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

April 10, 2018

Mr. Steven Capps
Senior Vice President
Nuclear Corporate
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENTS REVISING TECHNICAL SPECIFICATIONS FOR METHODOLOGY REPORTS DPC-NE-3008-P, REVISION 0, "THERMAL-HYDRAULIC MODELS FOR TRANSIENT ANALYSIS," AND DPC-NE-3009-P, REVISION 0, "FSAR / UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY" (CAC NOS. MF8439 and MF8440; EPID L-2016-LLA-0012)

Dear Mr. Capps:

By letter dated November 19, 2015, as superseded by application dated October 3, 2016, and as supplemented by letters dated November 10, 2016, October 9 and 30, 2017, and December 19, 2017, Duke Energy Progress, LLC (Duke Energy or the licensee) submitted a request, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.90, for Shearon Harris Nuclear Power Plant, Unit 1 (Harris) and H. B. Robinson Steam Electric Plant, Unit 2 (Robinson) to amend the technical specifications (TSs) to adopt the methodology reports DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis," and DPC-NE-3009-P, Revision 0, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," for application specific to Harris and Robinson. In its submittal, Duke Energy requested review and approval of these methodologies for plant-specific use only, allowing these methodologies to be used at Harris and Robinson to perform the subject analyses in-house.

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued Amendment No. 164 to Renewed Facility Operating License No. NPF-63 for Harris and Amendment No. 257 to Renewed Facility Operating License No. DPR-23 for Robinson. These amendments change the TSs in response to your submittal dated November 19, 2015, as superseded by application dated October 3, 2016, and as supplemented by letters dated November 10, 2016, October 9 and 30, 2017, and December 19, 2017.

**Enclosure 3 transmitted herewith contains Sensitive Unclassified Non-Safeguards Information. When separated from Enclosure 3, this document is decontrolled.**

The NRC has completed its review of the information provided by the licensee. The NRC staff's safety evaluation (SE) is enclosed. The NRC staff has determined that the enclosed SE (Enclosure 3) contains proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has prepared a redacted, nonproprietary version of the SE (Enclosure 4). However, the NRC will delay placing the nonproprietary SE in the public document room for a period of 10 working days from the date of this letter to provide Duke Energy the opportunity to comment on any proprietary aspects. If you believe that any information in Enclosure 4 is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the nonproprietary SE will be made publicly available.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions concerning this letter, please contact me at 301-415-6256 or by email at Dennis.Galvin@nrc.gov.

Sincerely,

Dennis J Galvin

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

Docket Nos. 50-261 and 50-400

Enclosures:
1. Amendment No. 164 to NPF-63
2. Amendment No. 257 to DPR-23
3. Safety Evaluation PROPRIETARY
4. Safety Evaluation NONPROPRIETARY

cc w/enclosures 1, 2, and 4: Distribution via Listserv **10 working days after issuance**

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 164
Renewed License No. NPF-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated November 19, 2015, as superseded by application dated October 3, 2016, and as supplemented by letters dated November 10, 2016, October 9 and 30, 2017, and December 19, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 164, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian W. Tindell

Brian W. Tindell, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and the Technical Specifications

Date of Issuance: April 10, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 164

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-400

Replace page 4 of the Renewed Facility Operating License No. NPF-63 with the attached revised page 4. The revised page is identified by amendment number and contains marginal lines indicating the areas of change,

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

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C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 164, are hereby incorporated into this license. Duke Energy Progress, LLC. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1] The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.
*On April 29, 2013, the name "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc." On August 1, 2015, the name "Duke Energy Progress, Inc." was changed to "Duke Energy Progress, LLC."

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

t. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated April 10, 2018. (Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

u. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated April 10, 2018. (Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

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

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

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

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

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

SPECIAL REPORTS

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

6.10 DELETED

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

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 257
Renewed License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated November 19, 2015, as superseded by application dated October 3, 2016, and as supplemented by letters dated November 10, 2016, October 9 and 30, 2017, and December 19, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 257, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian W. Tindell

Brian W. Tindell, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and the Technical Specifications

Date of Issuance: April 10, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 257

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

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace page 3 of the Renewed Facility Operating License No. DPR-23 with the attached revised page 3. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Replace the following page of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 257 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

26. BAW-10240(P)(A), “Incorporation of M5 Properties in Framatome ANP Approved Methods,” approved version as specified in the COLR.

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

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

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

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

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

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

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

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status,

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO RENEWED FACILITY OPERATING LICENSE

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

AND AMENDMENT NO. 257 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-23

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, DOCKET NO. 50-261

DUKE ENERGY PROGRESS, LLC

## 1.0 INTRODUCTION

By application dated November 19, 2015 (Reference 1), as superseded by application dated October 3, 2016 (Reference 2), and supplemented by letters dated November 10, 2016 (Reference 3), October 9, 2017 (Reference 4), October 30, 2017 (Reference 5), and December 19, 2017 (Reference 6), Duke Energy Progress, LLC (Duke Energy), the licensee for Shearon Harris Nuclear Power Plant, Unit 1 (Harris) and H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson), requested changes to the Harris and Robinson Technical Specifications (TSs). Specifically, the licensee requested U. S. Nuclear Regulatory Commission (NRC) review and approval of DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis" and DPC-NE-3009-P, Revision 0, "FSAR / UFSAR [Final Safety Analysis Report / Updated FSAR] Chapter 15 Transient Analysis Methodology," and incorporation of these reports into the lists of NRC-approved core operating limits report (COLR) references of Robinson TS 5.6.5.b and Harris TS 6.9.1.6.2.

The proposed TS revisions and methodology reports would allow the licensee to perform reactor safety analyses as part of the core reload design process at Harris and Robinson. The new methods for analysis using RETRAN-3D, SIMULATE-3K, and VIPRE-01 would replace the relevant AREVA methodologies. The licensee considers the safety analysis methods documented in DPC-NE-3008 and DPC-NE-3009 to be an evolution of the safety analysis methods described in DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology," (Reference 7) and DPC-NE-3001, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology" (Reference 8), which use RETRAN-02, SIMULATE-3K, and VIPRE-01. Like the DPC-NE-3000 methodology, which was only applicable to the Catawba Nuclear Station, McGuire Nuclear Station, and Oconee Nuclear Station (Catawba, McGuire, and Oconee, respectively), the DPC-NE-3008 and DPC-NE-3009 methodologies are only applicable to Harris and Robinson and were reviewed by the NRC staff on a plant-specific basis for those facilities.

The NRC staff accepted the original license amendment request for review by email dated January 8, 2016 (Reference 9). Requests for additional information (RAIs) regarding DPC-NE-3008 were submitted to the licensee on October 24, 2016 (Reference 10), and RAIs regarding DPC-NE-3009 were submitted to the licensee on September 8, 2017 (Reference 11),

Enclosure 4

and December 11, 2017 (Reference 12). The licensee responded to these requests in letters dated November 10, 2016 (Reference 3), October 9, 2017 (Reference 4), October 30, 2017 (Reference 5), and December 19, 2017 (Reference 6).

On April 5, 2016, the NRC staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (81 FR 19645) for the proposed amendments. Subsequently, by letters dated October 3, 2016 and November 10, 2016, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, the NRC published a second proposed NSHC determination in the *Federal Register* on May 2, 2017 (82 FR 20496), which superseded the original notice in its entirety. The supplemental letters dated October 9, October 30, and December 19, 2017, provided additional information that clarified the application, did not expand the scope of the application as noticed on May 2, 2017, and did not change the staff's proposed no significant hazards consideration determination as published on May 2, 2017.

## 2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(b) requires a final safety analysis report (FSAR) to be submitted as part of an application for an operating license. In particular, 10 CFR 50.34(b)(2) requires the FSAR to contain "evaluations required to show that safety functions will be accomplished" for plant structures, systems, and components. The models and methods proposed by the licensee in DPC-NE-3008 and DPC-NE-3009 will be used to perform safety analyses in Chapter 15 of the Harris and Robinson Updated FSARs (UFSARs).[1]

According to 10 CFR 50.36, licensees are required to include TSs "derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34." The analyses performed using the models and methods proposed by the licensee may be used to set TS limits or verify that TS limits are met for a given cycle.

Paragraph 10 CFR 50.36(c)(5) requires TSs to include items in the category of administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved TSs as specified in 10 CFR 50.4.

Harris was licensed to the standard of the general design criteria (GDC) included in Appendix A to 10 CFR Part 50. The analyses performed using the models and methods proposed by the licensee in DPC-NE-3008 and DPC-NE-3009 will be used to evaluate compliance with several GDC, including:

- Criterion 10, which requires the reactor core and associated coolant, control and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
- Criterion 15, which requires the reactor coolant system (RCS) and associated systems to be designed with sufficient margin to assure that the design conditions of the reactor

[1] The licensee refers to the UFSAR for Harris as an FSAR. However, since the Harris FSAR has been updated, the staff refers to it as an UFSAR throughout the safety evaluation (SE).

coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- Criterion 20, which requires protection systems to be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.
- Criterion 28, which requires reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity releases to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Robinson was not licensed to the 10 CFR Part 50 Appendix A GDC, and was instead licensed to an older draft of the GDC. The requirements are roughly similar. The Robinson design criterion discussed in UFSAR Section 3.1.2.6 corresponds to GDC 10, Sections 3.1.2.14 and 15 correspond to GDC 20, and Sections 3.1.2.30 through 3.1.2.33 correspond to GDC 28. There is no direct analogue in the Robinson design criteria to GDC 15, though the criterion discussed in Section 3.1.2.9 provides requirements for overpressure mitigation and other sections discuss the need to provide protection to the reactor coolant pressure boundary during reactivity transients.

Robinson UFSAR Section 3.1.2.6, “Reactor Core Design,” states, in part, that:

> The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated. (GDC 6)

Robinson UFSAR Section 3.1.2.9, “Reactor Coolant Pressure Boundary,” states, in part, that:

> The reactor coolant pressure boundary (RCPB) shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (GDC 9)

Robinson UFSAR Section 3.1.2.14, “Core Protection Systems,” states, in part, that:

> Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. (GDC 14)

Robinson UFSAR Section 3.1.2.15, “ESF [Engineered Safety Feature] Protection Systems,” states, in part, that:

> Protection systems shall be provided for sensing accident situations and initiating the operation of necessary ESF. (GDC 15)

Robinson UFSAR Section 3.1.2.30, “Reactivity Holddown Capability,” states, in part, that:

> The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public. (GDC 30)

Robinson UFSAR Section 3.1.2.31, “Reactivity Control Systems Malfunction,” states, in part, that:

> The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31)

Robinson UFSAR Section 3.1.2.32, “Maximum Reactivity Worth of Control Rods,” states, in part, that:

> Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot:
>
> a) Rupture the RCPB, and
>
> b) Disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core. (GDC 32)

Robinson UFSAR Section 3.1.2.33, “RCPB Capability,” states, in part, that:

> The RCPB shall be capable of accommodating without rupture the static and dynamic load imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33)

Guidance for the NRC staff’s review of models and methodologies and the associated design criteria are provided in NUREG-0800, the Standard Review Plan (SRP). Chapter 15 of the SRP provides staff guidance for the review of transient and accident analyses, and in particular, SRP Section 15.0 (Reference 13) provides general, overall guidance, SRP Section 15.0.2 (Reference 14) provides guidance on the review of transient and accident analysis methods, and the other SRP Chapter 15 sections correspond to the accidents the licensee intends to use DPC-NE-3009 to analyze (with the models proposed in DPC-NE-3008).

The guidance in SRP Section 15.0 largely assumes that codes and models have been reviewed and approved by the NRC staff. The codes that the licensee has proposed to use in this methodology report have been reviewed and approved by the NRC, either generically (in the case of VIPRE-01 and RETRAN-3D) or on a plant-specific basis in other Duke Energy methodology reports (in the case of SIMULATE-3K). However, the models proposed for use in performing the analyses of Chapter 15 transients for Harris and Robinson have been provided by the licensee in the present review and have not previously been reviewed by the NRC staff. Though there are substantial similarities between the models proposed in the current effort and previous Duke Energy models provided in DPC-NE-3000 (Reference 7) and DPC-NE-3001 (Reference 8) that provide a basis for the current review, there are also differences that must be addressed in this safety evaluation (SE). Additionally, the models and methods provided in DPC-NE-3000 and DPC-NE-3001 were for Catawba, McGuire, and Oconee; the licensing bases for these plants differ from those of Harris and Robinson, and these differences must also be addressed.

These differences in models and methods from previously NRC-approved methodologies like DPC-NE-3000 and DPC-NE-3001 will be reviewed according to SRP Section 15.0.2. Though it is not directly applicable to this circumstance and is largely written considering larger-scale code reviews and loss of coolant accident (LOCA) evaluation models, there are elements of the SRP section that are relevant to the review and will be discussed. The detailed review of models and methods for each individual UFSAR Chapter 15 transient was conducted in accordance with the corresponding SRP section for that transient.

Additionally, Chapter 4 of the SRP provides guidance on the review of the reactor core design. This is relevant to the review of DPC-NE-3008 and DPC-NE-3009 because this portion of the SRP provides guidance on the specified acceptable fuel design limits discussed in GDC 10, with which the Chapter 15 safety analyses will be used to demonstrate compliance. In particular, SRP Section 4.2 (Reference 15) provides staff guidance for the review of plant fuel system design (including an appendix that provides acceptance criteria for reactivity initiated accidents, like the control rod ejection (CRE) accident), SRP Section 4.3 (Reference 16) provides staff guidance for the review of plant nuclear design (including guidance on acceptable values of core physics parameters), and SRP Section 4.4 (Reference 17) provides staff guidance for the review of plant thermal and hydraulic design (including methods for setting and evaluating critical heat flux (CHF) limits).

The NRC staff, therefore, also used SRP Section 4.3 to provide guidance on the review of the generic safety analysis physics parameters provided in Section 4 of the DPC-NE-3009 and the physics parameters discussed for each transient in Section 5 of DPC-NE-3009. Acceptance criteria in these same transient sections for fuel rod thermal-mechanical and thermal-hydraulic limits were reviewed by the NRC staff considering SRP Sections 4.2 and 4.4, respectively.

## 3.0 TECHNICAL EVALUATION

### 3.1 Codes used in UFSAR Chapter 15 Non-LOCA Transient Analyses

The licensee's methodology for analyzing the Harris and Robinson UFSAR Chapter 15 non-LOCA transient events relies on three primary codes. SIMULATE-3K, a two-group nodal code for transient reactor core analyses, is used to calculate power responses and physics parameters, particularly for the CRE accident. RETRAN-3D, a system thermal-hydraulic transient analysis code, is used to analyze the nuclear steam supply system (NSSS)

thermal-hydraulic and system responses to transients. VIPRE-01, a subchannel thermal-hydraulic code, is used to evaluate the minimum departure from nucleate boiling ratio (MDNBR), and, in some circumstances, fuel temperature and enthalpy deposition. The following sections of this SE will discuss the codes and the selected modeling options in additional detail.

### 3.1.1 SIMULATE-3K

SIMULATE-3K is a two-group nodal code that models both neutronics and thermal-hydraulics for transient reactor core analyses. It is a transient version of SIMULATE-3 that uses the same nuclear data library but adds the time-dependent equations needed to solve transient, three-dimensional, coupled neutronics and thermal-hydraulic problems. SIMULATE-3K is used within DPC-NE-3009 for analysis of events where the core power distribution changes substantially over the course of the event. This means that it is primarily used to calculate the 3D core power distribution and transient core power response following a CRE accident. The fuel rod models in the code are also used when evaluating the CRE to identify limiting fuel pins for further evaluation using VIPRE-01. Another application includes determining the effects of core moderator voiding on reactivity and power distributions for certain transients.

Though SIMULATE-3K has not been approved on a generic basis by the NRC staff, it has been used in a number of licensing applications by the licensee, most recently in DPC-NE-3001, Revision 1[2] (Reference 18), which has been approved by the NRC staff (Reference 19). DPC-NE-3001, Revision 1 describes the use of SIMULATE-3K for analyses at Catawba and McGuire and DPC-NE-3005 (Reference 20), as approved, describes the use of SIMULATE-3K at Oconee. The NRC staff evaluated the applications of SIMULATE-3K within the DPC-NE-3009 methodology and determined that they are consistent with previously approved uses of the code. The core models used as inputs to SIMULATE-3K are the same as the SIMULATE-3 steady-state core models corresponding to the cycle being analyzed; SIMULATE-3 was approved for nuclear design and analysis at Harris and Robinson in DPC-NE-1008 (Reference 21).

### 3.1.2 RETRAN-3D

RETRAN-3D is a general-purpose thermal-hydraulic transient analysis code, with capabilities to model conductive heat structures, reactor systems (including control systems), three-dimensional reactor kinetics, and fuel decay heat. RETRAN-3D will be used to determine the system-level thermal-hydraulic response, and as such will serve as the primary analysis tool for the Harris and Robinson UFSAR Chapter 15 transients. Values of system pressures from RETRAN-3D will be directly used to confirm that pressure limits are met, while other system parameters (pressure, temperature, etc.) will be sent to VIPRE-01 or SIMULATE-3K for use in further analysis.

Since RETRAN-3D is a general purpose code, it includes a numerous modeling options and correlations that must be selected by the user. Plant geometry and systems must also be appropriately represented using the models available in the code. To enumerate some of the chosen modeling options and correlations, the licensee submitted RETRAN-3D plant models, which also discuss plant geometry and equipment, as part of DPC-NE-3008 and DPC-NE-3009 for the NRC staff's review.

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[2] Hereafter, referrals to DPC-NE-3001 are to Revision 1.

RETRAN-3D was approved for generic use by the NRC staff (Reference 22). The NRC staff's SE on RETRAN-3D includes 45 conditions and limitations on the use of the code, many of which deal with these options. Compliance with these conditions and limitations is discussed in Section 3.2, as are the models and correlations selected by the licensee for use in the Harris and Robinson thermal-hydraulic safety analysis methodologies.

### 3.1.3 VIPRE-01

VIPRE-01 (Reference 23) is a subchannel thermal-hydraulics analysis code, based on a homogeneous formulation of the mass, energy, and momentum equations with additional models for subcooled boiling and liquid/vapor slip. It has been reviewed and approved by the NRC staff (Reference 24). The licensee's use of the code is primarily described in DPC-NE-2005, Revision 4a (Reference 25). Subsequently, the licensee submitted DPC-NE-2005, Revision 5[3] (Reference 26), which was reviewed and approved by the NRC staff for use at Harris and Robinson (Reference 27).

VIPRE-01 will be used to evaluate the MDNBR, and, in some transient analyses, fuel temperature and enthalpy deposition. A **[[ ]]** model is used to evaluate departure from nucleate boiling (DNB) for UFSAR Chapter 15 transients, and was presented in DPC-NE-2005, which also documented the use of the statistical core design methodology for Harris and Robinson. DPC-NE-3008 also describes expanded models for Harris and Robinson **[[ ]]** as options for additional licensing calculations. These models are discussed in Section 3.3.3.3 of this SE, but are not used in any of the analyses described in DPC-NE-3009. However, DPC-NE-3009 presents two additional VIPRE-01 models for evaluating DNB for main steam line break (SLB) and rod ejection. These models will be discussed in the Sections 3.3.3.1 and 3.3.3.2 of this SE.

Limitations and conditions for the use of the VIPRE-01 base models are discussed in DPC-NE-3008. Other issues related to the use of VIPRE-01 as proposed in DPC-NE-3009 are discussed here, or in the appropriate sections.

## 3.2 RETRAN-3D Plant Models

Base plant models of Harris and Robinson using RETRAN-3D were provided in DPC-NE-3008. Two additional models with new nodalizations were added in DPC-NE-3009.

### 3.2.1 Base RETRAN-3D Plant Models for Harris and Robinson from DPC-NE-3008

The RETRAN-3D plant models for Harris and Robinson provided in DPC-NE-3008 are largely similar to those approved by the NRC for use at Catawba and McGuire in DPC-NE-3000, with some modifications due either to differences between Harris/Robinson and Catawba/McGuire or improvements that have been identified by the licensee since the DPC-NE-3000 models were first developed.

[3] Hereafter, referrals to DPC-NE-2005 are to Revision 5.

3.2.1.1 Geometry

3.2.1.1.1 Reactor Vessel

The reactor vessel model proposed for use with Harris and Robinson is essentially the same as in DPC-NE-3000 with two major differences.

First, Harris and Robinson do not have the same **[[** **]]** as Catawba and McGuire. **[[**

**]]**. The Harris and Robinson plant geometry is appropriately preserved in the proposed DPC-NE-3008 model and is therefore acceptable.

Second, Robinson has a **[[**

**]]**. Any differences in temperature or pressure that would result from **[[**
**]]** are expected to be minimal and have a negligible impact on the overall results.

3.2.1.1.2 Reactor Coolant System Loops

The RCS loops are modelled in essentially the same manner as in the NRC-approved DPC-NE-3000. The primary difference is that where the Catawba and McGuire base models included two loops (one for the RCS loop with the pressurizer, and a second lumped loop for the remaining three loops in the RCS), the Harris and Robinson base models explicitly include all three loops in the RCS. This is acceptable because it will more accurately model the plant response.

3.2.1.1.3 Steam Generators

The steam generators (SGs) are modeled with few differences from what is described in the NRC-approved DPC-NE-3000. The primary difference is that the DPC-NE-3008 model includes **[[**

**]]**. Primary steam separators are modeled explicitly, though secondary steam separators are still lumped into the steam dome region. The feedwater distribution ring is also modeled more explicitly. Tube plugging is considered, but the quantity varies depending on the assumptions of the transient and will be discussed in more detail in Section 3.2.1.2.9 of this SE.

The modifications to the DPC-NE-3000 SG model are expected to provide enhanced computational accuracy in the new Harris and Robinson base models. They are therefore considered to be acceptable to the NRC staff.

