertainty capture uprate at both McGuire and Catawba is incorrect since the Catawba MUR has not yet been approved. Power level uncertainties of 0.3% and 2.0% are currently applicable for McGuire and Catawba, respectively.

   The original technical justification for change 4-13 is modified for clarification. The revised technical justification is as follows: The intent of the change is to allow application of a power level uncertainty other than 2.0% if approved by the NRC as part of license amendment request, such as an MUR uprate. Application of the power level uncertainty will be applied to the rated thermal power level.

4. One of the revisions to Section 4.3.2, described in Change 4-14, allows for the use of both the high flux and high flux positive rate trips for the rod ejection analysis. Please justify why it is now reasonable to credit the high flux positive rate trip when it was not previously credited.

   Response:

   Ejected rod worths assumed in the original licensing analysis produced transient power responses that caused the high flux trip setpoint to be exceeded. However, if ejected rod worths are reduced to more representative bounding values, the high flux trip setpoint for the full power cases is not always achieved. The high flux positive rate trip must be credited for these cases in order for a reactor trip signal to be produced.

5. There are several changes made to Section 5.2.1.2, contained in Appendix B to the licensee's submittal, which do not have a technical justification in Appendix A. The staff therefore has the following questions:

   5.a The existing subsection discussing steam generator (SG) renodalization is amended such that it only applies to the hot zero power (HZP) steam line break (SLB) case. Is SG renodalization required for the hot full power (HFP) SLB case? Why or why not?

   Response:

   No, SG renodalization is not required for the HFP SLB case. The HZP case required use of a single volume steam generator due to stability issues that are not present at HFP. Therefore, the nodalization reviewed and approved in DPC-NE-3000-PA remains acceptable for use with the HFP SLB case.

5.b In the subsection regarding [ ] for the HFP SLB analysis, it is stated that [

] What impact does this modeling change have on the long-term cooling capabilities of the faulted steam generator?

Response:

Long-term core cooling is not a concern for HFP SLB since auxiliary feedwater is available to the three remaining intact generators. However, if a long-term cooling acceptance criterion were analyzed, retaining more secondary inventory in the faulted generator for boil off would be conservative by increasing any return to power that might occur. This is because the higher power requires more auxiliary feedwater to remove the additional heat. However, any return to power would be at core temperatures lower than the initial temperatures and be of short duration. In contrast, the feedwater line break event, which also results in a dried out steam generator, undergoes an almost immediate increase in core temperatures and is therefore more limiting with respect to long-term heat removal.

5.c Section 5.2.1.1 notes that one of the major differences between Catawba and McGuire is in their steam generators. Given that this is the case, are the model modifications proposed in the [ ] applicable to and/or necessary for both sites? When the changes are made, are they then consistent between the two sites?

Response:

There are 2 modeling changes described in Section 5.2.1.2 for [

] Duke will revise the wording to clarify as shown below:

Appendix B of LAR, Section 5.2.1.2 (additions underline, deletions strikethrough)

[

]

5.d One of the subsections discusses the [

] According to EPRI NP-1850-CCM, Volume 5, EPRI's RETRAN-02 modeling guidelines, the primary purpose of [

] Given that [

] is extremely important for a steam line break analysis, please discuss the impact, if any, that this change will have on the accuracy of the HFP SLB analysis.

Response:

Enthalpy transport is a method by which RETRAN-02 accounts for energy addition (or removal) between a volume inlet and exit and provides a more accurate solution given some basic model assumptions. At steady-state conditions, the enthalpy transport effect is that the volume enthalpy is the average of the inlet and exit enthalpies. The model was developed to provide a more representative enthalpy value for heated/cooled volumes, which gives more accurate volume temperatures for single phase liquid conditions. For this reason, it is used in the core and the primary and secondary side of steam generator tubes where heat transfer effects are important. On the secondary side, the improved enthalpy gives a more accurate quality and steady-state liquid mass inventory. The model is based on the solution of mass and energy conservation equations between the volume center and the exit junction under normal flow conditions and a known heat source from the previous time step conditions. When the junction flow reverses the assumptions used in the solution for the enthalpy transport model are no longer applicable, which can result in anomalous results. Provided flow is in the "forward" direction, the model is applicable and works as intended.

For a SLB, the upper volumes of the steam generator dryout and flow reversal occurs in some secondary junctions. These phenomena violate the assumptions of the enthalpy transport model and can result in artificial superheating of the volumes that dryout, where their temperatures can exceed the hot leg temperature, leading to reverse heat transfer. The solution generally fails and terminates under these conditions. Enthalpy transport is enabled on the primary side tube junctions and [ ]
This results in saturated secondary conditions following dryout (vs. the anomalous superheated condition with enthalpy transport enabled) and therefore has a negligible effect on heat transfer than if the enthalpy transport model worked as intended.

