DUKE ENERGY.

Kelvin Henderson
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CNS-16-017

March 11, 2016

10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications (TS) Amendments
TS 3.4.1, Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
Response to NRC Request for Additional Information (RAI)
(TAC Nos. MF6355 and MF6356)

References:

1. Letter from Duke Energy to NRC, "Proposed Technical Specifications (TS) Amendments TS 3.4.1, Reactor Coolant System (RCS) Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits", dated June 12, 2015 (ADAMS Accession Number ML15168A009)

2. Letter from NRC to Duke Energy, "Catawba Nuclear Station, Units 1 and 2: Request for Additional Information Regarding License Amendment Request to Change the RCS Minimum Required Flow Rates (TAC Nos. MF6355 and MF6356)", dated January 20, 2016 (ADAMS Accession Number ML16007A190)

The Reference 1 letter requested amendments to the Catawba Units 1 and 2 Facility Operating Licenses (FOLs) NPF-35 and NPF-52, respectively, and the associated TS to modify the subject TS to allow lower minimum values of RCS flowrate. The Reference 2 letter transmitted RAIs associated with this amendment request.

The purpose of this letter is to respond to the Reference 2 RAIs. Attachment 1 to this letter provides the RAI responses. The format of Attachment 1 is to restate each RAI question, followed by its associated response. Attachment 2 provides on compact disc the Duke Energy Topical Reports requested in RAI Question 1. Attachments 1 and 2 are considered proprietary to Duke Energy and/or to Westinghouse Electric Company, LLC (Westinghouse). Attachments 3 and 4 are non-proprietary versions of Attachments 1 and 2, respectively.

ADO1
NRR

As Attachments 1 and 2 contain information proprietary to Duke Energy and/or to Westinghouse, they are supported by affidavits signed by Duke Energy and/or by Westinghouse, the owners of the information. The attached Westinghouse affidavit (Attachment 5) and the Duke Energy affidavits included as part of the requested Topical Reports set forth the basis on which the information may be withheld from public disclosure by the NRC and address with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390. Accordingly, it is requested that the information that is proprietary to Duke Energy and/or to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390. Regarding Attachment 5, correspondence with respect to the copyright or proprietary aspects of this information or the supporting affidavit should reference Westinghouse letter CAW-16-4381 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

The conclusions of the No Significant Hazards Consideration and the Environmental Consideration contained in the Reference 1 letter are unaffected by this RAI response.

There are no regulatory commitments being made in conjunction with this RAI response.

Pursuant to 10 CFR 50.91, a non-proprietary copy of this amendment request supplement is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to L.J. Rudy at (803) 701-3084.

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

Executed on March 11, 2016.

Very truly yours,

Kelvin Henderson
Vice President, Catawba Nuclear Station

LJR/s

Attachments

xc (with Attachments 1, 2, 3, 4, and 5):

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J.A. Whited (addressee only)
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xc (with Attachments 3, 4, and 5 only):

S.E. Jenkins
Manager
Radioactive & Infectious Waste Management
Division of Waste Management
South Carolina Department of Health and Environmental Control
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ATTACHMENT 5

WESTINGHOUSE AFFIDAVIT

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Cranberry Township, Pennsylvania 16066
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CAW-16-4381

March 1, 2016

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Attachment 1 to LTR-LIS-16-78, Proprietary Markup of the Duke-Generated Responses to LOCA-Related RAIs on a Reduction to the MMF for Catawba Units 1 and 2 (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-16-4381 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Duke Energy.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-16-4381, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J.A. Gresham

James A. Gresham, Manager

Regulatory Compliance

CAW-16-4381
March 1, 2016

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission 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 and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Attachment 1 to LTR-LIS-16-78, Proprietary Markup of the Duke-Generated Responses to LOCA-Related RAIs on a Reduction to the MMF for Catawba Units 1 and 2" (Proprietary), for submittal to the Commission, being transmitted by Duke Energy letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the response to the LOCA related RAIs on a Catawba Units 1 and 2 Licensing Amendment Request (LAR) to reduce the reactor coolant system (RCS) minimum measured flow (MMF) , and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to provide input to Duke Energy to support their response to the U.S. Nuclear Regulatory

Commission in response to Request for Additional Information on a Licensing Amendment Request for the Reduction to the MMF for Catawba Units 1 and 2.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of Large Break Loss of Coolant Accident analysis.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC associated with the response to the LOCA related RAIs on a Catawba Units 1 and 2 Licensing Amendment Request (LAR) to reduce the reactor coolant system (RCS) minimum measured flow (MMF), and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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ATTACHMENT 3

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)

(NON-PROPRIETARY)

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO SUPPORT THE

MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

TAC NOS. MF6355 AND MF6356

By letter dated June 12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15168A009), Duke Energy Carolinas, LLC (Duke Energy), the licensee for Catawba Nuclear Station, Units 1 and 2, submitted a license amendment request (LAR) for changes to the Technical Specifications (TSs). Specifically, the licensee proposed to modify TS Table 3.4.1-1, "RCS [Reactor Coolant System] DNB Parameters", Parameter 3, "RCS Total Flow Rate", Limit as follows:

For Unit 1: From "≥ 388,000 gpm and ≥ the limit specified in the COLR (Unit 1)", to "≥ 384,000 gpm and ≥ the limit specified in the COLR (Unit 1)"

For Unit 2: From "≥ 390,000 gpm and ≥ the limit specified in the COLR (Unit 2)", to "≥ 387,000 gpm and ≥ the limit specified in the COLR (Unit 2)"

Based on its review of this LAR, the NRC staff has determined the following additional information is necessary to continue its technical review.

1. Page 3 indicates that the reanalysis to support the TS changes used the methodologies documented in the topical reports (TR) as follows:

   TR-1: DPC-NE-3001-P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology", Revision 0a
   TR-2: DPC-NE-3002-A, "FSAR Chapter 15 System Transient Analysis Methodology", Revision 4b
   TR-3: DPC-NE-2005-P-A, "Thermal Hydraulic Statistical Core Design Methodology", Revision 4a
   TR-4: DPC-NE-3000-P-A, "Thermal-Hydraulic Transient Analysis Methodology", Revision 5a

   Please provide TR-1 through TR-4, identify any differences from TRs originally approved for use in the Catawba licensing applications, and discuss how the Catawba reanalysis satisfies the restrictions and conditions specified in the applicable SERs for TR-1 through TR-4.