3.2.1.1.4 Pressurizer

The pressurizer geometry is modeled in the same manner as the NRC-approved DPC-NE-3000 model and is therefore acceptable. Modifications to the pressurizer models (e.g., use of inter-region heat transfer, local conditions heat transfer, etc.) will be discussed in Sections 3.2.2.3.1, 3.2.2.3.2, 3.2.2.3.3, and 3.2.2.3.4 of this SE.

3.2.1.1.5 Accumulators

The accumulators are modeled in essentially the same manner as the NRC-approved DPC-NE-3000, except the built-in RETRAN-3D accumulator model is now used; this will be discussed in Section 3.2.1.2.16 of this SE. **[[**
**]]**; this is more realistic than the DPC-NE-3000 model and is thus acceptable.

3.2.1.1.6 Feedwater

The feedwater injection into the SGs is similar to that used in the NRC-approved DPC-NE-3000, except the main feedwater (MFW) and auxiliary feedwater (AFW) injection sites are switched. Using fill junctions, MFW is injected **[[**
**]]**. The NRC staff reviewed the plant design information included in the UFSARs and determined that the modeling of feedwater was appropriate to the plant designs and is therefore acceptable.

3.2.1.1.7 Main Steam Line

The primary difference between the previously approved DPC-NE-3000 model and the proposed DPC-NE-3008 model is that all main steam lines are now explicitly modeled individually since each loop is explicitly modeled (instead of lumped, as discussed in Section 3.2.1.1.3). All main steam safety valves (MSSVs) and SG power-operated relief valves (PORVs) are now explicitly modeled, rather than lumped into a single valve. Steam dumps are also explicitly modeled. These changes to the main steam line modeling are consistent with changes to the RCS loop modeling and, in the case of the MSSV, SG PORV, and steam dump modeling, will more accurately represent the actual physical design of the plants. The NRC staff therefore determined that the main steam line modeling was acceptable.

3.2.1.2 Code Models and Options

3.2.1.2.1 Power

Plant power in the proposed Harris and Robinson base models is the same as in the NRC-approved DPC-NE-3000. The built-in RETRAN-3D point kinetics are used to simulate the power response during transients. Post-trip decay heat is modeled using the American Nuclear Society (ANS) ANS-5.1-1979 standard. These modeling choices are acceptable to the NRC staff for use in UFSAR Chapter 15 non-LOCA transient analyses at Harris and Robinson conducted using the methods discussed in DPC-NE-3008 and DPC-NE-3009.

3.2.1.2.2 Pumps

The Robinson base deck uses the built-in RETRAN-3D pump curves to simulate Westinghouse Model 93 pumps. The Harris base deck, like the McGuire deck presented in DPC-NE-3000,

**[[**

**]]**. Since the curves used for Harris are the same as used in the approved McGuire model, they are acceptable. **[[**

**]]**.

3.2.1.2.3 Valves

Basic RETRAN-3D valve models are used. Specific valve characteristics (flow area, open/close rate, etc.) depend heavily on the particular valve being simulated in the plant. This is the same approach as the NRC-approved DPC-NE-3000 model and is therefore acceptable.

3.2.1.2.4 Phase Separation

Algebraic Slip Model using the Chexal-Lellouche Correlation

In a departure from the modeling approach taken in DPC-NE-3000, the DPC-NE-3008 base models use the algebraic slip model with the Chexal-Lellouche drift flux correlation **[[**

**]]**. While the use of this correlation was previously approved for use in the Oconee once-through SG model, the feedring SG design at Harris and Robinson is substantially different.

The licensee's response to RAI 4 (Reference 3) provides justification for the use of Chexal-Lellouche and dispositions the relevant conditions and limitations from the NRC staff's SE on RETRAN-3D. **[[**

**]]**.

The source cited by the licensee for information on Chexal-Lellouche in the RAI response, Electric Power Research Institute (EPRI) Report TR-106326, "Void Fraction Technology for Design and Analysis," (Reference 28) **[[**

**]]**.

The NRC staff also notes RETRAN-3D Condition 16 states that Chexal-Lellouche cannot be used in situations where counter current flooding limitation (CCFL) is important, unless validation is provided for the specific geometry and flow conditions expected in the transient. **[[**

**]]**.

RETRAN-3D Condition 16 also states that results of analyses that use the Chexal-Lellouche correlation in pressure ranges of 12 to 17 mega Pascals (MPa) and 7.5 to 10 MPa (1740-2466 pounds per square inch absolute (psia) and 1088-1450 psia, respectively) must be carefully reviewed. **[[**

**]]** The licensee's demonstration analyses show that this range will be entered only for **[[**

**]]**

The NRC staff reviewed available validation data for Chexal-Lellouche and found that it provides overall acceptable results below approximately 10 MPa **[[**

**]]**. In response to an RAI on the generic review of RETRAN-3D, EPRI provided a paper that used Chexal-Lellouche within RETRAN-3D to predict data from experiments that covered a wide range of pressure, mass flux, and heat flux conditions (Reference 29). Of the data studied, those from the Large Scale Test Facility and Two-Phase Flow Test Facility most closely represent the conditions expected in the **[[**

**]]**. The paper demonstrated that Chexal-Lellouche, as implemented within RETRAN-3D, provides good predictions of the void fraction for these tests, and has a slight tendency to overpredict the void fraction.

Consistent with Condition 16, **[[**

**]]**. In light of the validation data discussed above, the NRC staff finds this to be an adequate disposition of Condition 16 on the use of RETRAN-3D.

Overall, the NRC staff found the use of the Chexal-Lellouche correlation proposed in DPC-NE-3008 to be acceptable with the caveats discussed above. Accordingly, the NRC staff considers RAI 4 regarding phase separation resolved.

<u>Accumulators</u>

Accumulators are now modeled with the built-in RETRAN-3D accumulator component rather than a bubble rise volume. Overall modeling of accumulators will be discussed in Section 3.2.1.2.16 of this SE.

**[[ ]]**

Consistent with the previously approved DPC-NE-3000, bubble rise volumes are used to model the **[[ ]]**. This is acceptable to the NRC staff.

Pressurizer

The DPC-NE-3008 base models for Harris and Robinson depart from the approved approach in DPC-NE-3000 **[[ ]]**. However, as discussed in the response to RAI 6 (Reference 3), **[[ ]]**.

3.2.1.2.5 Non-Conducting Heat Exchangers

DPC-NE-3008 described one use of non-conducting heat exchangers in the Harris and Robinson base plant models, to be used to simulate the heater banks in the pressurizer. This is consistent with the NRC-approved approach from DPC-NE-3000 and is thus acceptable. Additional uses in other components were discussed in DPC-NE-3009 and will be addressed in Section 3.2.2.3.3 of this SE.

3.2.1.2.6 Local Conditions Heat Transfer

The Harris and Robinson base models use the RETRAN-3D local conditions heat transfer model in bubble rise volumes in **[[ ]]**. This model is generally applied when there are stacks of heat conductors positioned adjacent to a bubble rise volume. When the model is used, the location of a given heat conductor relative to the vapor mixture interface is used to determine the local fluid conditions. The local fluid conditions are then used to select and evaluate the wall heat transfer model. The use of this model in **[[ ]]** is acceptable to the NRC staff, because it will improve the modeling of heat losses from **[[ ]]**.

In the SG, bubble rise volumes are used in the SG **[[ ]]**. However, given the design of the Harris and Robinson SGs and the proposed nodalization in these regions, the NRC staff expects **[[ ]]** will adequately represent the heat transfer. The NRC staff therefore determined that the licensee's use of the local conditions heat transfer model is acceptable.

3.2.1.2.7 Steady-State Model Initialization

The licensee's response to RAI 9 (Reference 3) provides information on the initialization process.

The primary system conditions are set by specifying the core power, pressurizer pressure, pressurizer level, flow rates, and reactor vessel lower plenum enthalpy (thus specifying the subcooling at the core entrance). The primary difference between the proposed method in DPC-NE-3008 and the NRC-approved method in DPC-NE-3000 is that the approved method specified the enthalpy in the cold leg. The difference is acceptable, because the lower plenum enthalpy will allow the RETRAN-3D steady-state initialization routine to be able to determine the heat balance across the core just as well as the cold leg enthalpy. When parallel flow paths exist, a junction form loss coefficient may be unspecified in one of the flow paths. If the flow is specified, the RETRAN-3D initialization then balances the pressure loss between the parallel paths.

Secondary system conditions are set by specifying MFW and main steam flow, MFW enthalpy, upper downcomer enthalpy, circulation flow, steam dome pressure, and levels in the steam dome and separator. The primary differences between the proposed method in DPC-NE-3008 and the NRC-approved method in DPC-NE-3000 result from the use of the bubble rise model in the **[[ ]]**. A wide range of parameters may be adjusted during the RETRAN-3D initialization to obtain the desired initial SG level, mass, and primary-to-secondary system heat transfer. This is consistent with the approved method in DPC-NE-3000 and is therefore acceptable.

The NRC staff's SE on the use of RETRAN-3D discussed the use of the steady-state initializer in Condition 29. The NRC staff position on this condition states that adjustments to plant parameters made during the initialization process should be reviewed by the user on an application-specific basis to ensure that they will not negatively impact the transient results. The NRC's SE also discusses techniques to ensure well-posed initial conditions, such as null transients to check that unwanted control or trip actions do not affect the solution and output checks on the adjustments made automatically by the code during the initialization routine. The licensee stated in the RAI response that these techniques will be used with the RETRAN-3D base models for Harris and Robinson. The NRC staff determined that the licensee's approach for initialization, and for ensuring that models initialized using automated approaches in RETRAN-3D would provide acceptable transient results, is acceptable. Accordingly, the NRC staff considers RAI 9 resolved.

3.2.1.2.8 Time Step Control

Time step control is set to be in automatic, using the iterative numerics option. This solution method uses predictive algorithms to calculate a time step size that will result in a stable, accurate solution for the fluid conservation equations; if this time step size fails to result in a converged solution in a specified number of iterations, a smaller time step size is used and the forward step is re-evaluated. This is consistent with DPC-NE-3000 and is acceptable because it will result an accurate and stable solution.

3.2.1.2.9 Enthalpy Transport Model

The enthalpy transport model is a junction enthalpy model that accounts for differences in enthalpy between the center of the volume and the exit caused by various factors. It is typically intended for use in large volume nodes that experience heat input, to improve mass and temperature distribution calculation accuracy. In DPC-NE-3000, the enthalpy transport model was used **[[ ]]**. In DPC-NE-3008, the model is used in the same places as the

DPC-NE-3000 model except for **[[ ]]**. This is because the model experiences issues that tend to cause anomalous conditions at low flow and low heat transfer conditions. Thus, the model is particularly problematic for initialization of certain transients.

The issue was discussed in detail in Revision 1 of DPC-NE-3001 (Reference 8), the licensee's response to the NRC's RAIs (Reference 30), and the NRC staff's SE approving the revised methodology report (Reference 19), where deactivation of the enthalpy transport model was found acceptable by the NRC staff. The modifications to the SG nodalization in the Harris and Robinson base models relative to the McGuire and Catawba models will allow the enthalpy distribution in the SG secondary side to be resolved at a fine enough level that the impacts of deactivating the enthalpy transport model will be mitigated. The NRC staff therefore finds this approach to be acceptable.

3.2.1.2.10 Temperature Transport Delay

The temperature transport delay model exists to better track temperature change fronts, which may result in single-phase fluid volumes when the fluid entering a volume is at a different temperature from, and does not mix with, the fluid in the volume. The standard method in RETRAN-3D for determining the junction enthalpy is to homogeneously mix the incoming fluid with the fluid already in the volume; thus, fluid of a different temperature entering a volume instantaneously affects the enthalpy at the volume outlet junction. The temperature transport delay model considers the movement of fluid through a region as a slug, and creates an enthalpy mesh substructure within the given control volume to track temperature front movement.

In DPC-NE-3000, the temperature transport delay option was used in all of the MFW and RCS piping volumes. In the Harris and Robinson base models described in DPC-NE-3008, the option is not used. However, if there are licensing applications of the base models where significant temperature changes across fluid volumes are encountered, the temperature transport delay option may be activated. The NRC staff determined that this approach is acceptable, but notes that a failure to observe the Courant limit for the transport delay mesh may result in anomalous mesh enthalpy calculations and potential code failures.

3.2.1.2.11 Heat Transfer Map

The **[[ ]]** available in RETRAN-3D is used in the Harris and Robinson base models, as in the NRC-approved DPC-NE-3000. The **[[**

**]]**.

3.2.1.2.12 Film Boiling and Critical Heat Flux

As in the NRC-approved DPC-NE-3000 models, the **[[ ]]** model is used to determine the film boiling heat transfer. This is acceptable to the NRC staff.

Because **[[**

**]]**. This is **[[**
**]]** and is acceptable to the NRC staff.

3.2.1.2.13 Volume Flow Calculation

In DPC-NE-3000, the licensee previously used the donor-cell option for calculating the volume flow for the momentum flux. In DPC-NE-3008, the Harris and Robinson base models use the built-in arithmetic average formulation (a second-order central differencing scheme) to calculate the momentum flux (because, according to the licensee, the donor-cell option was removed from RETRAN-3D). However, the arithmetic average formulation may result in numerically unstable results in certain flow situations, principally in those where the momentum flux dominates the other terms in the momentum conservation equation.

In the response to RAI 10 (Reference 3), the licensee stated that regular interactions with the RETRAN User Group and the code developer, Computer Simulation and Analysis/Zachry Nuclear Engineering, will help minimize the potential for the code to be applied in these scenarios. Additionally, the licensee stated that analysis results are carefully reviewed by engineers with many years of experience in applying RETRAN-02 and RETRAN-3D to other Duke Energy plants, helping to mitigate errors. If numerical instabilities are observed and attributed to the arithmetic average formulation for calculating the momentum flux, the momentum flux may be deactivated in the affected junctions if justified. Alternatively, the licensee would work with the code vendor to determine the appropriate other options to resolve the observed behavior, and would work to reactivate the donor-cell formulation if needed. The NRC staff determined that the licensee's response is consistent with the need to maintain quality assurance procedures for licensee use of third party codes as discussed in Generic Letter 83-11 (Reference 31) and reflects current best practices in software development, use, and maintenance. The licensee's proposed process for identifying and mitigating potential sources of numerical instability resulting from the momentum flux calculation is therefore acceptable. Additionally, the NRC staff expects that such instances of numerical instability resulting from the arithmetic average formulation of the momentum conservation equation will be rare. Accordingly, the NRC staff considers RAI 10 resolved.

3.2.1.2.14 Wall Friction

The Harris and Robinson base plant models presented in DPC-NE-3008 employ the Colebrook model to determine the turbulent friction. This is a generally accepted turbulent friction model and its use is acceptable to the NRC staff. The **[[ ]]** two-phase friction multiplier is also used; this was previously used in DPC-NE-3000 and is considered acceptable.

3.2.1.2.15 General Transport Model

Unlike the NRC-approved DPC-NE-3000 models for Catawba and McGuire, the base models for Harris and Robinson do not employ the general transport model. However, the general transport model is used to model the transport of boron within the RCS in certain licensing applications. These applications will be discussed in Section 3.2.2.3.5 of this SE.

3.2.1.2.16 Accumulators

The Harris and Robinson base plant models in DPC-NE-3008 use RETRAN-3D's built-in accumulator model. The NRC-approved models in DPC-NE-3000 use a bubble-rise model for the accumulators; however, because it uses an isothermal expansion for the cover gases, this approach has been found by the RETRAN code developer to overpredict the cover gas pressure and therefore also the accumulator flow. The accumulator model in the current version of RETRAN-3D, as discussed by the licensee in the response to RAI 12 (Reference 3), separately models the energy equation of the cover gas to appropriately simulate the polytropic expansion of the cover gas that occurs as the accumulator empties.

Section III.11.0 of Volume 4 of EPRI NP-7450, RETRAN-3D's applications manual (Reference 32), discusses the experimental validation of the accumulator model. The model was compared to accumulator data from the LOFT and SEMISCALE experiments, where it provided very good predictions of accumulator liquid level and pressure for both short transients (where the response is effectively isentropic) and long transients (where heat transfer from the accumulator vessel wall and liquid becomes important to the overall response, and the response shifts towards isothermal). The NRC staff therefore concluded that the new RETRAN-3D accumulator model is acceptable for use in the plant models for Harris and Robinson described in DPC-NE-3008 and DPC-NE-3009. Accordingly, the NRC staff considers RAI 12 resolved.

3.2.2 Alternate RETRAN-3D Plant Models from DPC-NE-3009

3.2.2.1 Two-Region Core Nodalization

One of the additional nodalizations provided in DPC-NE-3009 divides the core into two azimuthal regions **[[**
**]]**. This model is very similar to the model approved for use with SLB analyses at Catawba and McGuire in DPC-NE-3001, with modifications to suit the three-loop plant geometry at Harris and Robinson.

For the events analyzed with this model, the correct prediction of the asymmetric cooldown, and therefore the correct prediction of mixing between the faulted and unfaulted regions of the core, is considered by the NRC staff to be very important. The licensee's response to RAI 1a (Reference 5) indicated that **[[**

**]]**. The licensee performed a sensitivity study to show that, **[[**

**]]**. The NRC staff therefore finds the licensee's mixing assumptions to be acceptable. Accordingly, the NRC staff considers RAI 1a resolved.

3.2.2.2 Single-Volume Steam Generator Nodalization

The other nodalization provided in DPC-NE-3009 combines the SG secondary side volumes into a single volume for use in **[[**

**]]**. For the events analyzed with this model, the correct prediction of the heat transfer from the primary system side of the SG to the secondary system side is considered by the NRC staff to be important in determining the overall system transient response.

The licensee's response to RAI 1b (Reference 5) indicated that the primary-to-secondary heat transfer depends heavily on the elevation of the mixture level relative to that of the SG u-tube bundle. When the mixture level is above the tube bundle, the primary-to-secondary heat transfer is primarily driven by nucleate boiling in the SG secondary side, and the heat transfer is relatively insensitive to the local fluid conditions. When the mixture level is within the tube bundle, the heat transfer becomes a strong function of elevation (nucleate boiling below the mixture level, single phase forced convection to vapor above it). As such, the NRC staff's concern regarding the capability of the single-volume model to capture the appropriate heat transfer is most relevant to transients where there would be substantial periods of significant u-tube bundle uncovery. For the transients analyzed with this model, the licensee stated that the demonstration analyses for the SLB transient provided in DPC-NE-3009 showed that the water inventory on the SG secondary remains above 100 percent of the full power inventory at the time of maximum core heat flux. The licensee further stated that based on these results and engineering judgment, there is no expectation of significant u-tube bundle uncovery in the period of interest for the other events analyzed with this model (increase in feedwater flow, uncontrolled rod control cluster assembly (RCCA) bank withdrawal from subcritical or low power startup condition). Because the SLB is limiting from a SG inventory standpoint, the NRC staff agrees. The NRC staff thus determined that the simplified model is acceptable for these events.

The licensee's response to RAI 1b also proposed an additional usage of the simplified SG model for the loss of normal feedwater flow event in some circumstances, consistent with a model that was adopted in Revision 3 of DPC-NE-3002 (Reference 33).[4] In this report, the base SG model was found to result in an over-prediction of the primary-to-secondary heat transfer when the SG water inventory dropped below 10 percent of the full-power inventory (i.e., there was significant SG u-tube uncovery) following the reactor trip. This is non-conservative for heatup events like the loss of normal feedwater. Thus, the licensee proposed to use a simplified SG model for the post-trip phase of the loss of normal feedwater flow analysis if the minimum post-trip SG water inventory indicates significant tube bundle uncovery. Because this is consistent with the NRC staff's approval of the model in DPC-NE-3002, Revision 3, the NRC staff finds this usage of the simplified SG secondary model acceptable. Accordingly, the NRC staff considers RAI 1b resolved.

[4] DPC-NE-3002, Revision 4b in the reference contains all information pertinent to the simplified SG model from DPC-NE-3002, Revision 3.

3.2.2.3 Transient-Specific Code Options and Correlations from DPC-NE-3009

3.2.2.3.1 Two-Region Non-Equilibrium Volume Model

One model available in RETRAN-3D allows a volume to be represented as a two-region non-equilibrium volume. This model is planned for use in the pressurizer, per DPC-NE-3008. It will also be used **[[**

**]]**, as discussed in Section 5.0 of DPC-NE-3009. The two-region non-equilibrium volume model has been previously approved for these applications in DPC-NE-3000. Since the differences between Catawba/McGuire and Harris/Robinson are such that the use of this model would not be significantly different, the NRC staff therefore finds its use acceptable in DPC-NE-3009.

3.2.2.3.2 Inter-Region Heat Transfer Model

The inter-region heat transfer model evaluates the interfacial heat transfer between the liquid and vapor regions of a two-region non-equilibrium volume (such as will be used in the pressurizer and in the upper head under some circumstances). While the base models **[[**

**]]**.

As the licensee noted in DPC-NE-3000, the interphase heat transfer coefficient is a user input. Condition 18 on the NRC staff's use of RETRAN-3D requires user-supplied parameters for use in the pressurizer model to be justified. In response to RAI 2 (Reference 5), the licensee provided **[[**

**]]**.

The licensee's RAI response also indicated that in some circumstances, **[[**

**]]**.

The licensee also proposed to optionally model the heat transfer coefficient based on plant data using the method described in DPC-NE-3000. This is acceptable based on the NRC staff's previous approval of this approach. Accordingly, the NRC staff considers RAI 2 resolved.