6. A subsection describing the HFP core physics parameters is added to Section 5.2.2.1.

6.a It is stated that "The limiting core physics parameters are dependent on the break size." How is this dependence characterized? Is it that the parameter which is limiting changes depending on the break size? Please clarify.

Response:

For a given break size, there is a limiting MTC value that will result in the highest core power excursion prior to generating a high flux reactor trip signal. Therefore, MTC and break size are iterated upon, in 1 pcm/°F and 0.1 ft^2 increments, respectively. These increments are sufficient to determine the worst combination that will either generate the highest core power prior to incurring a reactor trip for CFM or the minimum DNBR during the event. The worst combination of break size and MTC can be different for each acceptance criterion and break location.

6.b It is also stated that "The moderator temperature coefficient (MTC) is determined by sensitivity analyses within the range of the least negative beginning-of-cycle value and the most-negative end-of-cycle value." What kind of sensitivity analysis is performed on the MTC, and how does that sensitivity analysis inform the SLB analysis?

Response:

See the response to question 6.a.

7. In Section 5.2.2.2, both the HZP and HFP SLB cases are revised such that the three-dimensional power distribution will be used as an input to the centerline fuel melt limit evaluation. Please describe in more detail how this will be accomplished.

Response:

Centerline fuel melt linear heat generation limits are generated for each unique fuel type in the reactor core using an NRC-approved fuel rod mechanical analysis methodology and fuel performance code. For example, the reload fuel rod design methodology used for analysis of Westinghouse RFA fuel is described in Chapter 4 of DPC-NE-2009-P, Revision 3a, "Westinghouse Fuel Transition Report". The fuel performance code used with this methodology is PAD.

RETRAN-02 is used to determine the core power response and the limiting state point conditions following a HFP or HZP steam line break. Core power level and thermal-hydraulic conditions from the RETRAN-02 systems analysis are used in subsequent SIMULATE-3 three-dimensional analyses to determine core peaking factors.

For the HFP SLB analysis, three-dimensional power distributions are generated with SIMULATE-3 [

] Three-dimensional Fq power distributions calculated by SIMULATE-3 are then compared against centerline fuel melt linear heat generation limits to confirm positive margin exists to the limit. Prior to comparison, the predicted Fq power distribution is increased by uncertainty factors as described in section 4.5 of the NRC-approved methodology report DPC-NE-2011-PA, Rev. 1a, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors". If a centerline fuel melt limit is exceeded, then either the HFP SLB transient analysis is evaluated, re-analyzed or the core redesigned to obtain acceptable peaking factors.

For the HZP SLB case, a three-dimensional Fq power distribution is generated using SIMULATE-3 as described in Section 5.2.2.2 and compared against centerline fuel melt linear heat generation limits to confirm positive margin exists to the limit. The predicted Fq power distribution is increased by the same uncertainties as for the HFP case prior to comparison against the centerline fuel melt linear heat generation limits. If a centerline fuel melt limit is exceeded, then either the HZP SLB transient analysis is evaluated, re-analyzed, or the core redesigned to obtain acceptable peaking factors.

8. Section 5.2.3.2, described in Change 5-6, proposes the use of the WRB-2M and BWU-N critical heat flux (CHF) correlations for HFP SLB analyses and the W-3S and WLOP CHF correlations for HZP SLB analyses.

   8.a What statistical methodology is used for the BWU-N departure from nucleate boiling ratio (DNBR) limit? The WRB-2M correlation is explicitly used with the statistical core design (SCD) methodology, but the limit listed for BWU-N is simply a correlation limit. Considering that both correlations are being used to analyze the same event, why do they not use the same statistical methodology?