Duke Energy Response:

Both proprietary and non-proprietary copies of the methodology reports TR-1 through TR-3 are included with this transmittal. Proprietary and non-proprietary copies of methodology report TR-4 have been previously submitted to the NRC and can be found under ADAMS Accession Number ML16032A004. The following are lists of all restrictions and conditions specified in SERs for all four reports. All conditions and restrictions in the SERs of all reports have been met in the evaluations performed for this proposed change to RCS flow; particularly, the transients which were re-analyzed for the purposes of this LAR met these restrictions and conditions. It should be noted that restrictions associated with Oconee Nuclear Station methodologies, as well as restrictions associated with fuel designs no longer in use within Duke Energy, are not included in these lists.

TR-1 DPC-NE-3001-PA Rev 0a: The only restrictions given on the use of the methodologies approved in this report are the future approvals of reports DPC-NE-1004, DPC-NE-3000, and approval of the RETRAN-02 MOD005 code for boron transport calculations.

TR-2 DPC-NE-3002-A Rev 4b: The following limitations are listed in the SER for Revision 1 of this report.

1. The acceptability of the use of (Duke's) approach to FSAR analysis is subject to the conditions of SEs on all aspects of transient analysis and methodologies (DPC-NE-3000, -3001, -3002, -2004, and -2005) as well the SEs on the RETRAN and VIPRE-01 computer codes.

2. There are scenarios in which an SGTR event may results in loss of subcooling and the consequent two-phase flow conditions in the primary system. In such instances, the use of RETRAN is not acceptable without a detailed review of the analysis.

3. In the future, if hardware or methodology changes are made, selection of limiting transients needs to be reconsidered, and (Duke) is required to perform sensitivity studies to identify the initial conditions in such a way to avoid conflict between transient objective, such as DNB and worst-case primary pressure.

4. It is emphasized that, when using the SCD methodology to determine DNBR, the range of applicability of the selected critical heat flux correlation must not be violated.

5. (Duke's) assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor is not acceptable; (Duke) is required to use 110% of design pressure for that limit.

TR-3 DPC-NE-2005-PA Rev 4a: The following required actions are listed in the SER for Revision 2 of this report for instances whenever a new fuel design is introduced.

1. The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

2. The SCD analysis shall be reviewed and revised as needed if the Mark-B11 CHF correlation range of applicability is changed.

The following required actions are listed in the SER for Revision 0 of this report:

1. The statistical core design (SCD) methodology developed by (Duke), as described in the submittal, is direct and general enough to be widely applicable to any pressurized-water reactor (PWR) fuel or reactor, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the critical heat flux (CHF) correlation subject to the conditions in the VIPRE safety evaluation report (SER). (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. This methodology is approved only for use in (Duke) plants.

2. Of the two DNBR limits, only the use of the single, most conservative DNBR limit is approved.

TR-4 DPC-NE-3000-PA, Revision 5a: The following required actions are listed in the SER for Revision 0 of this report:

RETRAN Findings: (The Commission finds) that (Duke's) RETRAN-02 models (are) acceptable for the simulation of the symmetric non-LOCA thermal-hydraulic transients for the McGuire, and Catawba Nuclear Units, subject to the limitations listed below.

(1) With respect to analyzing transients which result in a reduction in steam generator secondary water inventory, use of the RETRAN-02 steam generator modeling is acceptable, only for transients in that category for which the secondary side inventory for the effective steam generator(s) relied upon for heat removal never decreases below an amount which would cover enough tube height to remove decay heat.

(2) All generic limitations specified in the RETRAN-02 SER.

VIPRE Findings: (The Commission finds) that the subject topical report, together with (Duke) responses, contains sufficient information to satisfy the VIPRE-01 SER requirement that each VIPRE-01 user submit a document describing proposed use, sources of input variables, and selection and justification of correlations as it relates to use by (Duke) for FSAR Chapter 15 analyses regarding Oconee, McGuire and Catawba. We further find that the manner in which the code is to be used for such analyses, selection of nodalization, models, and correlations provides, except as listed below, adequate assurances of conservative results and is therefore acceptable. Furthermore, the use of the (Duke) developed statistical core design methodology as approved in the Staff Safety Evaluation Report on DPC-NE-2004, is approved for the transient application subject to the same conditions.

The following items are limitations regarding VIPRE-01 application presented in DPC-NE-3000 and its supplemental materials:

(1) Determination of acceptability is based upon review of selection of models/correlations for transients involving symmetric core neutronic and thermal-hydraulic conditions only. Thus, the VIPRE-01 models are approved for use in analyzing symmetric transients only;

(2) When using the (Duke) developed SCD method, the licensee must satisfy the conditions set forth in the staff's safety evaluation of DPC-NE-2004;

(3) Whenever (Duke) intends to use other CHF correlations, power distribution, fuel pin conduction model or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, (Duke) must submit its justification for NRC review and approval;

(4) Core bypass flow should be determined on cycle-by-cycle bases;

(5) All generic limitations specified in the VIPRE-01 SER.

As discussed on pages 9 and 15 of the LAR (and in the response to RAI #2 below), core bypass flows for Catawba Units 1 and 2 are calculated using the THRIVE methodology. These calculated core bypass flows, or flows which bound the calculated core bypass flows, are used in all re-analyses and evaluations discussed within the LAR. There are no cycle-specific (i.e., fuel design) changes that would lead to a core bypass flow which is different than those calculated with this methodology.

2. Page 9 and page 15 indicate that the reanalysis used the calculated core bypass flow rates of 6.49 percent and 6.71 percent of the total reactor coolant system (RCS) flow for Unit 1 and Unit 2, respectively. The core bypass flow rates are reduced from 8.5

percent for Unit 1, and 7.5 percent for Unit 2 assumed in the current analysis of record (AOR), resulting in a greater core flow rate for core heat removal.

Please discuss the conservatisms considered in the calculation of the core bypass flow and provide quantitative justification for the calculated bypass core flow of 6.49 percent and 6.71 percent for use in the reanalysis.

Also, please clarify if the current AOR for each of the transients and accident was incorporated in Chapter 15 of the updated final safety analysis report (UFSAR) for Catawba Units 1 and 2.

Duke Energy Response:

All transient analyses using the Statistical Core Design (SCD) methodology, as discussed in the Duke methodology report DPC-NE-2005-PA Revision 4a, use nominal (best estimate) values for key parameters such as core bypass flows. A 1.5% uncertainty in core bypass flow is accounted for as part of the SCD methodology. Therefore, for safety analyses conducted using the SCD methodology, it is appropriate to use the best estimate core bypass flow values calculated by Westinghouse, using the THRIVE methodology.