3.2.2.3.3 Non-Conducting Heat Exchangers

The non-conducting heat exchanger model allows energy to be transferred to or from a fluid volume without using a conductor. One application of this model is in simulating the pressurizer heaters; this application is consistent with previously approved models for McGuire and Catawba documented in DPC-NE-3000 and is thus acceptable. Other applications were proposed by the licensee in response to RAI 3 (Reference 4). These include the potential for modeling feedwater heaters for the SLB event, with heat addition rates based on plant design data with conservative adjustments to bound expected operation, and to artificially remove the heat added to the RCS by the reactor coolant pumps (RCPs) to simplify the steady-state initialization for transients initiated from a HZP condition. Because the former application will be based on design data with a conservative bias, and the latter is only used for convenience, the NRC staff finds them to be acceptable.

The model may also be used to simulate ambient heat losses. As discussed in the licensee's response to RAI 3, **[[**

**]]**. The NRC staff determined that this approach will adequately model the ambient heat losses, with appropriate conservative biases, and therefore found it to be acceptable. Accordingly, the NRC staff considers RAI 3 resolved.

3.2.2.3.4 Local Conditions Heat Transfer Model

In general, the local conditions heat transfer model is used to evaluate the wall heat transfer for bubble rise volumes. When a stack of heat conductors is positioned next to a bubble rise volume, this model evaluates the location of the liquid-vapor interface relative to the heat conductor midpoint elevation in order to determine the appropriate wall heat transfer correlation. The model is used in the pressurizer, and is also applied **[[**

**]]**. Use of the local conditions heat transfer model in the pressurizer was previously approved by the NRC staff in the SE for DPC-NE-3000.

As discussed in the licensee's response to RAI 4 in (Reference 5), the local conditions heat transfer model was originally developed for use with the bubble rise volume model, which constrains the liquid and vapor regions of the volume to the saturation temperature. However, it was subsequently extended for use with the two-region non-equilibrium volume model, which constrains neither the liquid nor the vapor region of the volume to the saturation temperature and allows a non-equilibrium temperature distribution to exist with a distinct interface between the regions and a distinct liquid level. The use of the local conditions heat transfer model with the two-region non-equilibrium volume model will therefore allow heat transfer between the fluid and the surrounding walls in **[[** **]]** to be appropriately characterized based on the local fluid conditions. This is especially important for **[[**

**]]**. The NRC staff therefore finds its use in these applications acceptable. Accordingly, the NRC staff considers RAI 4 regarding the local conditions heat transfer model resolved.

**[[**

**]].**

3.2.2.3.5 General Transport Model

The general transport model is used to model the transport of boron within the RCS. It is not used in the base model but is used to calculate boron concentration in the core resulting from emergency core cooling system (ECCS) injection during the SLB transient. The licensee also proposed that other events with ECCS injection may use this model to determine reactivity.

The NRC staff's SE on RETRAN-3D discusses Boron Transport in Section 4.1, stating:

> There are several models in RETRAN-3D to minimize numerical diffusion or provide front tracking for fluid temperature fronts: the method-of-characteristics, the transport delay model, and the enthalpy transport model. Each of these models is used in a particular circumstance as a user option. Boron transport is handled as a passive contaminant by the "general transport model" (Volume 1, Section VII-5.0 of Reference 4). This model uses a first order accurate upwind difference scheme with an implicit temporal differencing. This approach is highly diffusive, especially if the Courant limit is exceeded. This scheme can result in a front arrival that can be spread out over a long period and its amplitude reduced to about half that of the peak. Since RETRAN-3D has the same model as RETRAN-02 MOD003 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

The NRC staff concludes, based on the discussion in the RETRAN-3D SE, that the use of the general transport model to model boric acid transport is acceptable. However, the caveats expressed in the NRC's generic RETRAN-3D SE regarding the numerical scheme used for the general transport model and the Courant limit continue to be applicable. Because the present SE has not reviewed specific safety analyses, the licensee is responsible for ensuring that the RETRAN-3D SE conditions continue to be met for each plant-specific analysis.

3.2.2.3.6 Decay Heat Model

The base models provided in DPC-NE-3008 use the 1979 ANS standard for post-trip decay heat. However, for transients where post-trip decay heat is important, the post-trip decay heat will be biased high or low depending the limiting direction for the transient being analyzed. The conservatively low decay heat function will use or bound an assumed multiplier of 0.9 on the 1979 ANS decay heat curve (as was used in DPC-NE-3005), while the conservatively high decay heat function will meet or exceed the 1979 ANS decay heat curve plus two standard deviations (as was used in DPC-NE-3000).

In response to RAI 5 (Reference 4), the licensee stated that there are five transients for which the decay heat assumption is important. In these transients, the limiting conditions may occur substantially after the reactor trip. Table 1 below summarizes the decay heat biasing for these transients.

Table 1. Decay heat assumptions.

| Transient | Decay Heat Assumption |
|---|---|
| Steam System Piping Failure | Low, maximizes primary system overcooling |
| Loss of Non-Emergency alternating current (ac) Power to the Station Auxiliaries | High, maximizes post-trip primary system heatup |
| Loss of Normal Feedwater Flow | |
| Feedwater System Pipe Break | |
| SG Tube Rupture | For SG overfill evaluation, sensitivity studies will determine if high or low decay heat is limiting.<br><br>For the short-term core cooling analysis, high decay heat maximizes the post-trip primary system heatup. |

The decay heat multiplier is assumed to be 1.0 (i.e., the 1979 ANS decay heat curve is used directly) for all other transients, which were deemed to be insensitive to decay heat assumptions by the licensee. The NRC staff finds the biases proposed by the licensee to be acceptable for use in Harris and Robinson UFSAR Chapter 15 non-LOCA transient analyses conducted using the methods discussed in DPC-NE-3008 and DPC-NE-3009.

### 3.2.3 Compliance with RETRAN-3D Conditions and Limitations

RETRAN-3D has a large number of conditions and limitations imposed on its use by the NRC staff in the generic SE. There are 39 conditions and limitations in the NRC staff's SE on RETRAN-02. Six additional conditions and limitations were added in the NRC staff's SE on RETRAN-3D, bringing the total to 45.

A disposition of each of these limitations and conditions was provided in the response to RAI 13 (Reference 3). Many are dispositioned based on the prior position taken by the licensee in DPC-NE-3000. The NRC staff reviewed the disposition provided in the licensee's RAI response and found that all of the RETRAN-3D conditions and limitations were satisfied by the documentation provided in the combination of DPC-NE-3000, DPC-NE-3008, DPC-NE-3009, and the associated RAI responses.

The only condition on the use of RETRAN-3D that was not previously addressed in DPC-NE-3000 and is not addressed elsewhere in this SE is Condition 40. The licensee's disposition to this condition stated that several new RETRAN-3D control blocks that have not been previously reviewed and approved by the NRC staff are included in the Harris and Robinson base plant models. As discussed in the response to RAI 14 in (Reference 3), the new control blocks are "super blocks" that allow logic to be significantly simplified. The NRC staff finds the use of these blocks to be acceptable. Accordingly, the NRC staff considers RAI 14 resolved.

## 3.3 VIPRE-01 Correlations and Models

### 3.3.1 Critical Heat Flux Correlations

In DPC-NE-3009, the licensee proposed to use the High Temperature Performance (HTP) correlation for DNBR evaluations, as discussed in additional detail in DPC-NE-2005 (Reference 26). However, it is possible that the limiting DNB statepoint will fall out of the approved range of the HTP correlation parameters during a transient, usually at low pressures. As such, the licensee proposed the use of three alternate correlations, each with its own DNBR limit: the W-3S correlation, the Modified Barnett correlation, and the EPRI-1 correlation.

#### 3.3.1.1 W-3S

The W-3S correlation has been widely used in licensing applications at the NRC, and is known to provide fairly conservative predictions. It is acceptable to the NRC staff.

#### 3.3.1.2 Modified Barnett

The Modified Barnett[5] correlation described in DPC-NE-2015 (Reference 35), has been approved for use at Oconee by the NRC staff (Reference 36), as discussed in the licensee's response to RAI 6 (Reference 4). The Modified Barnett correlation has also been approved for AREVA's transient analysis methodology (Reference 37). The NRC staff therefore determined that the use of the Modified Barnett correlation is acceptable for use within DPC-NE-3009, but notes that the fuel currently in use at Harris falls outside of the range of applicability of the correlation.

#### 3.3.1.3 EPRI-1

The EPRI-1 correlation was developed by the EPRI in 1983 to be a generic CHF correlation based on rod bundle data from Columbia University, and was made for use with the COBRA-IIIC subchannel code. The database covers a wide range of conditions and was developed using data from a variety of fuel bundles to provide extended applicability.

Though the EPRI-1 correlation has been used in various applications both for the NRC and elsewhere in the industry, it has not received explicit NRC approval. Section 2.3 of the NRC staff's SE on the use of VIPRE-01 states that transient tests from the Thermal Hydraulic Test Facility at Oak Ridge National Lab were conservatively predicted using EPRI-1, but did not provide any statement that the correlation was approved for general use. EPRI-1 is also the default CHF correlation of recent versions of the NRC's FRAPTRAN transient fuel thermal-mechanical analysis code, replacing the W-3 correlation.

Because the EPRI-1 correlation has an extensive validation database, is in wide use in the industry as a general-purpose CHF correlation, and has been found to provide acceptable results in applications by the NRC, the NRC staff finds it acceptable for use at Harris and Robinson for safety analyses. However, the NRC staff's approval of EPRI-1 for use in DPC-NE-3009 is subject to the condition that, because of the general and wide-ranging database used to develop the correlation, it may not be used if a CHF correlation specifically

[5] The "Modified Barnett" correlation is also known as the Hughes correlation. This is the name under which it is listed as an acceptable correlation for use in LOCA analyses in 10 CFR 50 Appendix K.

developed for the fuel in use at the plant is applicable to the conditions being modeled (e.g., in the case of DPC-NE-3009 for Harris and Robinson, the HTP correlation must be used instead within its range of applicability).

3.3.2 Subcooled and Bulk Voiding Models

The licensee normally uses VIPRE-01 with the EPRI correlations for subcooled and bulk voiding, per the NRC-approved models discussed in DPC-NE-2005, Revision 5. However, those models have restrictions on time step size imposed by the NRC's SE on the use of VIPRE-01 in Condition 4, which states:

> If a profile fit subcooled boiling model (such as Levy and EPRI models) which was developed based on steady-state data is used in boiling transients, care should be taken in the time step size used for transient analysis to avoid the Courant number less than 1.

This is because subcooled and sometimes bulk void correlations may make the homogeneous flow solution numerically unstable at small timesteps. To avoid this condition in certain transients (particularly the CRE analysis, which may need to use a very small timestep to address the power response), the licensee initially proposed to use **[[**

**]]**.

The NRC staff therefore issued RAI 7, which asked the licensee to provide further justification for the use of **[[**

**]]**.

Such conditions would potentially result in incorrect predictions of the local void fraction, thus impacting the calculated MDNBR. In response to RAI 7 (Reference 5), the licensee proposed to use **[[**

**]]**.

Based on the licensee's RAI response, the NRC staff determined that the proposed method for determining the appropriate subcooled and bulk voiding models is acceptable. It will avoid the numerical instability associated with the use of subcooled boiling models based on steady-state data by using appropriate time step sizes, and, **[[**

**]]**.

Accordingly, the NRC staff considers RAI 7 resolved.

3.3.3 VIPRE-01 Models for Special Applications

VIPRE-01 models are constructed from an array of quasi-one-dimensional subchannels, where flow is primarily in the axial direction. Fuel rods, instrument tubes, and control rod guide tubes

are then associated with each subchannel as appropriate. A subchannel may represent a true subchannel from a fuel assembly in the core, or it may represent any number of lumped subchannels; the same is true for fuel rods. Usually, VIPRE-01 models provide detailed subchannel nodalization in a specific area of interest (the area in the hot assembly around the hot subchannel or hot rod), and progress towards a more lumped representation farther from the area of interest.

3.3.3.1 VIPRE-01 Model for Steam Line Break Analysis (DPC-NE-3009)

As discussed in Section 3.3 and shown in Figure 3-3 of DPC-NE-3009, the licensee created a specific VIPRE-01 model to analyze the MDNBR for SLBs. This model includes **[[**

**]]**. The overall approach is very similar to the one taken in the NRC-approved DPC-NE-3001, with differences in the specific nodalization that are a result of moving from a four-loop NSSS to a three-loop NSSS. As such, the NRC staff finds the use of the **[[** **]]** VIPRE-01 model for the SLB analysis to be acceptable.

3.3.3.2 VIPRE-01 Model for Control Rod Ejection Analysis (DPC-NE-3009)

As discussed in Section 3.3 of DPC-NE-3009, the licensee proposed the use of **[[**

**]]** for the CRE accident. Details of the analysis method were provided in Section 5.4.8.2 of DPC-NE-3009. The model will be evaluated in Section 3.5.21.2.2 of this SE.

3.3.3.3 Expanded VIPRE-01 Models (DPC-NE-3008)

In Sections 5.1 and 5.2, the licensee proposed expanded VIPRE-01 models for Harris and Robinson. In these models, **[[**

**]]**. The remainder of each adjacent assembly is modeled as one lumped subchannel per assembly, and the rest of the core is modeled as a single lumped subchannel. The exact nodalization depends on the plant, because Harris uses 17x17 fuel assemblies and Robinson uses 15x15 fuel assemblies.

The licensee's response to RAI 22 (Reference 3) discussed the benefits and some possible uses of these models. The detailed geometric representation of **[[**

**]]**, since it typically results in a gain of DNB margin. However, it also allows the **[[** **]]** to be explicitly modeled, which can also result in margin gains. Additionally, the detailed modeling allows flatter assembly power distributions based on the cycle-specific pin-power distribution from SIMULATE-3 to be explicitly evaluated. The detailed VIPRE-01 model can then be used to assess potential DNB impacts of fuel assembly design features (such as burnable poison loading patterns) and potential core issues (such as fuel assembly bow). Because a more realistic pin-power distribution can be used in the expanded model when compared to the generic reference distribution used in the **[[** **]]** model, it can provide a benefit in cycle-specific DNB evaluations. Another

potential usage is in calculating mixed core DNB penalties; however, the licensee stated in the RAI response that such a methodology would be submitted separately for NRC review and approval.

Pin-power distributions for input to the expanded VIPRE-01 models are made more conservative using the method discussed in the licensee's response to RAI-22b (Reference 3). In this method, the difference in peaking between each rod in the assembly and the rod with the maximum pin peaking factor is calculated from the SIMULATE-3 pin power distribution. The licensee refers to this as the "delta-power" value. A conservative multiplier is applied to each of these delta-power values to reduce them; the pinwise reduced delta-power values are then subtracted from the maximum peaking factor in the assembly, and the resulting pin peaking factors are applied to each pin to create a revised power distribution. This power distribution is both flatter and inputs more energy to the assembly. This is the same method used in DPC-NE-3000, Appendix E, for the Oconee expanded VIPRE-01 model, which has previously been found acceptable by the NRC staff.

As discussed in the response to RAI 22c (Reference 3), the licensee does not plan to use the expanded models to generate the statistical DNB limits using the statistical core design (SCD) methodology. Instead, the licensee posits that the SCD limits generated using the **[[ ]]** model from DPC-NE-2005 could be used with the expanded models. The licensee's justification for this is that the expanded models only add increased radial definition, but do not fundamentally change the statistical DNB behavior of the fuel design. The NRC staff accepts this justification, because the statistical DNB performance is mostly reliant on thermal-hydraulic statepoint uncertainty and CHF correlation uncertainty, neither of which is affected by the radial nodalization used.

The licensee's response to RAI 22d (Reference 3) addressed radial and axial nodalization sensitivity. As discussed in the RAI response, the licensee's analyses have confirmed the VIPRE-01 code manual's sensitivity studies that document (a) that one row of subchannels adjacent to the hot subchannel is sufficient to resolve the details of the flow field in the hot assembly and (b) if this level of detail is provided the flow in the hot subchannel is not sensitive to the nodalization of the rest of the core. No specific radial nodalization sensitivity study was performed for the expanded VIPRE-01 models because much more detail is included than that indicated by previous sensitivity studies. Previous sensitivity studies have also shown that, once sufficient axial nodal resolution is obtained, the VIPRE-01 prediction is insensitive to further refinement. The axial nodalization used is therefore the same as that used in the NRC-approved DPC-NE-2005 models. Because of the results of the past sensitivity studies, the NRC staff finds the licensee's justification that the expanded VIPRE-01 model results will be insensitive to axial and radial nodalization acceptable.

In response to RAI 22e (Reference 3), the licensee stated that the turbulent momentum factor, which controls how much momentum is mixed between adjacent subchannels, will be set to **[[ ]]**, consistent with the regular **[[ ]]** models from DPC-NE-2005. This means that **[[ ]]**. The licensee stated that this is conservative. The turbulent crossflow itself is modeled in VIPRE-01 as the product of the subchannel gap width, the average of the flow in the adjacent subchannels, and a turbulent mixing coefficient. The licensee stated in the RAI response that the turbulent mixing coefficient is a fuel-assembly dependent value (provided by the fuel vendor) that is only applied to adjacent subchannels within the same assembly. The turbulent mixing coefficient is set to zero in the subchannels between the assembly boundaries. The licensee

also stated that the use of turbulent mixing coefficients in expanded models for mixed-core applications will be submitted for NRC review and approval. Because the model **[[ ]]**, and because it uses the vendor-supplied value for intra-assembly turbulent crossflow and neglects inter-assembly turbulent crossflow, the NRC staff finds the treatment of turbulent crossflow to be acceptable.

The licensee benchmarked the expanded models against the Harris and Robinson **[[ ]]** models from DPC-NE-2005. For each plant, the expanded model and the **[[ ]]** model were applied to the same set of transient statepoints with the same set of code options, correlations, and parameters. The benchmarks are provided in Tables RAI-22-1 and RAI-22-2 (Reference 3). The transients analyzed include: increase in steam flow, loss of load, loss of forced reactor coolant flow, uncontrolled RCCA bank withdrawal from a variety of initial conditions and times in life, statically misaligned RCCA, dropped RCCA, and inadvertent opening of a pressurizer PORV. As expected, the expanded model always predicted a higher (less conservative) MDNBR, **[[ ]]**.

The NRC staff therefore determined that this benchmark demonstrated that the results from the expanded models were consistent with but slightly less conservative than the NRC-approved **[[ ]]**. Based on this consistency and the modeling choices that were justified by the licensee and discussed elsewhere in this section of the SE, the NRC staff finds the use of the expanded models in licensing applications to be acceptable. As stated by the licensee, the expanded models are not to be used in licensing applications to analyze mixed cores; this possible usage of the expanded models would need to be addressed in a separate report for the NRC's review and approval. Accordingly, based on the foregoing, the NRC staff considers RAI 22 (on DPC-NE-3008), including each of its subparts, resolved.

## 3.4 Safety Analysis Physics Parameters

The licensee's overall safety analysis philosophy is to perform a set of bounding analyses once, then check each cycle to ensure that the core design remains bounded by the analysis. The licensee stated that though the importance of physics parameters and their impact on the transient response varies from transient to transient, it is possible to identify the set of core physics parameters that affect a given transient's result. The behavior of each transient and the impact of the core physics parameters on that transient is explored through sensitivity studies. Once the behavior is understood, a bounding direction can be identified for each parameter or each transient, allowing bounding values to be selected.

Core physics parameters are calculated each cycle using an NRC-approved methodology (such as the CASMO-5/SIMULATE-3 methodology recently approved by the NRC for Harris and Robinson, as described in DPC-NE-1008 (Reference 21)), to verify that the values calculated for the reload core are bounded by the safety analysis. The licensee proposed that this verification could take one of two forms:

- Physics parameters may be explicitly calculated for the reload core and compared to the licensing safety analysis parameters, or
- When the core is similar to a previous reload core, the bounding values from the analysis may be compared to values from the similar previous reload design.

The licensee envisions this second option as being useful, for example, for a limited core redesign necessitated by fuel assembly damage that occurred during the reload. The licensee justified this option as acceptable because most core physics parameters have predictable behavior as a function of operating parameters (burnup, reactor power, moderator temperature, boron concentration, etc.). In response to RAI 8 (Reference 4), the licensee further clarified that a core design is considered to be "similar" to a reference design if **[[**

**]]**.

The first of these two options is standard practice in the nuclear industry and is therefore acceptable to the NRC staff. Given the constraints on the second option proposed by the licensee in the RAI response, and because **[[** **]]**, the NRC staff finds it acceptable.

The values of key parameters provided in Table 4-1 of DPC-NE-3009 are evaluated, according to the above methods, each reload cycle to ensure that the values assumed in the current licensing analysis bound each core. For high power level or high peaking factor events, the fuel thermal limits (i.e., DNBR and centerline fuel melt (CFM) limits) are also evaluated to ensure that accident analysis acceptance criteria are met and protection limits remain acceptable. The relevant accident analysis acceptance criteria will be discussed when discussing each transient in Section 3.5 of this SE. The acceptability of the fuel thermal limits is evaluated using 3D core power distributions from SIMULATE-3, including rod positions and xenon distributions allowed by the TS rod insertion and axial flux difference (AFD) limits. Key core thermal-hydraulic and system conditions from the limiting RETRAN-3D statepoint are also considered.

If the safety analysis physics values are exceeded or the fuel thermal limits are violated, the licensee proposed to either perform a cycle-specific evaluation or analysis to determine the acceptability of the core design, or else will redesign the core. This is standard practice in the industry and is acceptable to the NRC staff.

### 3.4.1 Key Parameter Selections

The key parameters selected for inclusion in the licensee's cycle-specific evaluations are included in Table 4-1 of DPC-NE-3009. The NRC staff reviewed the parameter selection and found it to be consistent with previously approved Duke Energy methodologies (e.g., DPC-NE-3001) and other, similar methodologies from fuel vendors (e.g., Westinghouse's reload SE methodology documented in WCAP-9272 (Reference 38). The NRC staff therefore consider the safety analysis physics parameters to be sufficient for determining whether the neutronic behavior of a given core reload design is bounded by the safety analysis.