   Response:

   BWU-N is coded into the VIPRE-01 computer code to be used automatically with the WRB-2M correlation whenever calculating DNBR below the first mixing vane grid. Since the statistical DNBR limit was calculated using the WRB-2M/BWU-N combination, the statistical limit applies to the combined form of the calculation. Duke will reword the paragraph submitted in the LAR to clarify as follows (underline denotes addition, strikethrough denotes deletion)

   5.2.3.2 Analysis Methodology

   HFP Case

   The VIPRE-01 model from Reference 5-7 is used along with the statistical core design (SCD) methodology (Reference 5-10). With the SCD methodology the uncertainties in many of the initial conditions (e.g., power, pressure, temperature, flow) are included in the statistical DNBR limit rather than in the initial conditions of the analysis. The DNBR correlations used for Westinghouse RFA fuel ~~is~~ are the WRB-2M correlation (Reference 5-11) above the first mixing vane grid and the BWU-N correlation (Reference 5-8) below the first mixing vane grid. The SCD limit for DNB analyses with RFA fuel is [1.30.] ~~The CHF correlation used below the first mixing vane grid is the BWU-N correlation (Reference 5-8) with a correlation limit of 1.21.~~ A transient VIPRE-01 analysis is performed to determine the limiting DNBR case and the limiting statepoint. Using this statepoint the maximum allowable pin radial peaks (MARPs) versus axial peaks for different peak locations are determined. The peaking margin is then computed for each assembly from the MARPs and the SIMULATE-3 predicted core power distribution. Positive peaking margin is then confirmed for each pin.

   8.b The WLOP correlation is being added as an alternate for the W-3S correlation. How does Duke propose to determine which correlation to use? Will one be a primary correlation, with the other used in applications outside the parameter range of the primary?

   Response:

   Duke will use the WLOP CHF correlation for Westinghouse fuel that falls within the WLOP range of applicability. W-3S is retained if the WLOP range of applicability does not apply. Furthermore, since the current analyses use W-3S, retention of W-3S allows for the orderly transition to WLOP upon approval.

8.c The DNBR correlation limit for the WRB-2M correlation is listed in Technical Specifications (TS) Section 2.1.1, "Reactor Core Safety Limits," for both Catawba and McGuire. Please explain why the limits for BWU-N, W-3S, and WLOP are not also required to be listed in the TS.

Response:

Technical Specification 2.1.1 is for steady-state operation, normal operational transients, and anticipated operational occurrences. W-3S and WLOP are only used for SLB, which is outside the conditions used to define the core safety limits. In VIPRE-01, BWU-N is implemented below the first mixing vane grid. However, the minimum DNBR resulting from the core safety limit lines occurs above the first mixing vane grid where the WRB-2M formulation is active. For these reasons BWU-N, W-3S and WLOP should not be included in TS 2.1.1.

9. Section 5.2.3.1 is revised to add a methodology to compute excore flux detector error due to overcooling for the HFP SLB case; Section 6.3.2.3 is revised in the same way for the Dropped Rod analysis. In both sections it is stated that "A conservative attenuation factor is assumed as a function of the change in reactor vessel downcomer density resulting from the change in temperature." How is this attenuation factor chosen or calculated?

Response: Question Withdrawn

10. There are a number of changes to the temperature feedback model for the HZP power case in Section 5.3.2.6, described in Change 5-8. One key change is [

]

10.a Provide clarification on [

]

10.b In what way is [

]

10.c The technical justification for Change 5-8 states: [

]

Response:

A single response is provided to answer questions 10.a through 10.c.

[

]

[

]

Conservatism in the analysis is maintained through the selection of conservative [

]

a. [

]

b. [ ]
c. [ ]
d. [ ]
e. [ ]

[

]

10.d In determining what is conservative, are the [

]

Response:

The SIMULATE-3 model is used as the reference reactivity solution because of the three-dimensional spatial detail of this model relative to the RETRAN-02 point kinetics model. The SIMULATE-3 model has been extensively benchmarked in the methodology report DPC-NE-1005-PA, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 MOX". Conservatism in the analysis is maintained as described in the response to questions 10a through 10c.

10.e In the sample SLB analysis performed using the revised methodology, Table 5-1 states that criticality occurs at 25 seconds, but Figure 5-7 seems to indicate criticality occurring at approximately 20 seconds. Please clarify the apparent discrepancy.

Response:

Figure 5-7 is correct, but the time of criticality is 21.6 seconds. The discrepancy is that Figure 5-7 plotted core reactivity while Table 5-1 keyed off neutron power. The revised Table 5-1 is provided below:

Table 5-1
Sequence of Events for Offsite Power Maintained Case
Zero Power Initial Condition

Event	Time (sec)
Break occurs / Operator manually trips reactor	0.01
Pressurizer level goes offscale low	12
SI actuation on low pressurizer pressure	19
Criticality occurs	21.6
Steam line isolation on low steam line pressure	24
SI pumps begin to deliver unborated water to RCS	38
Peak Heat Flux Occurs	118
High-head SI injection lines purged of unborated water / One train of SI fails	119

11. Change 6-1 revises Section 6.1.3 to remove language specifying a particular fuel performance code, replacing it with language saying that an "appropriate" fuel performance code will be used. How will the appropriateness of the fuel performance code be ensured? What will be done for transition cores where more than one fuel performance code may be applicable?