An overview of the THRIVE methodology used to calculate the core bypass flows is given below.

The THRIVE code predicts the reactor vessel pressure losses by classical analytical fluid mechanics. THRIVE solves the following continuity and momentum equations for a flow system that represents the entire reactor vessel and internals system:

$$W = \rho VA = constant$$

$$P_j = P_i + \sum_i^j (K + fl/D)\frac{\rho V^2}{2g_c}$$

Typically, in purely analytical hydraulic analyses, the fluid properties appearing in the equations are known to a high degree of confidence. Therefore, if the pressure loss determination fails, it is generally due to the inability to analytically predict hydraulic loss coefficients of complex geometries. In order to eliminate this potential problem in the THRIVE code, these coefficients were experimentally determined from tests on the 1/7 scale models of the San Onofre, Connecticut Yankee and 3XL pressurized water reactors. Thus, the pressure drops predicted by the code are the results of standard hydraulic methods utilizing coefficients with a sound experimental basis.

The best estimate core bypass flow calculation that was performed for Catawba Units 1 and 2 used fuel properties for Westinghouse 17X17 Standard RFA fuel with Quick Release Top Nozzle (QRTN) and Jedinstvo debris filter bottom nozzles. These evaluations were performed using the best estimate fuel assembly thermal hydraulic properties with 1400 WABA's and 0 WABA's to bound the best estimate fuel assembly thimble tube bypass condition along with the best estimate reactor internal hydraulic loss coefficients. The calculated best estimate core bypass flow at a total thermal design flow of 390,000 gpm (97,500 gpm / loop) is 6.49% for Catawba Unit 1 and 6.71% for Catawba Unit 2. These core bypass flows are best estimate values and contain no uncertainties.

Separate calculations were performed to incorporate uncertainties. The following is a list of the parameters that were adjusted to incorporate uncertainty, along with the applied uncertainty percentage:

Fuel Parameters

- *Constant multiplier on rod bundle friction. (± 10%)*
- *Core inlet loss coefficient. (± 35%)*
- *Non-mixing vane grid loss coefficient. (± 25%)*
- *Mixing vane grid loss coefficient. (+ 15% for mixing vane and +25% for IFM)*
- *Core outlet loss coefficient. (± 30%)*
- *Form loss coefficient for each axial step in core.*
    - *Bottom Nozzle (± 40%)*
    - *Protective Bottom Grid (± 25%)*
    - *Non-mixing vane grid (± 25%)*
    - *Mixing vane grid (± 15%)*
    - *IFM (± 25%)*
    - *Top Nozzle (± 50%)*
- *Coefficient of Reynold's number (Re) in core loss coefficient equation: K = COEFKC*Re**PWRKC (± 10%)*
- *Lower core plate loss coefficient. (± 50%)*
- *Upper core plate loss coefficient. (± 35%)*

Baffle-Former Region

- *Loss coefficient for Type 1 former holes. (- 10%)*
- *Loss coefficient for Type 2 former holes. (- 10%)*
- *Loss coefficient at baffle outlet (for upflow). (- 10%)*
- *Loss coefficient at bottom of baffle-barrel region. (- 10%)*

Head Cooling Flow

- *Head cooling flow nozzle loss coefficient. (± 10%)*
- *Guide tube downward loss coefficient. (- 10%)*

- *Hydraulic loss coefficient of support columns for downward flow. (- 10%)*

Outlet Nozzle Gap

- *Measured cold outlet nozzle gap. (+ 10%) but less than the maximum measured gap*

Core Cavity Gap

- *Core cavity gap. (+ 10%)*

Thimble Tube

- *Thimble Tube bypass flow. (± 10%)*

When these uncertainties are taken into account, the calculated core bypass flows for Catawba Units 1 and 2 are 7.300% and 7.483% respectively. These core bypass flow values with uncertainties incorporated are used in all safety analyses which do not use SCD methodology, such as the Rod Ejection accident (UFSAR Section 15.4.8).

There will be no UFSAR updates for Catawba Units 1 and 2 Chapter 15 accident analyses incorporating the proposed RCS flow reductions until the LAR is approved. The current AORs do not use the revised core bypass flows for Catawba Unit 1 or 2. All current Catawba Units 1 and 2 UFSAR accident analyses reflect the existing RCS flow TS requirements.

3. Pages 9 to 12 and pages 15 to 17 discuss the reanalysis and evaluation for Category-3 events to support the proposed TS changes reducing the RCS flow rates for Unit 1 and Unit 2, respectively.

Please clarify if the only change to the input parameter in the reanalysis compared with the AOR is the total RCS flow rates. Please justify any other changes in the input parameters, models, and methodologies used in the reanalysis.

Duke Energy Response:

Unit 1:

For Category 3 transients designated 3A through 3J, evaluations are performed using the total RCS flow and core bypass flow values discussed on page 9. There were no other input changes made or implied for these evaluations.

Several Category 3 transients were re-analyzed. These are designated as Cases 3L, 3M, 3O, 3P (peak RCS pressure only) and 3Q. The total RCS flow was updated to reflect the proposed total RCS flow of 384,000 gpm. The core bypass flow was

assumed to be 9.2%, which conservatively bounds the core bypass flow for all Catawba and McGuire units (excepting Case 3M, where a core bypass flow of 8.5% was assumed). There were no other changes to input parameters, models, or methodologies for these revised analyses.

For all other Category 3 transients, evaluations were done using engineering judgment as well as the results of previously documented analyses. There were no new analyses performed for these evaluations.

Unit 2:

For Category 3 transients designated 6A through 6K, evaluations are performed using the total RCS flow and core bypass flow values discussed on page 15. There were no other input changes made or implied for these evaluations.

Several Category 3 transients were re-analyzed. These are designated as Cases 6L, 6M, and 6O. The total RCS flow was updated to reflect the proposed total RCS flow of 387,000 gpm. The core bypass flow was assumed to be 7.5%, which conservatively bounds the core bypass flow for Catawba Unit 2. There were no other changes to input parameters, models, or methodologies for these revised analyses.

For all other Category 3 transients, evaluations were done using engineering judgment as well as the results of previously documented analyses. There were no new analyses performed for these evaluations.

4. Category-1 Events for Unit 1: Transients Bounded by Current RCS Flow Assumption

Page 7 of Attachment 1 to the LAR lists six Category-1 events. The LAR indicates that for Category-1 events, the total RCS flow assumed in the current AOR is based on either the mechanical design flow of 420,000 gpm (where maximum RCS flow rates are conservative) or the thermal design flow of 382,000 gpm (where minimum RCS flow rates are conservative). The LAR further states that the proposed minimum RCS total flow limit of 384,000 gpm has no impact on the analysis for this category events.