### 3.4.2 Key Parameter Methods

This section of the SE presents the NRC staff's review of the licensee's methods for determining each of the key safety analysis parameters.

3.4.2.1 Reactivity Insertion following Reactor Trip

This "parameter" includes both the minimum available tripped rod worth and the normalized reactivity insertion rate during a reactor trip. The minimum available tripped worth calculation assumes that the most reactive control rod is fully withdrawn from the core and that the remaining control rods drop from the power-dependent rod insertion limit (RIL). It is calculated to ensure that the TS shutdown margin is preserved. The normalized reactivity insertion rate is determined using **[[ ]]**, combined with a conservative relationship between rod position and normalized reactivity worth. This relationship, as discussed in the responses to RAIs 9 (Reference 4) and 15 (Reference 3), is calculated at hot full power (HFP) conditions assuming a bottom-peaked core power distribution and that the highest worth rod remains stuck out of the core. Inserted worth and rod position are normalized to produce a normalized rod worth versus position curve, which is multiplied by the minimum trip worth to obtain a trip reactivity versus rod position curve. The NRC staff determined based on the licensee's description that the reactivity insertion curve developed using this method will be appropriately limiting, and is therefore acceptable. Accordingly, the NRC staff considers RAIs 9 and 15 resolved.

3.4.2.2 Initial Power Distribution

The initial power distribution for transients is bounded by what is allowed by the power-dependent AFD limits and RILs. These limits ensure that the heat flux hot channel factor ($F_q$) and nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) limits are not exceeded. As discussed in response to RAI 10 (Reference 4), the initial power distribution also directly impacts the transient response for several transients by influencing the amount of reactivity available for withdrawal and rate of reactivity insertion due to control rod movements. Restricting transient initial power distributions to those allowed by the AFD and RILs was approved by the NRC staff and is discussed in more detail in DPC-NF-2010, Revision 3 (Reference 39) and DPC-NE-2011, Revision 2 (Reference 40).[6] Accordingly, the NRC staff considers RAI 10 resolved.

3.4.2.3 Power Distribution at the Limiting Transient Statepoint

If the power distribution changes over the course of the transient, the power distribution at the limiting transient statepoint (usually the time of MDNBR or maximum thermal power, but whatever statepoint is most limiting with respect to the acceptance criteria for each transient) must also be evaluated with respect to the safety analysis. Such changes in power distribution may be driven by control rod motion, moderator temperature changes, and so on. Because the power distribution will be explicitly evaluated at the limiting transient statepoint, the NRC staff found the method to determine this parameter to be acceptable.

3.4.2.4 Effective Delayed Neutron Fractions and Decay Constants

The key parameter is the total effective delayed neutron fraction, $\beta_{eff}$. The licensee proposed the use of time-in-life values with a high or low bias, depending on the transient. Additionally, the licensee also proposed to specify values for the six-group effective delayed neutron fractions and their associated decay constants at the beginning and the end of the cycle, though

[6] Hereafter, all references to DPC-NF-2010 are to Revision 3 and all references to DPC-NE-2011 are to Revision 2.

these values are unbiased (unlike $\beta_{eff}$). Because the bias included in $\beta_{eff}$ will appropriately bound expected conditions and will be appropriately selected for each transient, the NRC staff finds the determination of this parameter to be acceptable.

3.4.2.5 Prompt Neutron Lifetime

The licensee proposed to specify the prompt neutron lifetime based on typical time-in-life values, except, per the response to RAI 11 (Reference 4), **[[**

**]]**. The use of typical values for prompt neutron lifetimes except **[[**
**]]** is consistent with previously approved Duke Energy methodologies and is thus acceptable to the NRC staff. Accordingly, the NRC staff considers RAI 11 resolved.

3.4.2.6 Initial Fuel Temperature

Core average fuel temperatures are calculated using the three-dimensional core model in SIMULATE-3, and are dependent on time in life and thermal power level. As discussed in DPC-NE-1008 (Reference 21), the SIMULATE-3 fuel temperature dependence on power and burnup is characterized using SIMULATE-3K's fuel rod thermal model. The SIMULATE-3K methodology report, SSP-98/13, Revision 6, "SIMULATE-3K Models and Methodology" (Reference 41), states that this model **[[**

**]]**

The licensee stated that the fuel temperature values from SIMULATE-3 are accurate because accurate prediction of fuel temperature is critical to accurate reactivity predictions. Because the core model reactivity and power distribution predictions are continually checked against plant measurements, this provides assurance that the fuel temperatures are being accurately modeled. Local, hot spot fuel temperatures are calculated using vendor fuel performance codes that are applicable to the fuel in the core and take into account the initial power distribution and appropriate hot channel factors.

The NRC staff agrees that the core SIMULATE-3 model should be adequate for determining initial fuel temperatures, for the reasons discussed by the licensee. Additionally, the correlations used to characterize the burnup and temperature dependence of the material properties are well known to the NRC staff and appropriately include the effects of thermal conductivity degradation (TCD). The NRC staff also determined that it is appropriate to calculate the local, transient fuel temperature using the vendor fuel performance codes applicable to the fuel type(s) in the core. The treatment of initial fuel temperature by the licensee was therefore determined to be acceptable.

3.4.2.7 Shutdown Margin

The shutdown margin is determined by calculating the difference in reactivity between an all-rods-in (ARI) condition and an all-rods-out (ARO) condition, and then adjusting for power defect, highest worth stuck rod, control rods inserted to the power-dependent RILs, transient xenon, and the control rod worth uncertainty. Shutdown margin is calculated at beginning of

cycle (BOC) and end of cycle (EOC) conditions for each reload core at a variety of power levels, including (but not limited to) HFP and HZP. This is consistent with the NRC-approved DPC-NE-3001 and is thus acceptable.

3.4.2.8 Trip Reactivity

Trip reactivity is defined as the amount of negative reactivity inserted into the core following a reactor trip. It is evaluated for each reload core at HFP, assuming the highest worth rod is stuck in the core. For transients that assume the minimum normalized trip reactivity shape, negative reactivity insertion is conservatively delayed by assuming the reactor is operating with a bottom-skewed power distribution before the trip. This is consistent with the NRC-approved DPC-NE-3001 and is thus acceptable.

3.4.2.9 Rod Insertion Limits

It is the NRC staff's understanding from a review of other Duke Energy methodologies (especially that of DPC-NF-2010 and DPC-NE-2011) that the licensee's control RILs are defined somewhat arbitrarily and determined to be acceptable if they result in acceptable safety analysis results. This means that, in order to be deemed appropriate, a set of RILs must provide acceptable power peaking analyses, shutdown margin calculations, ejected rod worth calculations, and reactivity insertion assumptions for the safety analysis. Acceptability of RILs from a peaking standpoint is performed when setting the operating limits and reactor protection system (RPS) setpoints for a reload core design (as is done in Duke Energy methodology report DPC-NE-2011). RILs are calculated at a variety of power levels and burnups and may be dependent on either or both. The calculation of RILs is consistent with other approved Duke Energy topical reports (most notably DPC-NE-3001 and DPC-NE-2011) and is thus acceptable.

3.4.2.10 Maximum Differential Rod Worth from Subcritical

The maximum differential rod worth from subcritical is determined from the maximum amount of reactivity that can be withdrawn from the reactor by moving control rods starting from a subcritical condition. The calculation assumes that two sequential control banks are moving in 100 percent overlap with the reactor at HZP, and is calculated at BOC, EOC, and an intermediate burnup. It is performed to ensure that the inputs to the uncontrolled RCCA bank withdrawal from subcritical/low power transient are bounded.

In general, the licensee calculates rod worth assuming adverse power distributions, which in turn are developed from adverse xenon distributions. The method for creating these adverse xenon and power distributions is discussed in the licensee's response to RAI 12 (Reference 4), and is consistent with other Duke Energy methodologies for generating a set of power distributions (such as that approved by the NRC staff in DPC-NE-2011). **[[**

**]]**, and at a variety of times in life. Because the method is consistent with other Duke Energy methodologies and determines the limiting rod worth from a set of evaluations that spans the plant operating space, the NRC staff determined that it is acceptable. Accordingly, the NRC staff considers RAI 12 resolved.

3.4.2.11 Maximum Differential Rod Worth at Power

The maximum differential rod worth at power is determined from the maximum amount of reactivity that can be withdrawn from the reactor by moving control rods when at power. It assumes two sequential control banks are moving in normal overlap, while adhering to the RILs, and is performed at BOC, EOC, and an intermediate burnup. The maximum differential rod worth at power calculation is performed to ensure that inputs to the uncontrolled RCCA bank withdrawal at power (URBWAP) are bounded.

The maximum differential rod worth at power calculations are performed using the same assumptions as discussed in Section 3.4.2.10 above and are therefore acceptable.

3.4.2.12 Dropped Rod Worth

The worth of a dropped rod is determined by calculating the difference in reactivity between a core with the rod inserted fully and the rod inserted at either the HFP/ARO condition or the RIL. Possible dropped rod combinations are evaluated to determine limiting dropped rod worth values at BOC, EOC, and an intermediate burnup. These values are then compared to safety analysis values to ensure that the safety analysis remains bounding. This is consistent with the NRC-approved DPC-NE-3001 and is thus acceptable.

3.4.2.13 Available Worth for Withdrawal

The worth available for withdrawal is determined by determining the reactivity difference between a core with the controlling bank inserted to the RIL and a core with the control rods at ARO conditions. It is calculated at BOC, EOC, and an intermediate burnup. While this parameter was not discussed in other Duke Energy analysis methodologies, it is consistent with the approach used for other control rod worth parameters discussed in this section and is considered by the NRC staff to be acceptable.

3.4.2.14 Ejected Rod Worth

To determine the maximum ejected rod worth, core reactivity calculations are first performed assuming that control rods are at the RIL with a top-skewed power distribution. The axial power distributions are determined using the assumptions discussed above in Section 3.4.2.10. The control rod is then ejected from the RIL to the ARO position and the reactivity difference is calculated, with conservatism maintained by holding both the fuel and moderator temperature distributions constant. The calculation is performed at combinations of BOC/EOC and HFP/HZP.

Unlike previously approved Duke Energy methodologies for determining ejected rod worth, such as that described in DPC-NE-3001, DPC-NE-3009 states that only certain limiting control rod locations would be evaluated. In response to RAI 13 (Reference 4), the licensee clarified that rods that are known to be non-limiting prior to the analysis due to their power-dependent RILs, such as those from control bank B, would be excluded from the analysis. The initial analysis would then consider all rods in control banks D and C (assuming core symmetry), and subsequent evaluations would potentially only consider only a subset that are known to be limiting based on experience. The NRC staff finds this approach acceptable, but notes that significant changes in fuel management or core design strategy may necessitate a re-evaluation

of the potentially limiting control rod locations to ensure that the limiting ejected rod worth is captured. Accordingly, the NRC staff considers RAI 13 resolved.

3.4.2.15 Stuck Rod Worth

The stuck rod worth is determined by calculating the reactivity difference between the ARI condition and the condition with the stuck control rod positioned in the fully withdrawn position. Different stuck rod configurations are evaluated, assuming core symmetry, to find the highest worth stuck rod at BOC/HZP and EOC/HZP conditions. Both location and worth are recorded. While this parameter was not discussed in other Duke Energy analysis methodologies, it is consistent with the approach used for other control rod worth parameters discussed in this section and is considered by the NRC staff to be acceptable.

3.4.2.16 Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) is the change in reactivity resulting from a change in moderator temperature. It is influenced by a variety of conditions, including control rod position, cycle exposure, moderator pressure and temperature, and soluble boron concentration. These conditions are considered when calculating the MTC, which is done by inducing a change in moderator temperature and dividing the resulting predicted reactivity change by the temperature difference. Most positive MTC and most negative MTC are calculated and compared to safety analysis values. This is consistent with the NRC-approved DPC-NE-3001. The MTC calculation presented in DPC-NE-3009 considers moderator pressure in addition to temperature, which is reasonable because the moderator pressure may have a minor impact on the MTC, depending on the conditions in the RCS. The NRC staff therefore determined that it is acceptable.

3.4.2.17 Doppler Temperature Coefficient

The Doppler temperature coefficient (DTC) is the change in reactivity resulting from a change in fuel temperature. To determine the DTC, two cases are run which vary the fuel temperature about a mean value and then calculate the DTC as the reactivity difference divided by the temperature difference. Least and most negative DTC values for each core are calculated considering core burnup and power level. This is consistent with the NRC-approved DPC-NE-3001 and is thus acceptable.

3.4.2.18 Boron Concentration and Differential Boron Worth

Critical boron concentration and shutdown boron concentration are calculated as functions of reactor power, cycle exposure, reactor coolant temperature, and control rod position (as allowed by the RILs). Differential boron worth is calculated for various combinations of these variables. This is consistent with the NRC-approved DPC-NE-3001 and is thus acceptable.

3.5 Transient Analysis Methods

Methods for analysis for each of the UFSAR Chapter 15 transients were provided in Section 5.0 of DPC-NE-3009. Additionally, benchmark analyses for certain transients were provided in Section 4.3 of DPC-NE-3008, and demonstration analyses were provided in Section 6 of DPC-NE-3009. The primary difference between the two sets of analyses is that the benchmark

analyses from DPC-NE-3008 attempt to match the inputs from the analysis of record (AOR) as closely as possible, while the demonstration analyses from DPC-NE-3009 use the transient analysis methods discussed in that report. These analyses, used by the licensee to support the adequacy of the proposed models and methods, are discussed in this section of the SE.

Several generic considerations regarding the transient analysis methods are discussed below, and are followed by subsections representing each of the UFSAR Chapter 15 transients proposed for analysis by the licensee using the methods in DPC-NE-3008 and DPC-NE-3009.

Assumptions Regarding Plant Protection and Mitigating Systems

The DPC-NE-3009 methodology report has a section for each UFSAR Chapter 15 transient which lists the RPS trip signals and other plant design features that mitigate each transient. However, the licensee also stated in the methodology report that "systems referenced in mitigating event consequences are not comprehensive and other functions or signals may be credited where justified." While the licensee has made an attempt to list the applicable reactor trip signals for each transient, it is potentially possible that other signals may provide protection depending on the exact plant conditions during the transient (for instance, if an evaluated parameter has a slight impact on the transient's trajectory). The NRC staff therefore finds it acceptable to credit reactor trip signals that were not discussed in DPC-NE-3008 and DPC-NE-3009 for all the transients evaluated using these methodologies, provided that the trip signal is valid and appropriate instrumentation delays are included.

However, the NRC staff did not review the ability for mitigating systems beyond those explicitly discussed in the methodology report to impact the plant system response to transients. No implicit or explicit approval for credit of such systems may be presumed by the NRC staff's approval of this methodology report. As such, the NRC staff notes that credit for plant mitigating systems beyond those explicitly discussed in the methodology must be evaluated by the licensee to determine whether prior NRC approval is necessary.

Use of the Duke Energy Statistical Core Design Methodology

The licensee has developed a SCD methodology in DPC-NE-2005, which has been proposed for use within the DPC-NE-3009 methodology for determining the MDNBR for the UFSAR Chapter 15 non-LOCA transients. This methodology has been approved by the NRC staff for Harris and Robinson (Reference 27). The SCD methodology essentially allows the licensee to combine various plant-, fuel-, and code-related uncertainties into a statistical design limit for the DNBR. The thermal-hydraulic calculation for the plant may then be performed at nominal conditions and evaluated against the statistical DNBR limit (known as the SDL).

One caveat on the use of the SCD methodology is that the conditions where DNBR is calculated must fall within the range of statepoints considered when developing the SDL. For situations where one or more of the thermal-hydraulic boundary conditions falls outside of the range of statepoints used in defining the SDL, the licensee proposed to use the nominal correlation limit and to bias the boundary conditions conservatively to account for uncertainty–this is termed a "non-SCD" analysis in the methodology report. Because this is the standard method for evaluating the DNBR against its limit, this approach is acceptable.

The licensee also proposed that the SCD could still be used when the thermal-hydraulic statepoint being evaluated falls outside of those used in developing the SDL. This would be

accomplished by adding new statepoints to the SCD analysis (to bound the evaluated statepoint) and evaluating the SDL for those statepoints. As discussed in the response to RAI 15a in (Reference 4), the licensee's method in DPC-NE-2005 allows a new statepoint to be added to the SCD analysis without prior NRC approval if the statistical DNBR evaluation at the new statepoint does not exceed the existing SDL for the correlation/plant model. Because this is a feature of the NRC-approved DPC-NE-2005 methodology, the NRC staff finds this approach to be acceptable for use within the transient analysis performed using DPC-NE-3009.

The licensee proposed in DPC-NE-3009 that the selection of a particular transient analysis as SCD or non-SCD in the methodology may not match the final selection used in plant safety analyses, because the actual results of the analysis may indicate that the other method should be used. If events originally envisioned in the methodology as SCD transients fall outside of the SCD analysis range, the licensee's first course of action is to extend the SCD analysis to envelop the transient's limiting thermal-hydraulic statepoint, as discussed above. However, if this is not possible, these events will be performed as non-SCD analyses, with the state of the reactor system biased appropriately to maintain conservatism. As discussed in response to RAI 15b (Reference 4), initial core power will be biased high, reactor average temperature will be biased high, pressurizer pressure will be biased low, RCS flow will be biased low, and core bypass flow will be biased high. However, reactor coolant temperature and pressurizer pressure directly impact the OTΔT trip setpoint, and these biases may result in an earlier trip and a less limiting DNBR. Therefore, in transients where the mitigating trip is the OTΔT trip, the biases on pressurizer pressure and reactor average temperature will be evaluated to ensure that they are limiting. The affected transients are listed in the response to RAI 15b. The NRC staff determined that this approach is acceptable, because it will find the limiting DNBR condition with a consideration of the active mitigating trips. Accordingly, the NRC staff considers RAI 15 resolved.

Generic Key Parameter Selection and Biasing

Each transient discussed in Section 5 of DPC-NE-3009 has a table listing key plant parameters and the initial conditions assumed for these parameters in the analysis. Parameters are listed as being biased to one of the following conditions: SCD, low, high, evaluated, insensitive, zero, or N/A.

Analyses that are performed with the SCD methodology are initiated at nominal plant conditions. This means that power is chosen to be appropriate for the initial condition of the event, pressure is the plant reference pressure for power operation, the reactor coolant average temperature is the programmed value for the chosen power, and the RCS flow rate is at or below the TS minimum measured flow. In SCD analyses, flow uncertainties are included in the evaluation of the SDL and are therefore not included in the initial condition for flow. Core bypass flow is based on a best-estimate thermal-hydraulic analysis of the core bypass flow paths, if available; if not, a conservatively high core bypass flow will be assumed as discussed in the response to RAI 16c (Reference 4).

"Low" and "high" biases have slightly different meanings depending on the particular parameter, though in general they represent the decreases or increases, respectively, from nominal conditions needed to account for uncertainty. As discussed in the response to RAI 16a (Reference 4), the uncertainty used for a given parameter reflects the total uncertainty associated with the automatic control system for the parameter; if no control system exists, the indication uncertainty associated with the TS surveillance is used. Loop flow uncertainties

combine the calorimetric uncertainty with the loop RCS flow measurement uncertainty, as discussed in the response to RAI 16b (Reference 4). When SG tube plugging is biased high, it is assumed to be the maximum that can be supported by the analysis; when biased low, it is less than one percent. As discussed in response to RAI 16d (Reference 4), the low biased SG tube plugging value used in the DPC-NE-3009 RETRAN-3D base decks was based on the actual SG tube plugging values at Harris and Robinson.

For fuel temperature, which is a value calculated by the 3D core simulator (as discussed in Section 3.4.2.6 of this SE), the value in the analysis bounds the calculated values (either above or below, depending on the transient). For events initiated from zero power, the initial fuel temperature is the same as the moderator temperature. This is because the fuel is in thermal equilibrium with the coolant in a steady-state zero power condition. However, the gap conductance will still be biased in the conservative direction to provide a conservative transient response. This is acceptable to the NRC staff.

For some parameters in certain transient analyses, a bounding direction is not necessarily clear when specifying the initial and boundary conditions. To handle these within the generic methodology, the licensee has proposed to “evaluate” them, by performing sensitivity studies to determine which direction is most conservative. The licensee’s methodology also indicated that there potentially exist key parameters not listed in each Chapter 15 transient's section. The methodology states that, for these parameters, “the parameter selection should be justified in the internal analysis documentation.” The NRC staff approves of the approach of evaluating certain parameters to determine the limiting condition, since it assures the use of conservative initial and boundary conditions. Accordingly, based on the foregoing, the NRC staff considers RAI 16, including each of its subparts, resolved.

Generically Addressed non-LOCA Events and Analyses

In DPC-NE-3009, the licensee provided numerous statements that the response of a transient was bounded by another transient. Because SRP Section 15.0 instructs the reviewer to “evaluate licensees’ claims that individual AOOs [anticipated operational occurrences] and postulated accidents are limiting or nonlimiting, or bounded by other AOOs and postulated accidents,” the NRC staff issued RAI 14 (Reference 11). In response to RAI 14 (Reference 5), the licensee provided a table describing the events that are not applicable to Harris and Robinson (or could otherwise be addressed by a simplified evaluation). The NRC staff reviewed this table and determined that the licensee’s disposition of these events, most of which are boiling water reactor events that are not applicable to the pressurized water reactors at Harris and Robinson, was acceptable. Accordingly, the NRC staff considers RAI 14 regarding the generic analysis of events resolved.

The licensee’s response to RAI 14 additionally included a discussion of which transients are bounded by other transients and the underlying rationale. This discussion will be addressed for each transient in the sections that follow. Unless otherwise stated, the NRC staff finds the analysis methodologies to be applicable to both Harris and Robinson and may be used to evaluate the bounding nature of any event change.