    Response:

    The revision was made to remove the necessity of modifying the methodology report when changing fuel vendors and/or fuel types requiring the use of an alternate NRC-approved fuel performance code. The methodology used to calculate fuel performance parameters for fuel resident in a Duke reactor is specified in each units' core operating limits report (COLR). This methodology in turn references the fuel performance code used to perform fuel rod design analyses. Only NRC-approved fuel rod design methodologies and fuel performance codes are used to calculate fuel performance limits. The use of the code is restricted to the fuel types for which it was approved. For transition cores, fuel performance limits are calculated using the appropriate vendor's NRC-approved fuel performance code licensed for each fuel type resident in the reactor. Fuel type specific limits are applied to each fuel assembly type when assessing margin to limits. For transition cores, the COLR would include a list of the fuel design methodology applicable to each fuel type resident in the reactor core. For example, if the core had co-resident fuel from AREVA and Westinghouse, the COLR would cite fuel rod design methodology reports describing the use of both TACO3 and PAD.

12. Change 6-2 revises Section 6.3.1 to change the initial condition for the dropped rod analysis to use minimum rather than maximum average fuel temperatures. Please justify why it is conservative to use minimum values.

Response:

There are competing effects during the dropped rod transient since there is both a power decrease following the dropped rod and a subsequent power increase as control rods are withdrawn. Lower fuel temperatures reduce the fuel to moderator ΔT thereby diminishing heat transfer out of the fuel during the transient, which suggests a minimum fuel temperature is non-conservative for DNB. However, this also results in less fuel heatup during the power excursion which results in less Doppler feedback and a greater power excursion. Less Doppler feedback maximizes the power increase (and resultant heat flux increase) which is conservative for DNBR calculations. The net impact is that the calculated DNBR is slightly worse for minimum core average fuel temperature.

13. Change 6-8 revises Section 6.3.2.1 so that weighting factors are now used to account for the relative importance of the fuel assemblies in generating the excore detector response. How are these weighting factors determined?

Response:

Since the original development of the dropped rod accident methodology, methods to more accurately describe the excore detector response have been developed. Current dropped rod accident analyses model the excore detector response [ ] to the excore detector. Weighting factors generated by the excore detector model in SIMULATE-3K are proposed to be implemented. This model computes radial weighting factors for each core location using a "double kernel" method developed by H. Tochihara, E. Ochiai, and T. Hasegawa to approximate the radial neutron transport from each fuel assembly through the core baffle and reactor vessel to the detector. This method is described in Chapter 6 of Reference 4-4, "SIMULATE-3K Models and Methodology, SSP-98/13, Rev. 6, January 2009", and the following reference.

H. Tochihara, E. Ochiai, and T. Hasegawa, "Reevaluation of Spatial Weighting Factors for Ex-Core Neutron Detectors," Nuclear Technology, 58, 310 (1982).

Qualification of the excore detector weighting factors calculated by SIMULATE-3K was performed [

]

14. Change 6-9 revises Section 6.3.2.2 to include the high flux, over-temperature ΔT, and over-power ΔT trips for the dropped rod analysis. What has changed in the methodology that now allows these trips to be credited?

    Response:

    The limiting cases for DNB do not result in a reactor trip. The added trips are valid RPS trip functions and can be modeled. With changes in fuel management strategies, current analyses show these trips may become limiting for future core designs or trip setpoint changes.