The above quoted flow rates of 420,000 gpm and 382,000 gpm appear to be inconsistent with the values in FSAR Table 15.4 for two Category-1 events (Event 1D and Event 1E). The flow rates are 381,420 gpm for Event 1D, the turbine trip analysis addressing the peak RCS pressure (UFSAR 15.2.3) and 373,596 gpm for Event 1E, the analysis for the loss of non-emergency AC power to the station auxiliary (UFSAR 15.2.6), respectively. Please clarify the inconsistences discussed above and justify that the results of AOR for the above two events remain valid with the proposed lower RCS flow limit.

Also, UFSAR 15.8 indicates that Event 1F (for Unit 1), Anticipated Transients without Trip, is not analyzed. Please explain why this event is classified as a Category-1 event that is defined by the licensee as an event that is bounded by AORs using current RCS flow assumptions. Additionally, please address this question with respect to Event 4I for Unit 2.

Duke Energy Response:

The Catawba UFSAR states in Section 15.2.3 that Unit 2 results are presented, due to similar results between the two units. The Catawba Unit 2 analysis of record assumed 390,000 gpm RCS flow with 2.2% flow measurement uncertainty. This total RCS flow of 381,420 gpm is shown in Table 15.4.

However, a separate Turbine Trip analysis for Catawba Unit 1 is maintained. (Again, the results of the Unit 1 analysis are not discussed within the UFSAR.) In this AOR for Catawba Unit 1, an RCS flow of 382,000 gpm is assumed. Therefore, the evaluation presented in the LAR for Catawba Unit 1 correctly asserts that the AOR is not impacted by the proposed reduction in RCS flow to 384,000 gpm, since the Catawba Unit 1 AOR already assumes an RCS flow which bounds this value. The resulting peak primary system pressure for Catawba Unit 1 is 2649.3 psig. This is well below the peak primary pressure limit of 2733.5 psig.

The RCS flow given in Table 15.4 for Event 1E, the Loss of Non-Emergency AC Power to the Station Auxiliaries (Section 15.2.6), is 373,596 gpm. This represents the nominal RCS flow assumed in the AOR of 382,000 gpm, reduced by 2.2% to incorporate the flow measurement uncertainty. Therefore, the results of the AOR for this event as well as the Turbine Trip accident (peak primary pressure) remain valid with the proposed lower RCS flow limit.

The flows for the ATWS event (Section 15.8) cited in Tables 1 and 2 of the LAR reflect the generic Westinghouse analysis (which is cited as Reference 1 in Section 15.8.1 of the Catawba UFSAR). In this analysis, an RCS flow of 377,600 gpm was assumed. The UFSAR statement that this analysis "is not analyzed" reflects that no Duke-specific analysis is performed or presented.

5. Events 2G and 3N (Unit 1) and Event 5G (Unit 2) - Feedwater System Pipe Break (FLB, UFSAR 15.2.8, Short-Term)

For Events 2G and 3N (Unit 1) and Event 5G (Unit 2), the short-term FLB analysis is not reanalyzed at the proposed RCS flow rate, because the LAR indicates that the short-term-cooling results for the FLB analysis are bounded by the analysis for a loss of normal feedwater flow (LONF, UFSAR 15.2.7).

~~The above statement is apparently inconsistent with the statement discussed in the last paragraph of UFSAR 15.2.8.2, which states that the short-term-cooling (DNBR) results for the FLB incident "is bounded by that of the complete loss of forced reactor coolant~~

~~incident, thus, ensuring that the integrity of the core is maintained in the short term."~~[2] Indicate where, in the UFSAR, is it stated that a FLB is not limiting.

Additionally, please clarify the inconsistencies identified above regarding the minimum departure from nucleate boiling ratio (DNBR) results and justify that the short-term-cooling results for the FLB analysis meet the applicable acceptance criteria and are acceptable for cases with lower minimum RCS flow limits of 384,000 gpm and 387,000 gpm for Units 1 and 2, respectively.

Duke Energy Response:

The UFSAR does not state that the short-term FLB analysis is bounded by the analysis for a loss of normal feedwater flow (LONF). Rather than comparison with the analysis described in 15.2.7 for the LONF accident, the evaluation of the short-term FLB analysis should be performed as follows.

As given in UFSAR Table 15.4, the RCS flows for these analyses are 388,000 and 390,000 gpm respectively for Catawba Units 1 and 2. These analyses also assumed core bypass flows of 8.5% and 8% respectively for Catawba Units 1 and 2. Therefore, using the same evaluation approach as described on pages 4 and 11 of the LAR, it may be shown that the flow through the core in the current analysis of record is actually lower than that with the proposed reduction in RCS flow, once the lower unit-specific core bypass flow of 6.49% and 6.71% are also considered. Therefore, the proposed reduction in RCS flows are acceptable for the short-term FLB analysis.

Therefore, Events 2G and 5G in the LAR should instead be categorized as transients which are evaluated (Category 3), and placed in the same sub-category as events 3A to 3J for Unit 1, and events 6A to 6K for Unit 2.

It is concluded that that current AOR for the FWLB accident (short-term cooling case) will not be impacted by the proposed RCS flow rate reduction.

6. Event 3K – Excessive Increase in Secondary Steam Flow (UFSAR 15.1.3)

Page 10 indicates that the sensitivity analysis for Event 3K in UFSAR 15.1.3 showed that the results of the transient analysis are not sensitive to the RCS flow rate.

UFSAR Table 15.2 indicates that for the increase in steam flow event, UFSAR Section 15.1.3 considered two cases: (1) manual rod control with the most negative moderator coefficient; and (2) automatic rod control with the most negative moderator coefficient. Clarify if both cases were considered in the RCS flow rate sensitivity analysis.

Duke Energy Response:

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[1] The lined out portion of the question was included in the original Draft RAI. During a teleconference with the licensee, Duke staff indicated that the cited statement was not in the current UFSAR. Upon further review, the NRC staff identified that statement was removed in a previous revision to the UFSAR. Therefore, the NRC staff has modified the question accordingly.

The limiting case for the Increase in Steam Flow transient (UFSAR 15.1.3) is the case with manual rod control with the most negative moderator coefficient, or Case 1. This was the limiting case for previous analyses which were run at RCS flows of 382,000 gpm, and was the only one run later at 388,000 gpm.