### 3.5.1 Increase in Feedwater Flow

The increase in feedwater flow event is modeled as a failure of the MFW flow control valve to the open position, increasing the MFW flow. The DPC-NE-3009 methodology classifies it as an

ANS Condition II event for both Harris and Robinson, consistent with the plant UFSARs, and the corresponding acceptance criteria are applied. The acceptance criteria are to ensure that the MDNBR is above the 95/95 limit and that the CFM limit is not exceeded. The MDNBR is determined using the SCD methodology.

Peak primary and secondary pressure are considered to be bounded by the loss of external electrical load or turbine trip transients (depending on the plant) and are not analyzed. The increase in feedwater flow is an overcooling event that may result in an increase in power caused by moderator reactivity feedback. Though a power increase may occur, the secondary side heat sink remains available to remove heat from the primary system, unlike in the loss of load and turbine trip events, which are thus bounding with respect to pressure response. The NRC staff determined that this treatment of the primary and secondary pressure transients was therefore acceptable.

The DNB and CFM analyses are performed at combinations of BOC/EOC and HZP/HFP conditions, with rod control in both automatic and manual to identify the limiting case. RETRAN-3D is used for the systems analysis with the simplified SG secondary model discussed in Section 3.2.2.2 for transients initiated from HZP. The **[[ ]]** VIPRE-01 model is used to evaluate DNBR.

The initiating event in the RETRAN-3D model is a conservatively large step change increase in feedwater flow. Because feedwater heaters are not explicitly modeled in the base RETRAN-3D model, a step decrease in feedwater temperature is modeled coincident with the feedwater flow increase for transients initiated from HFP conditions. As discussed in the licensee's response to RAI 17 (Reference 4), the decrease in temperature is needed to bound the transient condition and is calculated based on an assumed constant heat addition rate from the feedwater heaters. This approach to modeling is acceptable to the NRC staff because it will bound the transient feedwater temperature and exacerbate the overcooling event. Accordingly, the NRC staff considers RAI 17 resolved.

The NRC staff determined that the licensee's approach to analyzing the increase in feedwater flow event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.1.1 Harris Increase in Feedwater Flow Benchmark (DPC-NE-3008)

The Harris increase in feedwater flow benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008, with minor modifications to the feedwater system connection to the SG to reduce flow instabilities that were observed in the process of performing the analysis. Inputs were otherwise selected to be consistent with the AOR, including the trip reactivity versus time curve as discussed in the licensee's response to RAI 15 (Reference 3). Though the AOR simulated the event to determine the MDNBR, the benchmark did not include a DNBR analysis (which would have required a separate VIPRE run).

In general, the timing of phenomena in the benchmark analysis is generally very close to the AOR, but temperature, reactivity, and pressure transients are not as severe. Heat is removed from the RCS slightly earlier in the benchmark than the AOR. Because a similar amount of energy is removed from the RCS over a slightly longer period of time, the affected loop $T_{cold}$ decreases less in the benchmark than the AOR. Because of this, the overall increase in $T_{avg}$ in

the unaffected loops (and the RCS as a whole) is less severe. Because the transient takes slightly longer to reach 1$[7] of positive reactivity addition from moderator temperature effects, the reactivity excursion from prompt criticality happens at a similar time. Doppler effects almost immediately turn the reactivity excursion around and the reactor trip follows within a couple of seconds. The negative reactivity insertion is slightly different from the AOR; though there is less negative reactivity added by Doppler effects, the total reactivity ends up lower than the AOR after the reactor trip.

Overall, the key phenomena (the RCS cooldown, reactivity transient, pressure transient, etc.) are predicted reasonably closely to the AOR, considering modeling and input differences. The NRC staff concluded that the benchmark provided an appropriate demonstration of the capabilities of the Harris RETRAN-3D base model. The validity of the RETRAN-3D analysis as applied to Harris and Robinson is also supported by the similarity of the AOR analysis in S-RELAP5 to the demonstration analysis.

### 3.5.2 Increase in Steam Flow

The increase in steam flow event is modeled as a ten percent step increase in steam flow. A reactor trip is not required to mitigate the event, but is credited if predicted to occur in the analysis. The DPC-NE-3009 methodology classifies the event as an ANS Condition II event for both Harris and Robinson, consistent with the plant UFSARs, and the corresponding acceptance criteria are applied. The acceptance criteria are to ensure that the MDNBR is above the 95/95 limit and that the CFM limit is not exceeded. The MDNBR is determined using the SCD methodology

Peak primary and secondary pressure are considered to be bounded by the loss of external electrical load or turbine trip transients (depending on the plant) and are not analyzed. The increase in steam flow is an overcooling event that may result in an increase in power caused by moderator reactivity feedback. Though a power increase may occur, the secondary side heat sink remains available to remove heat from the primary system, unlike in the loss of load and turbine trip events, which are thus bounding with respect to pressure response. The NRC staff determined that this treatment of the primary and secondary pressure transients was therefore acceptable.

The analysis is performed at combinations of BOC/EOC and HZP/HFP conditions, with rod control in both automatic and manual to identify the limiting case. RETRAN-3D is used for the systems analysis, and the **[[ ]]** VIPRE-01 model is used to evaluate DNBR.

The NRC staff determined that the licensee's approach to analyzing the increase in steam flow event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

### 3.5.3 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a SG relief or safety valve event is initiated by the opening of a single steam dump valve, SG PORV, or SG safety valve that will not reseat and cannot be

[7] To determine the reactivity in dollars ($), the reactivity, ρ, is normalized to the delayed neutron fraction, $\beta_{eff}$. One dollar of reactivity insertion puts the core into a prompt-critical condition.

isolated. The DPC-NE-3009 methodology classifies the transient as an ANS Condition II event for Harris and a Condition IV event for Robinson. This is consistent with the event classification in the plant UFSARs, though it is worth noting that the current Robinson UFSAR considers the inadvertent opening of a SG relief or safety valve to be bounded by the 10-percent load increase and the main SLB (for pre- and post-trip conditions, respectively). The Harris acceptance criteria are to ensure that the MDNBR, determined using the SCD methodology, remains above the 95/95 limit and that the CFM limit is not exceeded. The Robinson acceptance criterion is that the radioactive material release does not exceed the dose analysis.

For both plants, the peak primary and secondary pressure responses are considered to be bounded by the loss of external electrical load or turbine trip transients (depending on the plant) and are not analyzed. The inadvertent opening of a SG relief or safety valve increases the primary-to-secondary heat removal capability; as such, it is an overcooling event and may result in an increase in power caused by moderator reactivity feedback. Though a power increase may occur, the secondary side heat sink remains available to remove heat from the primary system, unlike in the loss of load and turbine trip events, which are thus bounding with respect to pressure response. The NRC staff determined that this treatment of the primary and secondary pressure transients was therefore acceptable.

The DNB response is also considered to be bounded by the increase in steam flow transient, because the maximum steam flow through the valves considered in this transient is less than that considered in the increase in steam flow. The SLB is also bounding with respect to DNB for when this event is initiated from lower modes, for a similar reason to the increase in steam flow. However, an inadvertent opening of a SG relief or safety valve analysis will be performed in the event that the SLB analysis, which is an ANS Condition IV event, predicts fuel failure. This is because the inadvertent opening of a SG relief or safety valve is an ANS Condition II event and is not allowed to fail fuel. The NRC staff finds this approach to be acceptable.

The analysis is performed using the same model and initial/boundary conditions as the main SLB analysis discussed in Section 5.1.4 of DPC-NE-3009, except for break area. As discussed in the licensee's response to RAI 18 (Reference 5), a range of break areas are considered to find the limiting condition, up to the largest flow area of any steam dump, relief, or safety valve at the full open position. This is acceptable, since the inadvertent opening of a SG relief or safety valve would essentially be the same as a SLB with an upper limit on the possible break size. Accordingly, the NRC staff considers RAI 18 resolved.

The NRC staff determined that the licensee's approach to analyzing the inadvertent opening of a SG relief or safety valve event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

### 3.5.4 Steam System Piping Failure

The steam system piping failure (hereafter referred to as the SLB) is initiated by a break in the main steam line between the SG outlet flow restrictor and the main steam isolation valve (MSIV). The DPC-NE-3009 methodology has classified it as an ANS Condition IV event for both Harris and Robinson, consistent with the plant UFSARs. The corresponding acceptance criterion is therefore to ensure that the radioactive material release does not exceed the dose analysis. To that end, the number of fuel pins exceeding the 95/95 DNBR limit and the percentage of fuel exceeding the CFM limit are both tallied. For transients initiated from a HFP

condition, the SCD methodology is used to determine MDNBR; for transients initiated from HZP, the standard, non-SCD methodology is used.

The peak primary and secondary pressure responses are considered to be bounded by the loss of external electrical load or turbine trip transients (depending on the plant) and are not analyzed. Though the SLB is an overcooling event and may result in an increase in reactor power and thus an increase in primary and secondary pressure, the MSSVs are available and may remove heat from the primary and secondary systems. The NRC staff determined that this treatment of the primary and secondary pressure transients was therefore acceptable.

The main analysis is performed at HZP. The licensee justified this based on the assumption that the highest worth control rod remains stuck out of the core, which could result in recriticality, which potentially challenges core limits. However, the licensee also analyzes the SLB at HFP conditions to confirm that the HZP condition is limiting. As discussed in the licensee's response to RAI 19 (Reference 5), the HFP SLB is more likely to be limiting from the time of initiation to the time of the reactor trip; following the trip, the return to power is more severe in the HZP case than the HFP case, and the HZP case is thus more likely to be limiting after the trip.

Analyses are also performed with and without a loss of offsite power (LOOP) to determine the limiting condition. The timing of the LOOP is dependent on the case analyzed. For a HZP SLB, the LOOP is assumed to be coincident with the break; for a HFP SLB, the LOOP is assumed to be coincident with the reactor trip signal.

The DNBR and fuel power peaking factors are analyzed using VIPRE-01 and SIMULATE-3, respectively, with thermal-hydraulic inputs from RETRAN-3D at the "limiting statepoint." The licensee defined the limiting RETRAN-3D statepoint as the system state based on the "time of maximum core average surface heat flux, with corresponding values of core inlet mass flow rate, core inlet temperature, and core exit pressure." As discussed in the response to RAI 20 (Reference 5), the licensee **[[**

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3.5.4.1 RETRAN-3D Models and Options

The system response to the SLB transient is analyzed using RETRAN-3D models and options that differ from the base model. The reactor vessel nodalization is the one described in Section 3.2.2.1 of this SE, and the SG secondary side nodalization is the one described in Section 3.2.2.2 of this SE. As discussed in those sections, these models are acceptable for use in Harris and Robinson UFSAR Chapter 15 non-LOCA transient analyses conducted using the methods discussed in DPC-NE-3008 and DPC-NE-3009.

3.5.4.1.1 Primary System and ECCS Models and Options

The general transport model is used to track boron injected from the ECCS. As was noted in Section 3.2.2.3.5 of this SE, this is generally acceptable. **[[**

**]]**. As discussed in the licensee's response to RAI 21 (Reference 4), a conservatively small boron worth is used, **[[**

**]]**. Purge volumes are also included in the analysis, and borated water from the refueling water storage tank does not reach the RCS until all of the unborated water has been injected.

Different signals are expected to actuate safety injection (SI) depending on the plant being analyzed. At Harris, SI is actuated on low pressurizer pressure or low main steam line pressure. At Robinson, SI is actuated on low pressurizer pressure or high steam flow coincident with low main steam line pressure. When the SLB is initiated from HFP conditions, the SI signal also causes a reactor trip.

As discussed in the response to RAI 26 (Reference 5), minimum flow is modeled from one high head SI (HHSI) pump. At Robinson, the second HHSI pump is considered to be lost by the single failure assumption. At Harris, the second HHSI pump is presumed lost by the single failure assumption and the third is presumed to be out of service. The HHSI pump injects water to the RCS cold legs, and interaction with accumulator flow is assumed to be negligible. Water injected from the ECCS is conservatively assumed to have a low boron concentration and temperature, based on the TS minimum values with consideration of uncertainties; this minimizes the negative reactivity insertion and exacerbates the cooldown.

Cold leg accumulators are included in the analysis using RETRAN-3D's built-in accumulator model. As discussed in the licensee's response to RAI 22 (Reference 4), the initial condition of the accumulator is specified using minimum values of pressure, temperature, inventory, and boron concentration. Because the accumulators are not expected to empty during the period of interest for the transient, non-condensable gases (like the nitrogen cover gas that pressurizes the accumulators) are not expected to enter the RCS and are accordingly not modeled. Depending on the initial conditions, the accumulators may or may not inject.

3.5.4.1.2 Secondary System Models and Options

The break, as discussed above, is modeled between the SG outlet flow restrictor and the MSIV. **[[**

**]]**. The Moody critical flow model, which was approved by the NRC staff in the generic SE on RETRAN-3D, is used.

Conservatism is added to the SG model by suppressing liquid carryout, which maximizes the primary system cooldown. This is achieved by **[[**

**]]**. As discussed in RAI 23 in (Reference 5), previously approved Duke Energy methodologies have gone a step further and **[[**

**]]**. The licensee's response to this RAI stated that if the bubble-rise velocity adjustment is found to be inadequate,

**[[** **]]**.
This adjustment was approved by the NRC staff in DPC-NE-3005 and is therefore acceptable. Accordingly, the NRC staff considers RAI 23 resolved.

MFW systems are modeled differently depending on the plant. At Harris, the MFW pumps take suction from the condensate booster pumps, which take suction from the condensate pumps. At Robinson, the MFW pumps take suction directly from the condensate pumps. On a LOOP, the condensate or condensate booster pumps trip, and then MFW pumps trip on one of three signals: condensate or condensate booster pump trip, low section pressure, or SI. There is no MFW trip when offsite power is maintained. Flow to the faulted SG is modeled to conservatively bound the expected hydraulic performance and system alignments of the condensate and MFW systems consistent with the initial condition of the reactor (HZP or HFP).

Flow from all AFW pumps is initiated based on a SI signal or LOOP, with no delay between the initiating signal and delivery of flow to the SGs. This is not consistent with the Harris and Robinson UFSARs, where AFW flow starts at the time of the break. As discussed in the licensee's response to RAI 24 (Reference 4), the initiation of AFW flow at the time of the break was judged to be overly conservative and inconsistent with other Chapter 15 safety analyses. The NRC staff agrees that the current modeling of AFW in the plant UFSARs is conservative, and changing the modeling will remove some conservatism. However, other modeling assumptions make up for this loss of conservatism, and regardless the demonstration analyses provided by the licensee show that AFW is actuated very soon (within seconds) after the transient initiation. As with MFW, flow to the faulted SG is modeled to conservatively bound the expected hydraulic performance of the AFW system. The licensee's response to RAI 24b stated that AFW is assumed to be terminated by operator action within 10 minutes following the SLB, consistent with the current plant UFSARs.

Main steam isolation is modeled differently depending on the plant. At Harris, main steam isolation signals typically occur on low steam line pressure; at Robinson, it is high steam line flow coincident with low steam line pressure. The MSIV closure time accounts for signal processing delays and assumes the valve closes conservatively slowly.

3.5.4.2 Modeling of Core Power Distributions and Reactivity Feedback

At HZP conditions, **[[**

**]]**. At HFP, the SIMULATE-3 3D power distributions corresponding to the limiting HFP statepoint are used to demonstrate that the DNB and CFM acceptance criteria are met.

Both the system analysis results from RETRAN-3D and the power distribution from SIMULATE-3 are used in the DNB evaluation, which is performed with VIPRE-01. The SIMULATE-3 3D power distribution is also used in the CFM evaluation.

3.5.4.2.1 RETRAN-3D Point Kinetics Temperature Feedback

For the RETRAN-3D point-kinetics temperature feedback at HZP, **[[**

**]]**.
This is consistent with the NRC-approved model discussed in DPC-NE-3001-P, Revision 1.

Since RETRAN-3D cannot be initialized from a subcritical condition, control rod insertion is modeled at the same time as break initiation for the HZP case. Sufficient negative reactivity is added to make the core subcritical by the TS limit on shutdown margin. At the end of insertion, all control rods are inserted except for the most reactive rod, which is modeled to be stuck in the fully withdrawn position.

For HFP point kinetics temperature feedback before the reactor trip, the licensee uses the most negative MTC corresponding to the time in cycle used in the analysis. Least negative DTC values are used, and control rod trip reactivity assumes the most reactive rod is stuck in the fully withdrawn position. Other kinetics parameters are specified at the time in cycle used to define the moderator and Doppler feedback. After the trip, **[[**
**]]**.

3.5.4.2.2 Kinetics for HZP with Offsite Power Lost

Duke uses SIMULATE-3K to account for effects on reactivity from the main SLB in the HZP case when offsite power is lost. This is because SIMULATE-3 does not account for the significant moderator void formation that occurs due to the low flow condition in the transient. As with the other cases, inputs for the SIMULATE-3K analysis are taken from the limiting RETRAN-3D statepoint. In general, the intent of the licensee's SIMULATE-3K analysis for this case is to demonstrate that the HZP with offsite power lost case is not limiting.

3.5.4.3 DNB Evaluation

The VIPRE-01 model discussed in Section 3.1.3 of this SE is used for the DNB evaluation. RETRAN-3D thermal-hydraulic conditions and SIMULATE-3 axial and radial power distributions are used as input to VIPRE to calculate the local coolant properties and DNBR. The licensee stated in its methodology that **[[**

**]]**.

The HTP correlation is used, unless it falls outside of the approved correlation range. If this is the case, one of the correlations discussed in Section 3.3.1 is used.

3.5.4.4 Centerline Fuel Melt Evaluation

The licensee develops linear heat generation rate (LHGR) limits for the fuel types in the core using NRC-approved fuel thermal-mechanical methodologies and fuel performance codes.

In the HZP case, RETRAN-3D is used to determine the core-wide power response following the main SLB, and this core response is used as an input to SIMULATE-3, which is used to determine peaking factors. These power distributions, with adders for the appropriate uncertainties, are compared to the LHGR limits to confirm that the response is acceptable.

In the HFP case, 3D power distributions are generated in SIMULATE-3 at **[[**

**]]**. Consistent with the plant maneuvering analysis approved in DPC-NE-2011, the control rod positions and xenon distributions are those allowed by RILs and AFD limits at the initial power level assumed in the transient analysis. The rest of the CFM evaluation (addition of peaking factor uncertainties, comparison to LHGR limits) is the same as the HZP analysis.

3.5.4.5 Cycle Specific Reload Evaluation

Each cycle is evaluated to confirm **[[**

**]]**. The core power distribution is evaluated to confirm that DNBR and CFM limits are not exceeded. If any of these reload checks fails, the core is further evaluated, re-analyzed, or redesigned. If the HFP case is found to be limiting, the power distributions calculated in the CFM evaluation are used to confirm the acceptability of the DNBR and CFM limits for reload cycles.

The licensee also stated that a power search could be performed **[[**

**]]**. This power search was described in more detail in the response to RAI 25 (Reference 4). **[[**

**]]**. Peaking factors from the core power distribution predicted by SIMULATE-3 are then used in subsequent calculations to confirm the acceptability of the DNBR and CFM limits.

3.5.4.6 Harris Steam Line Break Demonstration Analysis (DPC-NE-3009)

NRC staff reviewed the specific assumptions for the Harris SLB demonstration analysis presented in Section 6.1 of DPC-NE-3009 and found that they conformed to the method as discussed in Sections 3.5.4.1 and 3.5.4.2 of this SE.

In the transient, all three SGs depressurize until the MSIVs close at 9 seconds. At that point, the unfaulted SGs repressurize slightly until about 20 seconds, at which point the hot legs in the

unfaulted loops reach thermal equilibrium with the SGs, which are saturated. The faulted SG continues to depressurize rapidly until the MFW valve closes at 12 seconds, then slowly depressurizes throughout the remainder of the transient. The model appropriately captures the asymmetry of the response. The SGs and reactor coolant loops continue to slowly cool throughout the duration of the transient, and core power/core average surface temperature does not reach a peak until about the same time as boric acid begins reaching the core at 243 seconds.

In the Harris demonstration analysis, there is a power spike as the reactor returns to critical conditions. The licensee's response to RAI 27 (Reference 5) indicated that the total reactivity of the system at the point of recriticality was 0.95$, slightly avoiding prompt criticality. The licensee does not specifically search for a condition that becomes prompt critical during the transient, but rather attempts to specify the initial and boundary conditions such that the limiting condition is determined. Ultimately, the transient response is not significantly impacted by whether or not the reactor reaches a prompt critical state.

With this in mind, the NRC staff reviewed the overall transient response and found that it met with expectations. Key differences in the response between the demonstration analysis and the Harris UFSAR AOR can be attributed to modeling differences between the new methodology and the current methodology. For example, in the new methodology, the modeling of the MFW system bounds the increase in feedwater injected into the faulted SG, caused by the drop in SG pressure, until the MFW system is isolated. This allows mass to remain in the faulted SG throughout the transient. In the current AOR, the SGs empty at approximately 360 seconds and the SG tubes enter an oscillatory dryout-rewet cycle.

3.5.4.7 Robinson Steam Line Break Demonstration Analysis (DPC-NE-3009)

NRC staff reviewed the specific assumptions for the Robinson SLB demonstration analysis presented in Section 6.2 of DPC-NE-3009 and found that they conformed to the method as discussed in Sections 3.5.4.1 and 3.5.4.2 of this SE.

In the transient, all three SGs depressurize until the MSIVs close at approximately 19 seconds. Beyond that point, only the faulted SG continues to depressurize, until the MFW isolation valves close at approximately 40 seconds. Flow continues out the break at a much slower rate and the pressure in the faulted SG stabilizes. The unfaulted SGs reach approximately the same temperature as the associated loops following the MSIV closure and thus do not act as a heat sink or source for the remainder of the transient. Loop temperatures in the unfaulted loops thus remain relatively steady for the duration of the transient. The faulted loop cold leg temperature and the temperature in the faulted region of the core slowly increase after the initial depressurization until the faulted SG empties at around 540 seconds, at which point the temperature jumps up. The increasing cold leg temperature causes the positive reactivity added by the moderator temperature to slowly decrease over time, allowing the core power to drop for the remainder of the transient from its peak at approximately 50 seconds.