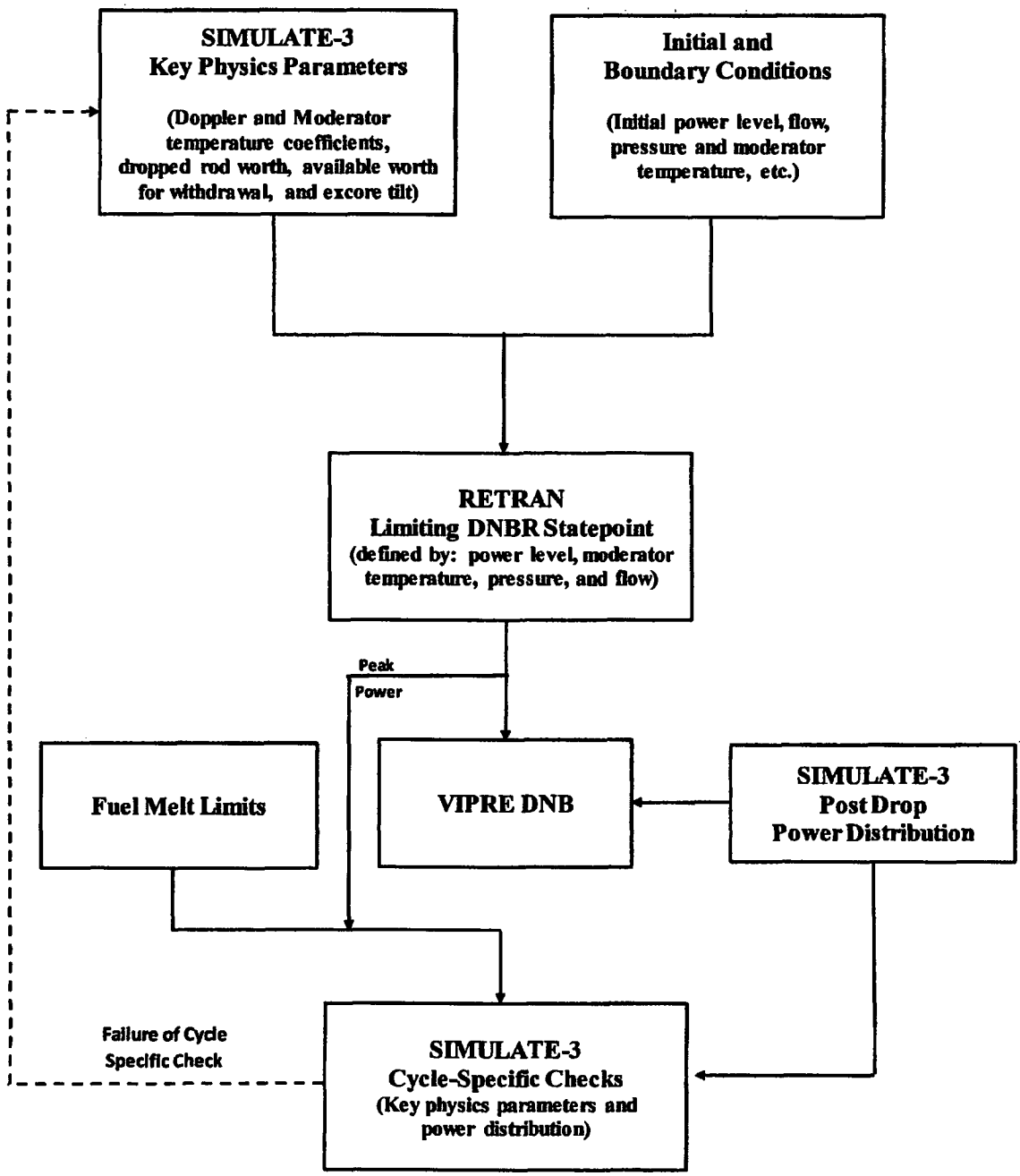
15. In Figure 6-10, which is a flowchart of the dropped rod accident analysis, the box labeled "Fuel Melt Limits" is not connected to any other boxes. How do the fuel melt limits factor in to the analysis?

    Response:

    A revised Figure 6-10 is provided. For each dropped rod combination, the post drop three-dimensional Fq power distribution is calculated and compared against fuel melt limits calculated to by an NRC-approved fuel performance code to confirm margin exists to these limits. The predicted Fq power distribution is increased by uncertainty factors that account for [

    ] If a centerline fuel melt limit is exceeded, either the dropped rod transient analysis is evaluated, re-analyzed or the core is redesigned to obtain acceptable peaking factors.

Revised Figure 6-10
Dropped Rod Accident Analysis Flow Diagram



Attachment C

Affidavit

AFFIDAVIT of Joseph Michael Frisco, Jr.

1. I am Acting Senior 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.

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. 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 in the submittal is that which is marked in the proprietary version of DPC-NE-3001-P, *Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology.* This information enables Duke Energy to:

      (a) Support license amendment requests for its McGuire and Catawba reactors.

      (b) Perform nuclear design calculations on McGuire and Catawba reactor cores containing low enriched uranium fuel.

   (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.

Joseph Michael Frisco, Jr. 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.

Joseph Michael Frisco, Jr.

Subscribed and sworn to me: June 17, 2014
Date

Debra Reese
Notary Public

My commission expires: September 6, 2015

SEAL

DEBRA REESE
NOTARY
PUBLIC
MECKLENBURG COUNTY, NC