7. Event 3L (Unit 1) and Event 6L (Unit 2) – Steam System Piping Failure (UFSAR 15.1.5)

(a) Page 10 and page 16 indicate that event 3L for Unit 1 and Event 6L for Unit 2 were reanalyzed at a RCS flow rate of 384,000 gpm and 387,000 gpm, respectively.

UFSAR Table 15.2 indicates that for the steam line break (SLB) event, UFSAR 15.1.5 discussed the results of analysis for two cases: (1) an SLB event with offsite power maintained; and (2) an SLB event with offsite power lost. Please clarify if both cases were considered in the reanalysis for both Unit 1 and Unit 2.

Duke Energy Response:

The Offsite Power Maintained case was the limiting case for the SLB event, and was the only one re-analyzed at an RCS flow of 384,000 gpm for Catawba Unit 1 and 387,000 gpm for Unit 2.

(b) The reanalysis of Event 3L and 6L indicate that DNB does not occur during the SLB event, since the calculated minimum DNBR is above the W-3S Critical Heat Flux (CHF) correlation limit of 1.45. Please indicate whether the use of W-3S correlation and associated DNBR limit of 1.45 in the SLB analysis for Catawba licensing applications has been previously reviewed and approved by the NRC.

Duke Energy Response:

The W-3S correlation was reviewed by the NRC and approved as stated in the SER for DPC-NE-3001-PA, Rev. 0.

8. Event 3M – Loss of Normal Feedwater (LONF) Analysis (UFSAR 15.2.7)

For the long-term LONF analysis discussed for Event 3M, the AOR assumes an RCS flow of 388,000 gpm. Because a subcooling of at least 40 ^{0}F at the RCS hot-leg exists throughout the transient, the licensee states that "the proposed reduction in RCS flow rate to 384,000 gpm will have an inconsequential impact on this transient."

The quoted flow rate of 388,000 gpm appears to be inconsistent with the UFSAR Chapter 15 information: UFSAR Table 15.2 indicates that for Unit 1, the LONF event (UFSAR 15.2.7) is analyzed for the long-term cooling capability as Case 1. As indicated in UFSAR Table 15.4, Case 1 of the LONF event is analyzed with a total RCS flow of 381,420 gpm.

Please clarify the apparent inconsistency for the RCS flow rates discussed in the information for Event 3M and UFSAR Table 15.4, and justify that the AOR Case 1 in UFSAR 15.2.7 for the long-term LONF analysis remains valid with a lower minimum RCS flow limit for Unit 1.

Duke Energy Response:

The RCS flow rate for the LONF long-term cooling transient AOR was misquoted in the LAR as 388,000 gpm. The AOR for Catawba Unit 1 actually assumes an RCS flow rate of 390,000 gpm. This is reflected in UFSAR Table 15.4, which gives an RCS flow for the LONF long-term transient as 381,420 gpm. This Table 15.4 value represents the nominal 390,000 gpm minus a flow measurement uncertainty of 2.2%.

The evaluation within the LAR for the lower minimum RCS flow limit remains valid. The proposed reduction in flow to 384,000 gpm for Unit 1 will have no impact on the LONF transient analyses. A significant amount of subcooling (≥ 40°F) will be maintained as demonstrated by several sensitivity analyses.

9. Event 3O (Unit 1) and Event 6M (Unit 2) – Reactor Coolant Pump Shaft Seizure – Locked Rotor (UFSAR 15.3.3)

Page 11 and page 16 indicate that Event 3O for Unit 1 and Event 6M for Unit 2 were reanalyzed at a RCS flow rate of 384,00 gpm and 387,00 gpm, respectively.

UFSAR Table 15.2 indicates that for the locked rotor event (Events 3O and 6M), UFSAR 15.3.3 considered three cases: (1) the worst peak pressure case; (2) the core cooling analysis with offsite power maintained; and (3) the core cooling analysis with offsite power lost. Please clarify if all three cases were considered in the updating analysis. Also please provide justification for the cases that were not considered in the reanalysis for both Unit 1 and Unit 2.

Duke Energy Response:

Events 3O for Unit 1 and 6M for Unit 2 are specifically discussing Reactor Coolant Pump Shaft Seizure (Locked Rotor) Case 1 (peak pressure). Events 3D and 6E are specifically discussing Locked Rotor - DNB cases (core cooling). Previous analyses have demonstrated that the case with offsite power lost is limiting for DNB (core cooling) analyses, and is the only case analyzed in the current AOR. The evaluations for Events 3D and 6E pertain to Locked Rotor Case 3.

10. Event 3P – Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (BWFS or BWALP, UFSAR 15.4.1)

The information for Event 3P indicates that for BWFS or BWALP (UFSAR 15.4.1), Case 1 addressing adequacy of the core cooling assumes RCS flow with three reactor coolant pumps operational based on nominal flow of 388,000 gpm. The quoted flow rate of 388,000 gpm is not consistent with the value included in FSAR Table 15.4 which indicates in the column designated as UFSAR Section 15.4.1 that the RCS flow rate used is 299,613 gpm for the Case 1 of the BWFS or BWALP analysis. Please clarify the inconsistency discussed above and justify that Case 1 of the AOR in UFSAR 15.4.1 remains valid with the required minimum RCS flow reduced from 388,000 gpm to 384,000 gpm.

Duke Energy Response:

The RCS flow given in Table 15.4 for Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR 15.4.1), Case 1 is 299,613 gpm. This represents the nominal RCS flow of 388,000 gpm with two adjustments.

The first adjustment is a 1% reduction which incorporates the Catawba TS 3.4.1 allowance to operate to RCS flows between 99% and 100% of the limit specified in the Core Operating Limits Report (COLR), as long as Reactor Thermal Power is reduced to ≤ 98% of the Rated Thermal Power Level. This reduction, which is referred to as the "stair-step" flow reduction, is factored into this transient analysis as discussed on page 198 of the LAR.

The second adjustment is a factor to adjust RCS flow at 4-pump operation to operation with 3 RCPs. This 78% factor is taken from RCP flow coastdown testing. These two adjustments reduce the total RCS flow from the nominal 388,000 gpm value to 299,613 gpm, as reported in CNS UFSAR Table 15.4.

The calculated DNBR of 2.973 in the current AOR (with a 4-RCP nominal RCS flow assumption of 388,000 gpm) shows that there is extensive margin between the analysis results and the acceptance criteria of 1.45. It can therefore be concluded that the proposed reduction in total RCS flow has no impact on this transient. Case 1 of the AOR remains valid with regard to the LAR evaluation for Event 3P.