The NRC staff reviewed the transient response and found it met with the NRC staff's expectation. The demonstration analysis was performed at HZP while the AOR in the Robinson UFSAR was performed at HFP; as such, it is difficult to compare the two. However, the transient behaves as expected given the modeling options.

3.5.4.8 Conclusion Regarding Steam Line Break Methodology

The NRC staff determined that the licensee's approach to analyzing the SLB event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. The approach taken in evaluating the MDNBR and margin to the CFM limits is acceptable.

3.5.5 Loss of External Electrical Load

The loss of load transient is initiated by a loss of load on the turbine generator, with offsite ac power remaining available. The DPC-NE-3009 methodology classifies it as an ANS Condition II event for both Harris and Robinson, consistent with the plant UFSARs, and the corresponding acceptance criteria are applied. The acceptance criteria are to ensure that the MDNBR is above the 95/95 limit and peak primary and secondary system pressure remain below 110 percent of design values. Three transient analysis cases, (1) core cooling (i.e., the DNBR evaluation), (2) primary pressure, and (3) secondary pressure, are analyzed separately because they require different limiting assumptions. The MDNBR is analyzed using the SCD methodology.

The loss of load is a loss of heat sink event that is caused by the closure of the turbine control valves. It is similar to the turbine trip, where heat sink is lost due to closure of the turbine stop valves. The turbine stop valves close faster than the control valves, and as such the turbine trip is a more severe transient when the reactor trip on turbine trip is not credited. Since this is the case for Harris, the loss of load is bounded by the turbine trip for Harris and is only analyzed for Robinson. The NRC staff finds this to be acceptable. With respect to DNB, the loss of load is bounded by the withdrawal of a single full-length RCCA transient, which experiences elevated total core power and local peaking. Thus, no DNB analysis is envisioned for the loss of load event, unless the ANS Condition III withdrawal of a full-length RCCA analysis predicts fuel failure, which is unacceptable for the loss of load. The NRC staff finds this to be acceptable.

NRC staff reviewed the initial and boundary conditions for the three transient analyses and found them to be acceptable. While the loss of external load transient is not analyzed at Harris in the current plant UFSAR (since it is said to be bounded by the turbine trip) the differences between the initial and boundary conditions in the current AOR at Robinson are minor. For the core cooling case, the availability of the main steam PORVs and steam dumps are said to be evaluated in the proposed method, while they are disabled in the Robinson UFSAR. Since evaluation would determine whether it is more limiting to have them enabled or disabled, this is acceptable. For the primary pressure case, the pressurizer heaters are available in the proposed method while they may or may not be available in the Robinson UFSAR (heaters are not discussed for the loss of load). Since use of the heaters would increase the primary pressure and exacerbate the transient, this is acceptable. For the secondary pressure case, the pressurizer level is high in the proposed method while it is nominal in the Robinson UFSAR. A high pressurizer level would result in an increased primary pressure because of the reduced steam space in the pressurizer, and would thus result in increased secondary pressure. It is therefore acceptable.

Core physics parameters are assumed at BOC/HFP conditions. This is consistent with the Harris and Robinson UFSARs, and would tend to exacerbate the core response to an overheating or overpressurization event. The NRC staff determined that the choice of time in

life is expected to yield the limiting safety margins and is therefore acceptable. No single active failure was identified that would make the accident conditions worse; this is consistent with the Harris and Robinson UFSARs and is thus acceptable.

The NRC staff determined that the licensee's approach to analyzing the loss of load event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.6 Turbine Trip

The rapid closure of turbine stop valves associated with a turbine trip causes a sudden reduction in steam flow and thus an increase in secondary and primary temperature and pressure. The DPC-NE-3009 methodology classifies it as an ANS Condition II event for both Harris and Robinson, consistent with the plant UFSARs, and the corresponding acceptance criteria are applied. The acceptance criteria are to ensure that the MDNBR is above the 95/95 limit and peak primary and secondary system pressure remain below 110 percent of design values. Three transient analysis cases, (1) core cooling (i.e., the DNBR evaluation), (2) primary pressure, and (3) secondary pressure, are analyzed separately because they have different assumptions. The MDNBR is analyzed using the SCD methodology.

The turbine trip is a loss of heat sink event caused by the closure of the turbine stop valves. It is similar to the loss of load, where heat sink is lost due to closure of the turbine control valves. The turbine stop valves close faster than the control valves, and as such the turbine trip is a more severe transient when the reactor trip on turbine trip is not credited. In the Robinson loss of load analysis, the turbine stop valve closure time is used in the loss of load analysis, so it is limiting with respect to the turbine trip; because of this, the turbine trip is only analyzed for Harris. The NRC staff finds this to be acceptable. With respect to DNB, the turbine trip is bounded by the withdrawal of a single full-length RCCA transient, which experiences elevated total core power and local peaking. Thus, no DNB analysis is envisioned for the turbine trip event, unless the ANS Condition III withdrawal of a full-length RCCA analysis predicts fuel failure, which is unacceptable for the turbine trip. The NRC staff finds this to be acceptable.

NRC staff reviewed the initial and boundary conditions for the three transient analyses and found them to be acceptable. While the turbine trip is not analyzed at Robinson in the current plant UFSAR (since it is bounded by the loss of external load transient) the differences between the initial and boundary conditions in the current AOR at Harris are minor. In the peak secondary pressure case, the Harris UFSAR lists the RCS flow rate as minimum while the proposed method assumes a high RCS flow rate. This is acceptable to the NRC staff as an increased RCS flow rate will result in improved primary-to-secondary heat transfer and will exacerbate the secondary pressure response. SG narrow range level is listed as nominal in all three cases in the Harris UFSAR, while the proposed method assumes that the core cooling case is insensitive to SG level (which implies nominal conditions) and the primary and secondary pressure cases assume high SG level. This is acceptable to the NRC staff because increasing the amount of liquid in the SGs increases the amount of energy stored in the secondary system, which will tend to exacerbate both the primary and secondary peak pressures.

Core physics parameters are assumed at BOC/HFP conditions. This is consistent with the Harris and Robinson UFSARs, and would tend to exacerbate the core response to an

overheating or overpressurization event. The NRC staff determined that the choice of time in life is expected to yield the limiting safety margins and is therefore acceptable. No single active failure was identified that would make the accident conditions worse; this is consistent with the Harris and Robinson UFSARs and is thus acceptable.

The NRC staff determined that the licensee's approach to analyzing the turbine trip event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.6.1 Harris Turbine Trip Benchmark (DPC-NE-3008)

The Harris turbine trip benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008, with a modification that removes the main steam lines downstream of the steam header to better match the transient response time in the AOR. Additionally, models of the pressurizer safety valves and MSSVs **[[** **]]**. As discussed in the response to RAI 16 (Reference 3), the opening and closing characteristics of these valves also include **[[** **]]**. The licensee justifies the approach to valve modeling based on the comparison in the system response between the benchmark analysis and the AOR.

Two cases were analyzed: primary system overpressurization and secondary system overpressurization. The secondary system overpressurization benchmark analysis is very similar to the AOR. The primary system overpressurization benchmark analysis is similar to, but not identical to, the AOR. Primary system phenomena, especially temperature and pressure rise, are captured, though the change in RCS temperatures is not as severe as the AOR. Maximum pressure is about the same as or even slightly higher than the AOR; this is because in the RETRAN-3D model the temperature is increasing and the pressurizer safety valves are not yet closed, whereas in the AOR the pressurizer safety valves close while the temperature is still increasing, causing the pressure to temporarily increase when the valve reseats. The pressurizer collapsed liquid level is slightly lower than the AOR at the time of peak pressure, but as the pressure decreases the level trends down slower in the benchmark than in the AOR, eventually resulting in a higher level in the pressurizer.

Overall, though there are some differences between the analyses in both of the scenarios, they are minimal and easily attributable to modeling differences. The NRC staff concluded that the benchmark provided an appropriate demonstration of the capabilities of the Harris RETRAN-3D base model–particularly the valve modeling. Accordingly, the NRC staff considers RAI 16 (DPC-NE-3008) resolved.

3.5.7 Inadvertent Closure of Main Steam Isolation Valves

As discussed in the licensee's response to RAI 14 (Reference 5), all acceptance criteria for the inadvertent MSIV closure event are bounded by the loss of load or turbine trip, depending on the plant. The NRC staff finds this acceptable since inadvertent closure of the MSIVs results in a loss of steam flow that is similar to a turbine trip, but because the MSIVs close more slowly than the turbine stop and control valves, the transient is less severe. Thus, the inadvertent MSIV closure event is not analyzed in the DPC-NE-3009 methodology.

### 3.5.8 Loss of Condenser Vacuum

As discussed in the licensee's response to RAI 14 (Reference 5), all acceptance criteria for the loss of condenser vacuum event are bounded by the loss of load or turbine trip, depending on the plant. The NRC staff finds this acceptable since a loss of condenser vacuum produces a turbine trip that precludes the use of the steam dumps, which redirect steam flow to the condenser. Since steam dumps are not credited for the loss of load or turbine trip transients within the licensee's methodology, those events bound the loss of condenser vacuum event. Thus, the loss of condenser vacuum event is not analyzed in the DPC-NE-3009 methodology.

### 3.5.9 Loss of Non-Emergency ac Power to the Station Auxiliaries

This transient is initiated by a complete loss of all non-emergency ac power, which causes a primary system heatup due to the loss of reactor coolant and MFW pumps (among other systems). The loss of non-emergency ac power is not analyzed at Robinson in the current plant UFSAR, since it is bounded by the loss of feedwater and loss of flow transients. Comparisons were made to the Harris UFSAR analysis to determine the acceptability of the assumptions for reactor trips, system availability, initial conditions, and boundary conditions. The reactor usually trips on RCP under-frequency or under-voltage, low RCS flow, or low SG water level. The licensee indicated that loss of gripper coil voltage may cause early control rod insertion, which may be credited. This stands counter to the Harris UFSAR, which specifically does not credit this early control rod insertion. However, it is consistent with the methodology approved by the NRC staff in DPC-NE-3002, Revision 0 (Reference 33),[8] and is thus considered acceptable.

The DPC-NE-3009 methodology classifies the loss of non-emergency ac power event as an ANS Condition II event for both Harris and Robinson, consistent with the plant UFSARs, and the corresponding acceptance criteria are applied. The acceptance criteria are to ensure that the MDNBR is above the 95/95 limit, that AFW provides adequate long-term core cooling, that peak primary and secondary system pressure remain below 110 percent of design values, and that liquid relief through pressurizer PORVs does not occur. Acceptance criteria for the long-term core cooling analysis are to demonstrate that the reactor coolant temperature is decreasing, that at least one SG is being refilled, and that no liquid is passed through the PORVs or pressurizer safety valves. Four transient analysis cases, (1) short term core cooling (i.e., the DNBR evaluation), (2) long-term core cooling, (3) primary pressure, and (4) secondary pressure, are analyzed separately because they have different assumptions. The MDNBR is analyzed using the SCD methodology.

The complete loss of forced reactor coolant flow (hereafter loss of flow) transient bounds the short-term core cooling capability (i.e., DNB) and peak primary pressure responses. The loss of non-emergency ac power causes the RCPs to shut off and coast down, like in the loss of flow transient. The primary difference between the loss of non-emergency ac power and the complete loss of flow is the timing of the control rod drop. In the loss of non-emergency ac power, the rods begin to fall (due to the loss of power), while in the loss of flow transient the rods only fall after instrumentation delays. The response of the loss of flow event is thus more limiting. However, if the ANS Condition III loss of flow analysis predicts fuel failure, which is unacceptable for the ANS Condition II loss of non-emergency ac power event, the method presented here will be used to analyze short-term core cooling. The NRC staff finds this to be acceptable.

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[8] DPC-NE-3002, Revision 4b in the reference documents the NRC approval of DPC-NE-3002, Revision 0.

The secondary pressure response for the loss of non-emergency ac power event is bounded by the loss of load or turbine trip events, depending on the plant. In all three events, secondary side pressure does not experience a substantial rise until after the turbine trip. In the loss of non-emergency ac power event, the reactor trips prior to the turbine trip, meaning that the heat generation is already decreasing by the time the turbine trip occurs. In the loss of load and turbine trip events, the reactor trip occurs well after the turbine trip, resulting in a much more severe secondary side pressurization transient because of the extra heat that must be removed from the RCS following the turbine trip. The NRC staff finds this justification to be acceptable.

Long-term core cooling capability is bounded by the loss of normal feedwater flow transient. The loss of non-emergency ac power is very similar to the loss of normal feedwater flow event when offsite power is lost after the turbine trip. As in the short-term core cooling case, the loss of non-emergency power results in an immediate reactor trip while the trip in the loss of normal feedwater flow is delayed. Because this results in a greater addition of heat prior to the trip for the loss of normal feedwater flow, it is considered bounding. The NRC staff finds this justification to be acceptable.

NRC staff reviewed the initial and boundary conditions for the four transient analyses and found them to be acceptable. The differences between the initial and boundary conditions in the current AOR at Harris are minor. Pressurizer heaters, sprays, and PORVs are disabled for all four analyses except for the peak secondary pressure analysis, where the availability of pressurizer sprays are evaluated. While this is not consistent with the current Harris UFSAR analysis, it is consistent with the previously approved Duke methodology documented in DPC-NE-3002, Revision 4 (Reference 33). The NRC staff thus finds it to be acceptable.

Core physics parameters are assumed at BOC/HFP conditions for the short-term core cooling and peak pressure evaluations. This is consistent with the Harris and Robinson UFSARs, and would tend to exacerbate the core response to an overheating or overpressurization event. For the long-term core cooling case, EOC conditions are assumed; this maximizes the decay heat and is expected to be limiting. The NRC staff therefore determined that the choice of time in life is expected to yield the limiting safety margins and is thus acceptable.

The flow coastdown is modeled in the same way as in the loss of forced reactor coolant flow transient, which is reviewed by the NRC staff in Section 3.5.12 of this SE.

The limiting single failure is assumed to occur in the AFW system. AFW actuates on either low SG level or loss of non-emergency ac power with a conservative delay regardless of the actuating signal. Minimum flow from at least one AFW pump is assumed. As discussed in the response to RAI 28 (Reference 4), the licensee believes that the limiting single failure will likely result in the loss of one of three AFW pumps. However, this is inconsistent with the plant AORs at Harris and Robinson; therefore, the licensee will evaluate the treatment of the AFW system (i.e., the number of SGs receiving flow and number of AFW pumps available) to determine whether two pumps may be credited. The NRC staff finds the use of either one or two AFW pumps to be acceptable within the DPC-NE-3008/DPC-NE-3009 methodology, but has neither reviewed nor approved any changes to the current licensing basis failure modes analysis or single failure assumptions. Accordingly, the NRC staff considers RAI 28 resolved.

The NRC staff determined that the licensee's approach to analyzing the loss of non-emergency ac power event was appropriate. The acceptance criteria, initial and boundary conditions,

system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.10 Loss of Normal Feedwater Flow

The loss of normal feedwater flow event is initiated by a loss of all normal feedwater flow to the SGs, which causes an increase in the secondary system temperature and pressure, and thus an increase in the primary system temperature and pressure. The DPC-NE-3009 methodology classifies it as an ANS Condition II event for both Harris and Robinson, consistent with the plant UFSARs, and applies the corresponding acceptance criteria. The acceptance criteria are to ensure that the MDNBR is above the 95/95 limit, that AFW provides adequate long-term core cooling, that peak primary and secondary system pressure remain below 110 percent of design values, and that liquid relief through pressurizer PORVs does not occur. Acceptance criteria for the long-term core cooling analysis are to demonstrate that the reactor coolant temperature is decreasing, that at least one SG is being refilled, and that no liquid is passed through the PORVs or pressurizer safety valves. Though both the Harris and Robinson UFSARs state that DNB is bounded by the loss of forced flow transient, evaluating this transient to ensure DNB criteria are met is reasonable. The reactor usually trips on OTΔT, high pressurizer pressure, or low SG water level. This is consistent with the discussions in the Harris and Robinson UFSARs.

Core cooling (i.e., the DNBR evaluation), primary pressure, and secondary pressure cases are analyzed separately because they have different assumptions. The MDNBR is analyzed using the SCD methodology. Analysis is performed with offsite power maintained as well as lost in order to determine the limiting condition. For the LOOP case, the LOOP occurs on a turbine trip and causes the RCPs to start coasting down after a conservative delay. This is consistent with the Robinson UFSAR, and, because it evaluates the limiting condition, is acceptable to the NRC staff.

The licensee considers peak primary and secondary pressure responses to be bounded by the loss of load or turbine trip events, depending on the plant. As in the loss of normal feedwater events, these events involve a mismatch between the primary heat source and the secondary heat sink. However, while the reactor and turbine trip are simultaneous in the loss of normal feedwater flow events, the reactor trip is delayed relative to the turbine trip in the loss of load and turbine trip events. Thus the mismatch in heat generation and heat removal in the turbine trip and loss of load, and therefore the primary and secondary pressure responses, bounds the loss of normal feedwater. The NRC staff finds this justification to be acceptable.

NRC staff reviewed the initial and boundary conditions for the four transient analyses and found them to be acceptable. The differences between the initial and boundary conditions in the current AORs at Harris and Robinson are minor. The long-term and peak primary pressure cases are designed with initial conditions that minimize the primary-to-secondary heat transfer. The peak secondary pressure case, by contrast, maximizes the primary-to-secondary heat transfer. These conditions are acceptable for the given scenarios. Primary and secondary control system availability is reasonable for all four transient analysis cases, and will tend to exacerbate the response for the figure of merit. For example, for the peak primary pressure case, pressurizer sprays and PORVs are unavailable, as are main steam PORVs and steam dumps. These assumptions would prevent systems from actuating to decrease primary and secondary system pressure and are thus conservative.

Core physics parameters are assumed at BOC/HFP conditions for the short-term core cooling and peak pressure evaluations. This is consistent with the Harris and Robinson UFSARs, and would tend to exacerbate the core response to an overheating or overpressurization event. For the long-term core cooling case, EOC conditions are assumed; this maximizes the decay heat and is expected to be limiting. The NRC staff therefore determined that the choice of time in life is expected to yield the limiting safety margins and is thus acceptable.

As in the loss of non-emergency power to the station auxiliaries transient, the limiting failure is assumed in the AFW system. The same considerations discussed in Section 3.5.9 of this SE apply.

The NRC staff determined that the licensee's approach to analyzing the loss of normal feedwater flow event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.10.1 Robinson Loss of Normal Feedwater Flow Benchmark (DPC-NE-3008)

The Robinson loss of normal feedwater flow benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008 without modification. The case analyzed assumed the RCPs were running, which is limiting from a long-term core cooling capability standpoint. Inputs were matched to the AOR, and for further consistency with the AOR the high pressurizer pressure trip was used to trip the reactor.

The reactor vessel outlet temperature is slightly different between the benchmark and the AOR; this may be due to **[[**
**]]**, similar to that discussed in the licensee's response to RAI 19 (Reference 3), and is expected to have no significant impact on the results. Otherwise, the primary system response in the benchmark very closely follows the AOR curve until the point of the reactor trip, which occurs about 2 seconds earlier in the benchmark than the AOR.

After the trip, the vessel inlet, outlet, and average temperatures converge much more quickly than in the AOR. The slight increase in temperature seen following the reactor trip in the vessel outlet and vessel average temperatures occurs in both the AOR and the benchmark analysis, as discussed in the licensee's response to RAI 18 (Reference 3). As explained in the RAI response, it is caused by changes in primary-to-secondary heat transfer resulting from the turbine trip that is coincident with the reactor trip and the later MSSV opening.

The SG pressure is slightly different between the AOR and the benchmark–the maximum value is not quite as high, though it does rise to the MSSV setpoint. The NRC staff concludes that much of the difference in secondary pressure and SG liquid inventory between the AOR and the benchmark is driven by differences in modeling the MSSVs. This in turn influences the primary system response.

The peak primary system pressure and trends in long-term primary pressure and SG liquid inventory from the benchmark match the AOR well; these are the key attributes evaluated in the safety analyses. Ultimately, the NRC staff determined that differences observed in the transient responses can be attributed to modeling differences between the AOR and the benchmark analysis. The NRC staff therefore concluded that the benchmark provided an appropriate

demonstration of the capabilities of the Robinson RETRAN-3D base model. Accordingly, the NRC staff considers RAIs 18 and 19 (DPC-NE-3008) resolved.

3.5.11 Feedwater System Pipe Break

The feedwater system pipe break transient is initiated by a double-ended guillotine break in the MFW line to a SG. The overall trajectory of the feedwater line break transient is as follows. The feedwater line breaks, causing the RCS to be overcooled as the faulted SG blows down. This results in a short term power excursion that is terminated by a reactor trip. Once the reactor has tripped, the primary system continues to cool down from heat transfer to the faulted SG. Once the faulted SG empties, termination of MFW to the unfaulted SGs may cause a long-term primary system heatup.

The limiting break is located between the final feedwater line check valve and the SG inlet nozzle. Because of the SG design at Robinson, the feedwater system pipe break is considered to be bounded from an overcooling standpoint by the main SLB. The NRC staff therefore compared the proposed method to the Harris UFSAR analysis to determine the acceptability of the assumptions for reactor trips, system availability, initial conditions, and boundary conditions.

The reactor normally trips on OTΔT, high pressurizer pressure, or low SG water level. The OTΔT and high pressurizer pressure trips are not currently credited in the Harris UFSAR analysis; however, the use of these trips is reasonable provided other aspects of the analysis are sufficiently conservative.

The proposed methodology classifies the feedwater system pipe break to be an ANS Condition IV event for both Harris and Robinson. Though as discussed previously the Robinson UFSAR does not provide an analysis of the feedwater system pipe break, this classification is consistent with the Harris UFSAR. The acceptance criteria are to ensure that the dose limits are not exceeded, that the AFW system is capable of providing adequate long-term core cooling, and that the peak primary system pressure is kept below 120 percent (for Harris, consistent with the plant UFSAR for a double-ended guillotine break of the MFW line) or 110 percent (for Robinson) of design pressure, and are appropriate considering the event classification.

Three transient analysis cases, (1) short-term core cooling (which evaluates DNBR to ensure that assumptions for fuel failure from the dose analysis are not exceeded), (2) long-term core cooling (which evaluates the adequacy of long-term core cooling provided by the AFW system), and (3) peak primary pressure, are conducted separately because they require different limiting assumptions. The MDNBR is analyzed using the SCD methodology. All three cases are analyzed with offsite power both maintained and lost to determine the limiting condition.

Peak secondary pressure is considered to be bounded by the loss of external load or turbine trip, depending on the plant, and is not analyzed. The immediate initial response of the feedwater line break is to quickly and substantially depressurize the secondary system. The long-term heatup only occurs well after the reactor trip, when the only heat being produced is from decay heat and any RCPs that are still running. Because the MSSVs are designed to remove this heat load, secondary system pressurization does not occur. The NRC staff finds this justification that the secondary system pressurization is bounded acceptable.

The short-term core cooling and peak primary pressure cases use BOC core physics parameters. As discussed in the response to RAI 29 (Reference 4), BOC physics parameters are acceptable for evaluating the RCS heatup aspects of the feedwater system pipe break transient. If the analysis is performed to evaluate the RCS cooldown aspects of the transient, most-negative MTC and least negative DTC would be used. This is acceptable to the NRC staff because BOC parameters are appropriate for the heatup case and EOC parameters are appropriate for the cooldown case. The long-term core cooling case uses EOC parameters. This is acceptable because the decay heat drives the response after the reactor trip, and EOC parameters will increase the decay heat. Accordingly, the NRC staff considers RAI 29 resolved.

The short- and long-term core cooling cases assume pumped SI is minimized. This is acceptable because it will minimize core cooling. The peak primary pressure case assumes maximized pumped SI. This is acceptable because it will increase the RCS inventory and exacerbate the primary system pressure response.

NRC staff reviewed the other initial and boundary conditions for the three transient analyses and found them to be acceptable. The differences between the initial and boundary conditions in the current AOR at Harris are minor. The system availability assumptions for pressurizer heaters, spray, and PORVs, as well as main steam PORVs and steam dumps, are reasonable for the identified transient scenarios and will tend to exacerbate the system response. The limiting single failure is assumed to be in the AFW system, and the considerations discussed in Section 3.5.9 of this SE apply.

The event is considered to be ended when the operator terminates pumped SI and assumes control of AFW. At Harris, these actions are assumed to occur within 30 minutes (1800 seconds), consistent with the current plant licensing basis. At Robinson, termination of pumped SI is considered to be unnecessary because the shutoff head of the HHSI pumps is well below the operating RCS pressure. This is acceptable to the NRC staff.

However, the feedwater line break event in the current Robinson licensing basis is assumed to be bounded by the SLB. The licensee's response to RAI 30 (Reference 5) stated that the bounding nature of the SLB means that the operator action to isolate AFW to the faulted SG at 10 minutes is implicitly credited for the feedwater line break. In response to RAI 30.1 (Reference 6), the licensee stated that cycling AFW flow (as originally discussed in DPC-NE-3009) would be considered a new operator action that would be evaluated under 10 CFR 50.59 to determine if the change could be made without further NRC review and approval.

As such, the licensee proposed that the operator action to isolate AFW to the faulted SG would be modeled at 10 minutes in the event that a Robinson feedwater line break analysis is performed. The licensee stated in response to RAI 30.1 (Reference 6) that the applicability of this operator action is based on the overall similarity of the plant's short-term response to the feedwater line break event to the short-term response to the main SLB event. The licensee concluded that the operator action to isolate AFW to the faulted SG at 10 minutes is acceptable because of this similarity, because the procedures invoke the operator action based on SG pressure (for which the response is very similar between a SLB and feedwater line break), and because simulator validation has shown that the action can be performed within 5 minutes for the main SLB. The licensee stated that the faulted SG would be identified about two minutes later in the feedwater line break than the main SLB; this still places the termination of AFW within the 10 minute action time for the SLB. The licensee additionally stated in the RAI 30.1

response that it was not planned to credit the operator action for any other event, including (but not limited to) overheating or overpressurization events. The NRC staff finds this justification to be adequate, and determined that the 10 minute time to isolate AFW at Robinson in the feedwater line break analysis is acceptable. Accordingly, the NRC staff considers RAIs 30 and 30.1 resolved. If system changes are made that result in a response that differs significantly from the main SLB the NRC staff would no longer consider the 10 minute time to isolate AFW based on the SLB analysis to be appropriate.

Overall, the NRC staff determined that the licensee's approach to analyzing the feedwater line break event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, single failure assumptions, and assumed operator actions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.11.1 Harris Feedwater Line Break Benchmark (DPC-NE-3008)

The Harris feedwater line break benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008 without modification. Two cases were analyzed in the benchmark: one with a LOOP, and one where offsite power was maintained. For consistency with the AOR, the licensee only credited the low-low SG water level signal for the reactor trip. However, the licensee noted that differences in key models between RETRAN-3D and ANF-RELAP (used for the AOR), especially the critical flow model, may contribute to differences in system transient responses.

For the case where offsite power is maintained, the primary system response is very similar between the benchmark and AOR. Minimum system pressure in the RETRAN-3D analysis is slightly lower and occurs slightly later than in the AOR. The peak cold leg temperature for the intact loops, which occurs around 20 seconds, is slightly lower than in the AOR; in the faulted loop over this same period, the cold leg temperature is higher than the AOR. This is judged by the NRC staff to be the result of differences in SG modeling between the AOR and the benchmark. These differences would also result in differences in the SG pressure response, where the peak secondary pressure in the benchmark analysis is substantially lower in both the faulted and intact loops than in the AOR.

SG level in the faulted and intact loops is substantially different between the benchmark and the AOR. Level in the faulted SG is initially higher in the benchmark than the AOR, but drops faster and is eventually lower than the AOR before falling off the range of the SG narrow range level indication. In the benchmark analysis, SG level in the intact loops dips after the faulted generator empties, but recovers almost immediately after the turbine trip. This behavior is not observed in the AOR, but as discussed in the licensee's response to RAI 17 (Reference 3), it is likely caused by **[[**

**]]**. Neither phenomenon is expected to contribute significantly to the overall transient response.

For the case where offsite power is lost, the overall response is very similar to the case where offsite power is maintained, with the previously discussed key differences between the benchmark and AOR retained. The primary difference between the LOOP and non-LOOP begins when power is lost at 983 seconds–this is based on a 15 minute delay from the loss of subcooling margin in the RCS that was assumed in the AOR. The transient response following

the LOOP is very similar until the transient is terminated at 1800 seconds, when manual control of AFW is assumed.

The NRC staff reviewed the licensee's benchmark analysis and associated RAI responses and concluded that differences between the results could be appropriately attributed to modeling differences between the benchmark and the AOR, particularly with regard to the SGs. The NRC staff therefore concluded that the benchmark provided an appropriate demonstration of the capabilities of the Harris RETRAN-3D base model. Accordingly, the NRC staff considers RAI 17 (DPC-NE-3008) resolved.

3.5.12 Partial or Complete Loss of Forced Reactor Coolant Flow

The partial and complete loss of forced reactor coolant flow events are initiated by a trip of the RCPs caused by a loss of power to the pump motors. A partial loss of flow is caused by the trip of a single pump; a complete loss of flow is caused by the trip of all three pumps. The methodology classifies these events in a manner that is consistent with the plant licensing bases. A partial loss of flow is not considered at Robinson since it is not part of the licensing basis, but is considered as an ANS Condition II event at Harris. The complete loss of flow is an ANS Condition III event, although the event is treated as a Condition II event at Robinson.

The event is mitigated by reactor trips on low coolant flow, RCP under-frequency, or RCP under-voltage. In the Robinson UFSAR, credit is only taken for the low coolant flow trip; however, all three trips are credited in the Harris UFSAR analysis. All three trips are valid for the analysis and thus the proposed method is acceptable.

Acceptance criteria for the partial loss of flow event at Harris and the complete loss of flow event at Robinson are to ensure that the MDNBR is greater than the 95/95 limit and that peak primary system pressure is less than 110 percent of the design pressure. Acceptance criteria for the complete loss of flow event at Harris are to ensure that the fuel failure assumptions made in generating the dose analysis are not exceeded and that the peak primary system pressure is less than 110 percent of the design pressure.

Core cooling (e.g., the DNBR evaluation) and peak primary pressure analyses are performed separately because they have different assumptions. The MDNBR is analyzed using the SCD methodology. Both core cooling and peak primary pressure analyses are conducted with BOC core physics assumptions. In the partial loss of flow event, one pump trips; in the complete loss of flow event, all pumps trip. **[[**
**]]**. As demonstrated in the response to RAI 32 (Reference 4), the current Harris UFSAR flow coastdown curve is much more conservative than the one assumed in the demonstration analysis; however, both are more conservative than plant measurements. Accordingly, the NRC staff considers RAI 32 resolved.

Peak secondary pressure is considered to be bounded by the loss of load or turbine trip, depending on the plant, and is not analyzed. The loss of flow transient does not result in a significant heatup, and therefore experiences roughly the same transient on the secondary system as a typical reactor trip, which does not present a challenge to the secondary system pressure limits. The loss of load or turbine trip is therefore limiting. The NRC staff finds this justification to be acceptable.

The peak primary pressure evaluation assumes that offsite power is maintained. In response to RAI 31 (Reference 4), the licensee stated that, for the MDNBR evaluation, the status of offsite power has a negligible impact on the results of the complete loss of flow transient. This is because the loop transit time is long compared to the difference between the time of the reactor trip (when offsite power would be lost) and the time of MDNBR. For the partial loss of flow, a LOOP following the reactor trip does result in a more conservative MDNBR; however, the case remains bounded by the complete loss of flow, provided the ANS Condition II criteria of the partial loss of flow are met. Because these assumptions regarding the status of offsite power are reasonable and are likely to find the limiting conditions for peak pressure response and MDNBR, the NRC staff finds them acceptable. Accordingly, the NRC staff considers RAI 31 resolved.

NRC staff reviewed the other initial and boundary conditions for the three transient analyses and found them to be acceptable. The differences between the initial and boundary conditions in the current AORs at Harris and Robinson are minimal, and make the proposed methodology as conservative as or more conservative than the existing UFSAR analyses. They are thus acceptable. No limiting single failure was identified, consistent with Harris and Robinson UFSARs.

3.5.12.1 Harris Complete Loss of Forced Reactor Coolant Flow Benchmark Analysis (DPC-NE-3008)

The Harris complete loss of flow benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008 without modification. The benchmark matched as closely as possible all of the AOR inputs, including the assumed flow coastdown curve. As will be discussed in Section 3.5.12.2 of this SE, the primary differences between the DPC-NE-3008 benchmark analysis and the DPC-NE-3009 demonstration analysis are the initial pressurizer level and the flow coastdown curve. Neither of these parameters has a significant impact on the transient and the trends are almost identical (though exact values differ somewhat).

Overall, the primary system response is very similar between the benchmark and the AOR. In the benchmark, peak pressurizer pressure is slightly lower than the AOR, and pressurizer level is correspondingly slightly lower. Core exit temperature is slightly higher than the AOR – it is judged by the NRC staff that this is at least partially a result of where the core exit temperature is defined, as discussed previously in Section 3.5.10.1 of this SE. Otherwise, the primary system response is nearly identical between the AOR and the benchmark. The NRC staff therefore concluded that the benchmark provided an appropriate demonstration of the capabilities of the Harris RETRAN-3D base model.

3.5.12.2 Harris Complete Loss of Forced Reactor Coolant Flow Demonstration Analysis (DPC-NE-3009)

NRC staff reviewed the specific assumptions for the Harris complete loss of forced reactor coolant flow demonstration analysis presented in Section 6.3 of DPC-NE-3009. The assumed initial and boundary conditions were found to conform to the method discussed in Section 3.5.12 of this SE.

At the initiation of the transient, all three RCPs trip and begin to coast down. At 0.6 seconds a reactor trip signal is generated, and control rods begin to insert at 1.2 seconds. Loop mass flow

rate decreases throughout the transient. The MDNBR is reached as rods fall into the reactor and pressurizer level, pressurizer pressure, and primary system temperature are increasing.

The transient proceeds as would be expected and aligns closely with the UFSAR AOR, though there are differences in initial conditions. The total increase in pressurizer level and pressure is similar, even though the pressurizer level assumed in the licensee's analysis is lower than the AOR. Maximum hot leg temperature is consistent. The one area where the demonstration analysis differs slightly from the AOR is the flow coastdown curve, which is much more conservative in the AOR than in the analysis under review. However, as discussed above and in the response to RAI 32 (Reference 4), both flow coastdown curves are more conservative than the plant measurements presented.

The NRC staff reviewed the transient response and found that it met with expectations, and aligned closely with the UFSAR analysis aside from minor differences already identified.

3.5.13 Reactor Coolant Pump Shaft Seizure (Locked Rotor) or Shaft Break

The locked rotor event is initiated by the instantaneous seizure of an RCP or failure of a RCP shaft, which causes flow in the affected loop to rapidly decrease. The initiating event is modeled as an instantaneous decrease in the RCP speed to zero. The transient is mitigated by RPS trips on low RCS flow.

The DPC-NE-3009 methodology classifies the locked rotor event as an ANS Condition IV transient for both Harris and Robinson, consistent with the plant UFSARs, and the corresponding acceptance criteria are applied. Acceptance criteria are to ensure that assumptions made in generating the dose analysis are not violated and that peak primary system pressure remains below 120 percent or 110 percent of design value (Harris and Robinson, respectively). Core cooling (e.g., MDNBR evaluation) and peak primary pressure analyses are performed separately because they have different assumptions. The MDNBR is analyzed using the SCD methodology. BOC core physics assumptions are used.

Peak secondary pressure is considered to be bounded by the loss of load or turbine trip, depending on the plant, and is not analyzed. The locked rotor event is primarily analyzed due to DNB concerns that result from the sudden loss of forced reactor coolant flow. The transient does not result in a significant heatup, and any heatup is quickly terminated by a reactor trip. The secondary system therefore experiences roughly the same transient as a typical reactor trip, which does not present a challenge to the secondary system pressure limits. The loss of load or turbine trip is therefore limiting. The NRC staff finds this justification to be acceptable.

Both core cooling and peak primary pressure cases are analyzed with and without offsite power lost to determine the limiting condition. For cases where offsite power is maintained, the unaffected RCPs continue to run. For cases where offsite power is lost, power is lost coincident with the turbine trip, which causes the unaffected RCPs to begin to coast down.

The licensee stated that the VIPRE-01 evaluation of DNBR uses the fuel pin heat conduction model to determine the fuel centerline temperature. As discussed in the response to RAI 33 (Reference 4), **[[**

**]]**.

NRC staff reviewed the other initial and boundary conditions for the RCP seizure or shaft break analyses and found them to be acceptable. The differences between the initial and boundary conditions in the current AOR at Harris and Robinson are minimal (though Harris only analyzed the DNB case for the locked rotor event). The differences observed by the NRC staff (such as high versus nominal initial pressurizer level, high versus nominal fuel temperature, etc.) will serve to make the proposed methodology more limiting than the current AOR. No limiting single failure was identified, consistent with Harris and Robinson UFSARs.

The NRC staff determined that the licensee's approach to analyzing the RCP shaft seizure or shaft break event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.13.1 Robinson Locked Rotor Benchmark Analysis (DPC-NE-3008)

The Robinson locked rotor benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008 without modification. The benchmark matched as closely as possible all of the AOR inputs. A separate VIPRE-01 calculation was performed using the system response data from RETRAN-3D to determine the timing of MDNBR.

Overall, the system response is very similar between the benchmark and the AOR. The chief difference is in the pressurizer and core exit pressure response, where the curves look fairly different between the two analyses after the time of the MDNBR. As discussed in the licensee's response to RAI 20 (Reference 3), the difference in pressurizer pressure response is believed to be the result of differences in **[[**

**]]**. The difference in core exit pressure response is believed by the licensee to be the result of surge line pressure drop modeling differences – this is because steady-state calculations show good agreement in the pressure difference between the pressurizer and core. This is demonstrated well by the initial condition of core exit pressure and pressurizer pressure shown in Figure 4.3-55 of DPC-NE-3008. **[[**

**]]**. However, the exact pressure response, though different between the benchmark and the AOR, has a minimal impact on both the timing and value of the MDNBR. The NRC staff therefore concluded that the benchmark provided an appropriate demonstration of the capabilities of the Robinson RETRAN-3D base model. Accordingly, the NRC staff considers RAI 20 (DPC-NE-3008) resolved.

3.5.13.2 Harris Locked Rotor Demonstration Analysis (DPC-NE-3009)

NRC staff reviewed the specific assumptions for the Harris locked rotor demonstration analysis presented in Section 6.4 of the methodology report. The assumed initial and boundary conditions were found to conform to the method, as discussed in Section 3.5.13 of this SE.

The affected RCP is chosen on Loop 2, which has the pressurizer surge line connected to it. The licensee stated that the choice of the affected loop has a negligible impact on the transient results. In the response to RAI 34 (Reference 4), a sensitivity study was presented that

changed the affected loop to Loop 3. In moving from Loop 2 to Loop 3 there was a minor increase in core exit pressure (on the order of 10 psi), and a difference in MDNBR below the reported level at 0.001. The MDNBR occurred at 3.9 seconds in both cases. The NRC staff is thus satisfied that the choice of the affected loop has a negligible impact on the transient response. Accordingly, the NRC staff considers RAI 34 resolved.

At the transient initiation, RCP speed instantaneously drops to zero. Flow in the affected loop quickly drops and soon becomes negative, while flow in the unaffected loops increases to slightly above 100 percent. The reactor trip system trip setpoint on low RCS flow is reached almost instantaneously. Primary system temperatures, pressurizer level, and pressurizer pressure all increase. Core power remains relatively constant until control rods begin to insert at just past 1 second into the transient.

The major difference between the demonstration analysis and the AOR is that the AOR assumes a positive MTC while the demonstration analysis assumes an MTC of zero. Thus, in the AOR, when the RCS begins to heat up as the loop flow in the affected loop slows down and reverses, there is a positive reactivity addition and core power increases until the reactor trips. The peak core average surface heat flux is slightly higher than the initial. Aside from this modeling difference between the demonstration analysis and the AOR, the demonstration analysis proceeds as would be expected and is found to be acceptable by the NRC staff.

3.5.14 Uncontrolled Rod Control Cluster Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition

The uncontrolled RCCA bank withdrawal from subcritical event is initiated by the uncontrolled withdrawal of a control rod bank, starting from a subcritical or low power condition. The event is mitigated by RPS trips on power range high neutron flux (low setting), power range high neutron flux rate (which does not exist at Robinson), or the high pressurizer pressure trip. The DPC-NE-3009 methodology classifies this event as an ANS Condition II transient for both Harris and Robinson, consistent with the plant UFSARs, and applies the corresponding acceptance criteria. Acceptance criteria are to ensure that the MDNBR is greater than the 95/95 limit, that the CFM limit is not exceeded, and that peak primary pressure is less than 110 percent of the design value.

Peak secondary pressure is considered to be bounded by the loss of load or turbine trip, depending on the plant, and is not analyzed. Though the uncontrolled RCCA bank withdrawal from low power does result in large increases in neutron power, the event is brief enough and terminated quickly enough that excessive energy deposition does not occur. Thus, significant secondary system pressurization does not occur prior to the reactor trip. This is in contrast to the turbine trip and loss of load, where the reactor continues to operate at power for a period after the secondary system is isolated from the heat sink. The loss of load and turbine trip are therefore more limiting for the secondary pressure response. The NRC staff finds this justification to be acceptable.

Core cooling and peak primary pressure analyses are performed separately because they have different assumptions. The core cooling case evaluates both the DNBR and the margin to the CFM limit. The proposed methodology analyzes the MDNBR using the SCD methodology, which, as discussed in the response to RAI 35 (Reference 4), is likely to be applicable at the time of MDNBR. If the limiting DNBR statepoint falls out of the range of the SCD methodology, the methods discussed in Section 3.5 of this SE (or the response to RAI 15 (Reference 4)) will

be used to handle the analysis. In response to RAI 36 (Reference 5), the licensee stated that for reactivity insertion events that present a challenge to both DNB and CFM limits, a spectrum of reactivity insertions and rates are considered. **[[**

**]]**.

BOC core physics assumptions are used, consistent with the plant UFSARs. For both the core cooling and peak pressure cases, the simplified SG secondary side RETRAN-3D nodalization is used. This is consistent with the intent of this nodalization, which makes initialization from a low/zero power condition possible.

The core cooling case is initialized with initial and boundary conditions that maximize the secondary system pressure and minimize the primary system pressure. This will minimize the primary-to-secondary heat transfer for the duration of the event, and is conservative. Two RCPs are assumed to be in operation; this is consistent with the Harris and Robinson UFSARs. The initial and boundary conditions are thus acceptable.

The peak primary pressure case is initialized with initial and boundary conditions that maximize the primary system pressure. Considering the purpose of the scenario is to analyze the peak primary pressure, this is conservative. Cases are run with either two pumps running or three pumps running, to determine the limiting condition. Because the limiting condition will be evaluated in the analysis, the initial and boundary conditions for the peak primary pressure case are acceptable.