11. Event 3Q (Unit 1) and 6O (Unit 2) – Uncontrolled RCCA Bank Withdrawal at Power (RWDAP, UFSAR 15.4.2)

Page 12 and page 17 indicate that Event 3Q for Unit 1 and Event 6O for Unit 2 were reanalyzed at a RCS flow rate of 384,000 gpm and 387,000 gpm, respectively.

UFSAR Table 15.2 indicates that for the RWDAP event, UFSAR 15.4.2 considered five cases:

(1) Bank withdrawal from 10 percent power core cooling;
(2) Bank withdrawal from 8 percent power peak RCS pressure;
(3) Bank withdrawal from 50 percent power core cooling;
(4) Bank withdrawal from 98 percent power core cooling; and
(5) Bank withdrawal from 100 percent power core cooling.

Please clarify if all five cases were considered in the reanalysis. Provide justification for the cases that were not considered in the updating analysis for Unit 1 and Unit 2.

Duke Energy Response:

For the Uncontrolled RCCA Bank Withdrawal at Power event, all five cases were considered for both units. Case 2 was re-analyzed at the proposed RCS flow of 384,000 gpm (Event 3Q) for Unit 1, and for 387,000 gpm (Event 6O) for Unit 2.

Cases 1, 3, 4, and 5 are evaluated at the proposed RCS flows for Units 1 and 2 respectively (Events 3E and 6F).

12. Events 3R (Unit 1) and 6P (Unit 2) - Startup of an Inactive RCP at an Incorrect Temperature (UFSAR 15.4.4)

    The information for Event 3R for Unit 1 and Event 6P for Unit 2 indicates that the current AOR is based on the calculated three-pump flow, starting from a nominal power and four pump flow of 388,000 gpm. The above quoted flow rate of 388,000 gpm is not consistent with the value of 272,747 gpm included in in FSAR Table 15.4 for UFSAR 15.4.4 case, the event of the startup of an inactive RCP at an incorrect temperature. Please clarify the inconsistency for the flow rates discussed above and justify that the AOR in UFSAR 15.4.4 remains valid with a lower minimum RCS flow limit of 384,000 gpm and 387,000 gpm for Units 1 and 2, respectively.

    Duke Energy Response:

    An initial RCS 4-pump flow of 375,669 gpm was assumed. This represents a nominal RCS flow of 388,000 gpm, reduced by the 1% stair-step factor (see Question #10) and by a 2.2% flow measurement uncertainty. When an RCP is tripped in the RETRAN analysis from this initial RCS 4-pump flow, a 3-pump flow of 272,747 gpm (net) is established in the AOR. It is this 3-pump RCS flow that is reported in Catawba UFSAR Table 15.4.

    As discussed in the evaluations for Events 3R and 6P, a previous analysis with different fuel types and with a nominal full power flow of 382,000 gpm gave similar results to the current AOR. Both cases showed similar core responses and considerable margin to the limiting DNBR in both cases. DNB is not a concern for either the current AOR or for the previous analysis (both applicable for Catawba Units 1 and 2). Therefore, it is concluded that the LAR evaluations for events 3R and 6P remain valid.

13. Events 3S (Unit 1) and 6Q (Unit 2) – Steam Generator Tube Rupture (SGTR, UFSAR 15.6.3)

    (a) Overfill analysis

    For the SGTR analysis discussed in the information for Event 3S (Unit 1), and 6Q (Unit 2), the licensee indicates that "the overfill analysis determined the assumed RCS flow to be inconsequential." Please provide analyses or other information to justify that the AOR overfill analysis is insensitive to the RCS flow and remains valid with the proposed RCS flow limit of 384,000 gpm, and 387,000 gpm for Units 1 and 2, respectively.

Duke Energy Response:

To determine the margin to SG overfill during a SGTR event for Catawba Units 1 and 2, separate analyses were performed. The methodology used to perform these analyses is based on WCAP-10698-P-A, "SGTR Analysis Methodology To Determine the Margin to Steam Generator Overfill". A Safety Evaluation Report was issued by the NRC staff on March 30, 1987 accepting the WCAP-10698 methodology as being conservative with respect to prediction of margin to overfill as a result of SGTR for Westinghouse plants. Separate analyses (with different SG models and operator action times) were performed as allowed by the Safety Evaluation Reports issued for the Catawba SG PORVs (TAC Nos. M98107 and M98108).

In the WCAP-10698 methodology, neither initial RCS flow rates nor core bypass flows were identified as being important parameters with regard to the analysis results. Since reactor trip is assumed to occur at 3 minutes following the SGTR (Catawba Unit 1) or at about 6 minutes following the SGTR (Catawba Unit 2), and since the SG mass in the AOR for both units reaches maximum values at times greater than 1 hour following the SGTR, the conclusion that initial RCS flows and core bypass flows assumed for the SGTR overfill analyses are inconsequential is justified.

(b) Dose Input Analysis

The information discussing Events 3S and 6Q indicates that "the dose input analysis was performed at 390,000 gpm plus uncertainty." The above quoted RCS flow rate is inconsistent with that for Case 1 shown in UFSAR Table 15.4, the dose input analysis, which is based on the flow rate of 373,599 gpm. Please clarify the above inconsistency for the flow rate and justify that the AOR dose input analysis in UFSAR 15.6.3 remains valid for lower required minimum RCS flow limits of 384,000 gpm, and 387,000 gpm for Units 1 and 2, respectively.

Duke Energy Response:

The RCS flow rates for the Steam Generator Tube Rupture AORs for Catawba Units 1 and 2 (performed for dose input) were misquoted in the LAR as 390,000 gpm. The AOR for Catawba Units 1 and 2 (Events 3S and 6Q) assume an RCS flow rate of 382,000 gpm; this is reflected in UFSAR Table 15.4, which gives an RCS flow for the LONF long-term transient as 373,599 gpm. This value is the nominal 382,000 gpm minus a flow measurement uncertainty of 2.2%.

The evaluation within the LAR for the lower minimum RCS flow limit remains valid. The proposed reduction in flow to 384,000 gpm for Unit 1 and to 387,000 gpm for Unit 2 will have no impact on the SGTR analyses

performed for dose input. Separate sensitivity analyses conducted for Catawba Unit 1 at initial RCS flows of 390,000 gpm and 388,000 gpm have confirmed that the initial RCS flows and core bypass flows have an insignificant impact on the results of the analyses.

14. Events 3T (Unit 1) and 6R (Unit 2) – Loss-of-Coolant Accident (LOCA)

(a) Large-Break LOCA (LBLOCA)

For the LBLOCA analysis discussed in the information for Event 3T (Unit 1) and Event 6R (Unit 2), the LAR indicates that a Westinghouse analysis determines that "the variations in the global model calculations are such that the 95th percentile peak clad temperature is not impacted." Please provide a discussion of the "Westinghouse analysis" used to determine the effects of the RCS flow changes on the peak clad temperature (PCT) during LBLOCA conditions and address the acceptability of the "Westinghouse analysis" for supporting the TS changes in reducing the minimum required RCS flow limit. The requested information should include the methods and RCS flow rates used for the analysis, and the results of the analysis to support the licensee's position stating that the LBLOCA PCT is not affected by the variations of the RCS flow rate, which is 390,000 gpm (indicated in Note 16 of FSAR Table 15.4 for Case 1, UFSAR Section 15.6.5) used in the AOR for LBLOCA.

Duke Energy Response:

An RCS flowrate of 390,000 gpm was assumed for the Catawba/McGuire Large Break LOCA analysis of record. To support the requested change (reduction) in RCS Minimum Measured Flow (MMF), Westinghouse evaluated a corresponding reduction in Thermal Design Flow (TDF) to determine the impact to the LBLOCA analysis. Thermal Design Flow is defined by subtracting the plant flow measurement uncertainty from the MMF. A 2.2% flow measurement uncertainty is applicable to Catawba Units 1 and 2.

Table 1: Reduced RCS Flow Rates

Unit	RCS Flow Rate Used in LBLOCA Analysis of Record	Reduced MMF Requested in LAR	Reduced TDF Used in LBLOCA Evaluation
Catawba Unit 1	390,000 gpm	384,000 gpm	375,552 gpm
Catawba Unit 2	390,000 gpm	387,000 gpm	378,486 gpm

Thus, from Table 1, the largest difference between the assumed LBLOCA AOR and the new values occurs for Catawba Unit 1, where the difference is 14,448 gpm, 3.7% less than the AOR assumed flow. Because resistance is proportional to the flowrate squared, this represents an effective increase in total loop resistance of approximately 7.8% compared with the AOR.

Since the McGuire and Catawba units are analyzed by a single analysis of the most limiting plant configuration, the 7.8% resistance increase was evaluated.

The effect of uncertainty in break path resistance is accounted for in the LBLOCA AOR through a resistance ratio that considers the relative resistances from the core to the vessel- and pump- sides of the break:

[

]^{a,c}

Uncertainty in the break path resistances is accounted for by using a single uncertainty parameter. As described in Section 25-3 of Reference 1, the parameter varied is [

]^{a,c}

In an LBLOCA analysis, the steady-state loop flow rate is [

]^{a,c} The primary physical influence of these parameters is to affect the behavior in the core during blowdown; changes in effective break size and relative break path resistances influence the location of flow stagnation in the core, which can create more or less limiting conditions for blowdown cooling. Later,

during reflood, the broken cold leg nozzle loss can affect water retention in the downcomer, which influences cladding heatup, but this is related specifically to the broken cold leg nozzle loss coefficient and not strongly influenced by loop piping resistance or the resistance ratio. The Catawba and McGuire plants are late-reflood limited, so it is expected that the resistance ratio has an indirect influence on PCT by virtue of its influence during blowdown.

Regarding blowdown, a larger range of core stagnation conditions is analyzed by [

]^{a,c}

Reference 1 WCAP-12945-P-A, Vol. 1, Rev. 2, and Vol. 2 through 5, Rev. 1, "Code Qualification Document for Large Break BELOCA," 1998.

(b) Small-Break LOCA (SBLOCA)

For the SBLCA analysis discussed in Events 3T and 6R, the licensee indicated that a RCS flow change would not significantly affect four significant factors dominating the SBLOCA analysis. The factors discussed were decay heat, RCS mass, break flow, and ECCS delivery.

Please provide a discussion or analysis to justify that an RCS flow change would not significantly affect the four factors dominating the SBLOCA analysis. As part of this discussion, address the consideration that the RCS flow rate will determine the values of the heat transfer coefficients and thus, the heat removal rate from the RCS primary to secondary side. The heat removal rate would determine the changes of the temperature, pressure, and void fraction of the RCS and steam generator, which, in terms, could affect the RCS water level, RCS mass, break flow, and ECCS delivery, resulting in changes in the results of the SBLOCA analysis.

Duke Energy Response:

Minimum Measured Flow is not directly used as an input to the Catawba Small Break LOCA analyses. The NOTRUMP-EM SBLOCA model is initialized using the Thermal Design Flow (TDF) to establish the plant specific steady state RCS thermodynamic conditions associated with normal full power operation.

A reduced initial RCS flow rate may impact the heat balance between the primary system and the steam generator (SG) secondary leading to a small

change in SG secondary mass and operating pressure. These small changes in SG initial conditions could result in a small shift in the time at which the SG safety valves lift in the NOTRUMP calculations, potentially having a second order effect on the timing of loop seal clearing. Considering the other factors impacting the timing of loop seal clearing (break size, core exit vapor flow, SG condensation rates, etc.) and the duration of a typical SBLOCA transient, the potential timing shift due to the reduced TDF is considered insignificant to the outcome of the SBLOCA transient.

In a design basis SBLOCA licensing calculation using the NOTRUMP-EM, it is assumed that a loss of offsite power (LOOP) occurs coincident with reactor trip. The LOOP results in a reactor coolant pump (RCP) trip. With the RCPs tripped, the RCS flow is driven by pump coast down, natural circulation, two phase flow, and ultimately single phase vapor flow; all of which are independent of initial RCS flow.

Furthermore, the LOOP and coincident reactor trip occur at approximately 57 seconds into the limiting SBLOCA transient, which has a PCT value of 1323°F occurring at 3449 seconds. Considering the above discussion, a reduction in the initial RCS flow would not have any significant impact on the nature of the SBLOCA transient progression and subsequent PCT prediction.

Based on the previous discussion, the reduction in MMF (and TDF) given in Table 1 for Catawba Units 1 and 2 is concluded to have no significant impact on the outcome of the SBLOCA transients documented in the Catawba AORs, and the results and conclusions of the Catawba SBLOCA AORs remain applicable.

15. Category-1 Events for Unit 2: Transients Bounded by Current RCS Flow Assumption

Page 13 of Attachment 1 to the LAR lists nine Category-1 events for Unit 2. The licensee indicates that for the Category-1 events, the total RCS flow assumed in the current AOR is based on either the mechanical design flow of 420,000 gpm (where maximum RCS low rates are conservative) or the thermal design flow of 382,000 gpm (where minimum RCS flow rates are conservative). The licensee further states that the proposed minimum RCS total flow limit of 387,000 gpm has no impact on the analysis for this category events.

The above quoted flow rates of 420,000 gpm and 382,000 gpm appear to be inconsistent with the values in UFSAR Table 15.4 for the following Category-1 events:

Event	Flow Rate (gpm)	
• 4D (UFSAR 15.1.3)	388,000	
• 4E (UFSAR 15.2.3)	381,420	(Peak RCS Pressure case)
• 4F (UFSAR 15.2.6)	373,596	

- 4G (UFSAR 15.2.7) 376,530 (Long-Term Core Cooling case)
- 4H (UFSAR 15.2.8) 373,596 (Long-Term Core Cooling case)

Please clarify the inconsistences discussed above and justify that the results of AOR for each of the above five events remained valid with the total minimum RCS flow limit reduced from 390,000 gpm to 387,000 gpm.

Duke Energy Response:

4D: UFSAR Section 15.1.3, pg. 15.1-5 states that only Catawba Unit 1 results are presented for the Excessive Increase in Steam Flow accident. The RCS flow given in UFSAR Table 15.4 reflects this. A separate analysis for Catawba Unit 2 is maintained (with 385,000 gpm RCS flow assumed) but the results of this analysis, which represents the AOR for Catawba Unit 2, are not presented in the UFSAR due to the overall similarities with the Unit 1 results.

The LAR includes Event 4D with several other accident analyses (4A through 4I) which are categorized as having initial RCS flows of either 420,000 gpm (conservatively high) or 382,000 gpm (conservatively low), and therefore are not impacted by the proposed RCS flow reduction. The initial RCS flow in the current AOR for the Catawba Unit 2 Excessive Increase in Steam Flow accident (4D) is 385,000 gpm. While this flow is not equal to the 382,000 gpm value given as the bounding low value for accident analyses in this category, it is still bounding with respect to the proposed Catawba Unit 2 RCS flow of 387,000 gpm. Similar to the results for Catawba Unit 1, the Unit 2 analysis for this accident has an approximate 6% margin to the OP Delta T trip throughout the event. No fuel pins experience DNB, and thus no fuel failure occurs during the accident. The proposed minimum flow rate of 387,000 gpm is determined to be acceptable.

4E: Similar to RAI Question #4 above, this question pertains to the Turbine Trip accident. Catawba Unit 2 results are presented in the UFSAR per page 15.2-4. The following response is similar to that for Question #4 above.

The Catawba UFSAR Section 15.2.3 states that Unit 2 results are presented, due to similar results between the two units. The Unit 2 analysis of record assumed 390,000 gpm RCS flow with 2.2% flow measurement uncertainty. This total RCS flow of 381,420 gpm is shown in Table 15.4.

The calculated peak pressure for this analysis is 2661.8 psig, which is 72 psi below the peak primary pressure limit of 2733.5 psig. It is not expected that a reduction in RCS flow to 387,000 gpm for Unit 2 would challenge this limit due to the extensive amount of margin in the current AOR.

4F: The RCS flow given in UFSAR Table 15.4 for the Loss of Non-Emergency AC Power to the Station Auxiliaries accident (Event 4F) is 373,596 gpm. This represents a nominal RCS flow of 382,000 gpm minus a flow measurement uncertainty factor of 2.2%. The accident evaluation in the LAR remains valid

since the assumed RCS flow in the AOR is already less than the proposed flow of 387,000 gpm.

4G: The RCS flow given in Table 15.4 for the LONF (long-term) accident analysis represents a nominal flow of 385,000 gpm minus a 2.2% flow measurement uncertainty. The initial RCS flow for the Event 4G AOR is not ≤ 382,000 gpm, but is still below the proposed RCS flow of 387,000 gpm for Unit 2. The accident evaluation in the LAR remains valid for Event 4G since the assumed RCS flow in the AOR is already less than the proposed flow of 387,000 gpm.

4H: The RCS flow given in UFSAR Table 15.4 for the Feedwater System Pipe Break (long-term) accident analysis (Event 4H) is 373,596 gpm. This represents a nominal RCS flow of 382,000 gpm minus a flow measurement uncertainty factor of 2.2%. The accident evaluation in the LAR remains valid for Event 4H since the assumed RCS flow in the AOR is already less than the proposed flow of 387,000 gpm.

16. Event 6N – Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR 15.4.1)

The information discussing Event 6N indicates that the minimum DNBR for Case 1 AOR is 3.395 and peak RCS for Case 2 have a margin of 150 psi relative to the design value of 2735 psi. It does not provide the values of the RCS flow used in the AOR. In accordance with FSAR Table 15.4, it appears that the RCS flow rates used are 299,613 gpm and 375,669 gpm for Case 1 and Case 2, respectively. Please verify that the AOR for Case 1 and Case 2 in FSAR 15.4.1 remains valid with the required minimum RCS flow reduced from 390,000 gpm to 387,000 gpm for Unit 2.

Duke Energy Response:

Question #10 above pertains to the Bank Withdrawal from a Subcritical or Low Power Startup Condition accident analysis for Unit 1 (Event 3P), and poses a similar comment about the RCS flow given in UFSAR Table 15.4 as Question #16. The Catawba UFSAR presents results for the Bank Withdrawal from a Subcritical or Low Power Startup Condition accident analysis for Unit 1 only, due to the similarity in system response. The response for Question #10 is also applicable here with regard to the RCS flows given in Table 15.4 and the comment raised in Question #16.

A separate analysis for this accident is maintained for Catawba Unit 2, although the results are not presented in the UFSAR. The Catawba Unit 2 AOR results for Case 1 are given in the LAR; the minimum DNBR of 3.395 is well above the acceptance criteria of 1.45. The Catawba Unit 2 AOR results for Case 2 are also discussed in the LAR; there is about 150 psi of margin between the AOR results and the acceptance criteria for peak primary pressure. Both of the Catawba Unit 2 AORs assume an initial 4-pump RCS flow of 390,000 gpm. (There are reductions applied for the "stair-step" flow reduction and for the flow measurement

instrumentation uncertainty in Cases 1 and 2, and for 3-pump operation in Case 1. These are also discussed in the Question #10 response addressing the Unit 1 analyses.) It is concluded that the proposed flow reduction for Catawba Unit 2 will not have an appreciable impact on the results due to the large amount of margin between the current AOR results and the corresponding acceptance criteria for this accident.

ATTACHMENT 4

REQUESTED TOPICAL REPORTS ON COMPACT DISC

(NON-PROPRIETARY)