The methodology report states that no single failure will impact the results of this event. This is consistent with the Harris and Robinson UFSARs.

The NRC staff determined that the licensee's approach to analyzing the uncontrolled RCCA bank withdrawal from subcritical event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. Accordingly, the NRC staff considers RAIs 35 and 36 resolved.

3.5.15 Uncontrolled Rod Control Cluster Assembly Bank Withdrawal at Power

The uncontrolled RCCA bank withdrawal at power (URBWAP) event is initiated by the uncontrolled withdrawal of a control rod bank at a reactor power level of 2 percent or greater. The event is mitigated by reactor trips on OTΔT, overpower differential temperature (OPΔT), or power range high neutron flux (high setting). The licensee noted that the analysis of the URBWAP was specifically designed to challenge these RPS trips. The DPC-NE-3009 methodology classifies this event as an ANS Condition II transient for both Harris and Robinson, consistent with the plant UFSARs, and applies the corresponding acceptance criteria. The acceptance criteria are to ensure that the MDNBR is greater than the 95/95 limit, that the CFM limit is not exceeded, and that peak primary pressure is less than 110 percent of the design value.

Peak secondary pressure is considered to be bounded by the loss of load or turbine trip, depending on the plant, and is not analyzed. Over the course of the URBWAP transient, there is a substantial increase in heat generation, and SGs eventually increase in pressure until

reaching the SG PORV or safety valve setpoint. However, the licensee does not consider the secondary system pressure response to be a concern because the heat sink is not isolated from the SGs until after the reactor trip. As discussed in other sections of this SE, the loss of load or turbine trip is limiting because the SGs are isolated from the heat sink while the reactor continues to operate at power. The total demand on the SG PORVs or safety valves will be lower in the URBWAP than in the turbine trip or loss of load. The NRC staff finds this justification to be acceptable.

Core cooling and peak primary pressure analyses are performed separately because they require different limiting assumptions. The core cooling case evaluates both the DNBR and the margin to the CFM limit. As with the uncontrolled RCCA bank withdrawal from a subcritical or low power condition, both DNB and CFM analyses are performed using a spectrum of cases to find the limiting condition, which may differ for DNB and CFM evaluations, and the DNBR is analyzed using the SCD methodology.

Core cooling cases are run with a variety of initial conditions and core physics parameters to determine the limiting scenario:

- Initial power is evaluated in a range from low power to HFP
- Core physics is evaluated at BOC and EOC
- Primary system pressure is evaluated at low and high values
- Secondary system pressure is evaluated at low and high values
- Reactivity insertion rate is evaluated in a range from low to maximum

All of these cases are consistent with the Harris and Robinson UFSARs. The primary and secondary system availabilities are selected to minimize the DNBR. The initial and boundary conditions for the core cooling case are therefore acceptable.

Peak primary pressure cases are run assuming low initial power, BOC core physics, a maximized reactivity insertion rate, and high secondary pressure. Primary pressure is evaluated. These assumptions tend to maximize the power and temperature excursions and thus exacerbate the primary system pressure response. The primary and secondary system availabilities are selected to maximize the primary system pressure. The initial and boundary conditions for the peak primary pressure case are therefore acceptable.

No single active failure was identified in the methodology that could impact the transient. This is consistent with the Harris and Robinson UFSARs.

The NRC staff determined that the licensee's approach to analyzing the URBWAP event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

#### 3.5.15.1 Robinson Uncontrolled Rod Control Cluster Assembly Bank Withdrawal at Power Benchmark Analysis (DPC-NE-3008)

The Robinson URBWAP benchmark analysis uses the base RETRAN-3D model presented in DPC-NE-3008 without modification. The benchmark matched as closely as possible all of the AOR inputs. The MDNBR was not analyzed in the benchmark analysis.

Overall system response is very similar between the benchmark and the AOR. The indicated core power response is identical. Indicated primary temperatures are slightly lower in the benchmark than in the AOR; because the OTΔT and power range high flux trip setpoints are nearly coincident, the slight difference in indicated temperature may be responsible for the switch from OTΔT to power range high flux trip from the AOR to the benchmark. Peak pressurizer pressure is reached at nearly identical timing in the AOR and the benchmark, but the peak pressure itself is slightly lower in the benchmark. Pressurizer level is correspondingly slightly higher as well.

The drop in pressurizer pressure following the trip is faster in the benchmark than in the AOR. As discussed in the licensee's response to RAI 21 (Reference 3), this is likely to be the result of **[[**

**]]**. However, because the MDNBR occurs almost immediately after RCCA insertion is initiated, the licensee judged that the difference in modeling would have a negligible impact on the MDNBR. The licensee concluded, therefore, that the modeling of **[[**

**]]** in the base model as described in DPC-NE-3008 is adequate and will be retained for consistency with other approved Duke Energy RETRAN models for Catawba, McGuire, and Oconee.

The NRC staff reviewed the licensee's assessment of the pressure response and determined that it was acceptable. While the NRC staff believes that additional volumes in the pressurizer would provide improved accuracy overall, the accuracy provided in the base model is sufficient for the transients of interest. The model is not sufficient for transients where rapid decreases in RCS volume will be encountered (i.e., LOCA); however, since it is not proposed for use in these scenarios the model as proposed is acceptable. The NRC staff therefore concluded that the benchmark provided an appropriate demonstration of the capabilities of the Robinson RETRAN-3D base model. Accordingly, the NRC staff considers RAI 21 resolved.

3.5.15.2 Robinson Uncontrolled Rod Control Cluster Assembly Bank Withdrawal at Power Demonstration Analysis (DPC-NE-3009)

NRC staff reviewed the specific assumptions for the Robinson uncontrolled RCCA bank withdrawal demonstration analysis presented in Section 5.4.2 of DPC-NE-3009. The assumed initial and boundary conditions were found to conform to the method, as discussed in Section 3.5.15 of this SE.

The licensee stated that a variety of core cooling analyses were performed at BOC/HFP conditions with a range of reactivity insertion rates. The reactor trip that controls the event changes depending on the reactivity insertion rate, with low reactivity insertion rate cases tripping on OTΔT, moderate reactivity insertion rates tripping on OPΔT, and high reactivity insertion rates tripping on power range neutron flux (high setting). The limiting case was predicted to be near the transition from the OTΔT-controlled region to the OPΔT-controlled region. The analysis presented as the demonstration had a reactivity insertion rate of 1.0 percent mille per second (pcm/s), which places it at or near this transition point.

The rod bank is withdrawn from the core at a reactivity insertion rate of 1.0 pcm/s, causing core power and primary system temperatures to gradually increase. As temperature increases, level in the pressurizer swells. Pressurizer pressure remains relatively constant for the initial portion of the transient, as a result of pressurizer sprays, then increases as reactor differential temperature approaches the OTΔT setpoint.

The transient in the demonstration analysis proceeds as expected, and is almost identical to the URBWAP transient from the Robinson UFSAR with a reactivity insertion rate of 1.0 pcm/s - with the exception that the UFSAR analysis goes through the transient almost twice as fast. The NRC staff attributes this to the reactivity coefficients assumed in the analysis: the AOR uses a positive MTC, while the demonstration analysis assumes an MTC of zero. The DTC assumed in the demonstration analysis is also more negative than that of the AOR. The result is that the overall change in core power occurs faster due to reactivity feedback. The NRC staff finds these differences to therefore be reasonable and the demonstration analysis to be acceptable.

3.5.16 Dropped Full-Length Rod Control Cluster Assembly or Rod Control Cluster Assembly Bank

The dropped rod event is initiated by one or more control rods dropping into the core at full power. The outcome of the event changes depending on whether rod control is in automatic or manual. With rod control in automatic, the mismatch between reactor power and the reference temperature causes the rod control system to withdraw the regulating bank of control rods. The positive reactivity insertion from rod withdrawal, combined with decreasing reactor temperature and a negative moderator temperature coefficient, can cause the reactor to return to or exceed the initial power level. With rod control in manual, the temperature decrease from the dropped rod (or bank) combined with a negative MTC can still result in a power increase. The event is mitigated by reactor trips on low pressurizer pressure, high flux, high negative flux rate (which does not exist at Robinson), OTΔT, or OPΔT. The proposed methodology classifies this event as an ANS Condition II event for Harris and Robinson. This is consistent with the plant UFSARs. The acceptance criteria are to ensure that the MDNBR is greater than the 95/95 limit and that the CFM limit is not exceeded. The MDNBR is evaluated using the SCD methodology.

Pressure limits are not challenged by this event and are not analyzed. Though the initial phase of the transient is an immediate decrease in reactor power, the combination of reactivity feedback and rod control motion may cause the reactor to return to or exceed the initial power level. However, significant pressurization of the primary or secondary systems does not occur because the secondary system is not isolated from the heat sink until after the trip. This is in contrast to the limiting events, the loss of load or turbine trip (depending on the plant), where the reactor continues to operate at full power after the secondary system is isolated from the heat sink, resulting in heat input to the secondary system that cannot be removed and producing a limiting pressure response. The NRC staff finds this justification to be acceptable.

The analysis method is as follows. RETRAN-3D is used to generate conditions for the statepoint corresponding to the time of the MDNBR. SIMULATE-3 is used to calculate the post-drop core power distributions based on the RETRAN-3D conditions. Thermal-hydraulic and system conditions from RETRAN-3D and core power distributions from SIMULATE-3 are used as inputs to VIPRE-01 to calculate the MDNBR. **[[**

**]]**. These adjustments are based on calculations performed using the

SAS2H/ORIGEN module of SCALE and MCNP, in the same manner as in previously approved Duke Energy methodologies (e.g., DPC-NE-3005, DPC-NE-3001, Revision 1). The adjustments are therefore acceptable to the NRC staff.

Two separate analyses are performed, with rod control in automatic and manual. With rod control in automatic, the event is evaluated at **[[**
**]]**. The limiting single failure is **[[**
**]]** - this is consistent with the analysis in the Harris UFSAR, though the Robinson UFSAR does not assume any single failure. The MTCs and DTCs are selected to be the least negative for the time in life, and the rod worth available for withdrawal is maximized. The purpose of these assumptions is to maximize the power overshoot when the rods are withdrawn. With rod control in manual, the event is primarily a cooldown event caused by the dropped rod. No single failure acts to make the transient worse. EOC conditions are thus considered because they will maximize the positive reactivity addition.

The NRC staff reviewed the proposed assumptions for thermal-hydraulic initial and boundary conditions and system availability. They are consistent with the SCD methodology, or will otherwise tend to exacerbate the response.

The NRC staff determined that the licensee's approach to analyzing the dropped RCCA or RCCA bank event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

3.5.17 Withdrawal of a Single Full-Length Rod Control Cluster Assembly

The withdrawal of a single full-length RCCA event is initiated by the uncontrolled withdrawal of a single control rod at power. The main difference between the single rod withdrawal event and the uncontrolled rod bank withdrawal event (discussed in Section 3.5.15 of this SE) is that the single rod withdrawal is a highly localized event that results in severe radial power peaking. Mitigation is provided by reactor trips on OTΔT and OPΔT.

The DPC-NE-3009 methodology classifies this event as an ANS Condition III event for Harris and Robinson. This is consistent with the plant UFSARs. The corresponding acceptance criterion is applied, which is to ensure that the radioactive release does not violate the assumptions made in the dose analysis. The analysis tallies the number of fuel pins that violate the DNBR 95/95 limit and the percentage of fuel that violates the CFM limit. The proportion of the fuel that violates these limits is assumed to fail and is compared to the dose analysis assumptions to ensure acceptability.

The licensee considers peak primary system pressure to be bounded by the URBWAP event. The overall system response to the withdrawal of a single RCCA is similar to that of the URBWAP, but with an increased local peaking. The demonstration analyses from Sections 6.5 and 6.6 of DPC-NE-3009 show that the secondary pressure response of the URBWAP is much more severe than the single RCCA withdrawal. The NRC staff therefore finds this justification to be acceptable.

Secondary system pressure is considered to be bounded by the loss of load or turbine trip, depending on the plant. Over the event, there is an increase in core power and heat flux, and SG eventually increase in pressure until reaching the SG PORV or safety valve setpoint.

However, as in the URBWAP, the licensee does not consider the secondary system pressure response to be a concern because the heat sink is not isolated from the SGs until after the reactor trip. As discussed in other sections of this SE, the loss of load or turbine trip is limiting because the SGs are isolated from the heat sink while the reactor continues to operate at power. The total demand on the SG PORVs or safety valves will be lower in the withdrawal of a single RCCA than in the turbine trip or loss of load. The NRC staff finds this justification to be acceptable.

The MDNBR is evaluated using the SCD methodology. BOC core physics are used. Evaluations are performed to determine the limiting assumptions for a variety of parameters, including core power, primary and secondary system pressures, and reactivity insertion rate. The range of core powers considered is the same as for the URBWAP transient. Availability of pressurizer pressure and level control systems is also evaluated, because the reactivity insertion rate may affect the limiting condition. Because the limiting condition will be considered, these evaluations are acceptable. The limiting single failure is **[[ ]]**. This appears to be more conservative than the Robinson and Harris UFSARs, which assume no limiting single failure, and is thus acceptable.

The licensee explained the DNB and CFM evaluations in more detail in the response to RAI 37 (Reference 4). The system thermal-hydraulic response is evaluated using RETRAN-3D, and **[[ ]]** are used to develop maximum allowable radial peaking (MARP) limits. These MARP limits translate the DNBR limit into a local power distribution limit, given the local thermal-hydraulic conditions, and are discussed in more detail in DPC-NE-2004 (Reference 43) and the NRC staff's SE for adoption of DPC-NE-1008, DPC-NF-2010, and DPC-NE-2011 at Harris and Robinson (Reference 44). SIMULATE-3 is then used to generate core power distributions for comparison to MARP limits and CFM limits. The initial conditions for these power distributions include **[[ ]]**.

The NRC staff determined that the licensee's approach to analyzing the withdrawal of a single RCCA event was appropriate. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. Accordingly, the NRC staff considers RAI 37 resolved.

3.5.17.1 Robinson Withdrawal of a Single Full-Length Rod Control Cluster Assembly Demonstration Analysis

The NRC staff reviewed the specific assumptions for the Robinson withdrawal of a single full length RCCA demonstration analysis presented in Section 5.4.4 of DPC-NE-3009. The assumed initial and boundary conditions were found to conform to the method, as discussed in Section 3.5.17 of this SE. The initial and boundary conditions are similar to those of the URBWAP demonstration analysis, with assumptions tweaked slightly to increase primary system pressure, minimize secondary system pressure, and overall result in increased primary to secondary heat transfer, which delays the OTΔT trip. The reactivity insertion rate, as

discussed in the response to RAI 38 (Reference 4), was assumed to be 0.95 pcm/s. This value was selected as the limiting value based on a sensitivity study that examined reactivity insertion rates from 0.1 pcm/s to 2.35 pcm/s.

Pressurizer control systems were evaluated differently between the bank and single rod withdrawal analyses. In the URBWAP demonstration analysis, heaters are disabled while sprays and PORVs are available; in the single rod withdrawal analysis the heaters and PORVs are available while the sprays are disabled. These assumptions for the rod withdrawal analysis are designed to maximize the primary system pressure.

The transient proceeds in a very similar manner to the URBWAP demonstration analysis. From a system transient perspective, the primary difference is caused by the different modeling of the pressurizer systems. As the coolant swells due to the power increase and resulting RCS temperature increase, the pressurizer level increases. The pressure increases to the PORV setpoint at approximately 46 seconds, and soon after this the backup heaters are turned on to control the pressurizer level.

Though the withdrawal of a single full-length RCCA transient is analyzed in the Robinson UFSAR, no plots of the system transient are provided. However, in the AOR, the highly localized peaking that results from the rod withdrawal is predicted to cause DNB in the immediate area near the withdrawn rod. This is not predicted in the demonstration analysis, which states that the MDNBR was found to be 1.571. However, as discussed in Section 6.6 of the DPC-NE 3009 and the response to RAI 37 (Reference 4), this was based on a simplified evaluation of DNB that did not reflect the local spatial effects from SIMULATE-3. The licensee clarified in the RAI response that future cycle-specific evaluations will not use this simplified approach and the DNBR limit may be approached or exceeded. This is acceptable to the NRC staff provided the acceptance criterion–that the dose analysis assumptions are not violated-continues to be met. Accordingly, the NRC staff considers RAIs 37 and 38 resolved.

3.5.18 Static Misalignment of a Full-Length Rod Control Cluster Assembly

The static misalignment of a full-length RCCA event is initiated by a control rod that is statically out of alignment with the rest of its bank due to the malfunction of a control rod drive mechanism. The misaligned rod may be higher or lower than the rest of the bank. A subgroup of rods in the bank may also be postulated to be misaligned, rather than just a single rod. This potentially results in a less asymmetric power distribution, and the result is compared to the result of the single misaligned rod case to determine which is more limiting.

The DPC-NE-3009 methodology classifies this event as an ANS Condition II event, consistent with the plant UFSARs. The corresponding acceptance criteria are applied, which are to ensure that the DNBR remains above the 95/95 limit and that the CFM limit is not exceeded. Peak primary and secondary system pressure responses are considered to be bounded by other events and are not analyzed; this is because the reactor is at steady-state conditions and there is no transient power, pressure, temperature or flow conditions. The primary concern is increased peaking factors, which may lead to a loss of DNBR margin. The MDNBR is evaluated using the SCD methodology.

The steady-state three-dimensional power distribution that results from static misalignment of a single control rod (or group of control rods) is evaluated to confirm that DNB and CFM limits are not exceeded. The licensee stated in response to RAI 39 (Reference 4) that it does not matter

which code is used to perform this analysis, as long as it is NRC approved for performing three-dimensional neutronic analyses and is applicable to the Harris and Robinson cores. The NRC staff finds this to be acceptable. Axial power shapes are considered from the spectrum of shapes allowed by the power-dependent AFD limits. Accordingly, the NRC staff considers RAI 39 resolved.

Two cases are considered for the single control rod misalignment: the full insertion of a rod from control bank D with the rest of the bank within the power-dependent RILs, and the full withdrawal of a rod from control bank D with the rest of the bank at the power dependent RIL. The NRC staff agree that these represent the limiting conditions with regard to single rod misalignment. Misalignment of a subgroup is analyzed using the same approach as for a single rod.

The NRC staff determined that the licensee's approach to static misalignment of an RCCA event was appropriate. The acceptance criteria, initial and boundary conditions, and overall method of analysis are consistent with the Harris and Robinson licensing bases or are otherwise acceptable.

### 3.5.19 Chemical and Volume Control System Malfunction that Decreases the Boron Concentration of the Reactor Coolant

The chemical and volume control system (CVCS) malfunction that decreases the boron concentration of the reactor coolant event is an accidental dilution transient caused by a scenario in which the boric acid concentration of the reactor coolant makeup water from the CVCS is lower than that of the existing reactor coolant. The event is mitigated by manual operator action. The purpose of the analysis is to determine whether operator action can be accomplished before the reactor reaches a critical condition, within appropriate time limits.

The methodology classifies the event as an ANS Condition II transient. This is consistent with the Harris and Robinson UFSARs. The acceptance criteria are to ensure that the reactor does not become critical within 30 minutes when the plant is in MODE 6, and 15 minutes for other modes. The time considered to be the duration of the event depends on the plant licensing basis, and begins at either the time when the dilution is initiated or when the plant alarm announces an unplanned dilution. At Harris the acceptance criterion for the dilution transient in MODE 5 is based on the alarm timing, and the acceptance criteria for MODES 1-4 are based on the time of dilution initiation. An uncontrolled dilution in MODE 6 at Harris is not analyzed–it is precluded by administrative controls and is not included in the current licensing basis. At Robinson, as discussed in the response to RAI 40 (Reference 5), the acceptance criteria for all modes are based on the time of dilution initiation.

The licensee's response to RAI 41 (Reference 5) provides the methodology used to determine the dilution time. Rather than using a transient system analysis code to model the flow of boric acid into the reactor, the licensee solves a simple differential equation to determine the time rate of change of boric acid concentration in the RCS. This is consistent with previously approved Duke Energy methodologies. The dilution source is conservatively considered to be unborated. The dilution time is a function of the mass of water in the RCS, the mass flow rate of unborated water from the CVCS, and the initial and final boron concentrations. The initial boron concentration is the concentration at dilution initiation or the concentration at alarm annunciation, depending on the plant licensing basis for the event considered; final boron concentration is the critical boron concentration at which shutdown margin is lost. Both boron

concentrations are determined using SIMULATE-3. The reload analysis checks the ratio of the initial to final boron concentration in each mode and compares it to the ratio in the analysis. Both the method of initial safety analysis proposed by the licensee and the reload analysis check were determined to be acceptable to the NRC staff.

The system is sensitive to the assumed initial and boundary conditions. The mass of water in the RCS available for dilution is one of the most significant factors for determining the time available before the reactor reaches a critical condition. For a given dilution flow rate, a smaller RCS mass will result in a shorter time to reach criticality. The licensee proposed to assume a minimum RCS mixing volume (and thus mass) for each mode of operation to minimize the time to reach a critical condition.

- In MODES 1-3, and in MODES 4 or 5 when at least one RCP is running, the volume assumed for mixing includes the reactor vessel and all of the reactor coolant loops, but does not include the pressurizer or pressurizer surge line. This is appropriate because the RCS loops will experience forced mixing flow from the RCPs but there is not expected to be insurge or outsurge causing mixing with the pressurizer surge line or pressurizer.
- In MODE 4 without an RCP running, the dilution volume is the reactor vessel minus the upper head region, the reactor heat removal (RHR) system, and portions of the hot and cold legs between the RHR inlet and outlet connections. This is appropriate because the reactor may be running on a single train of RHR and mixing can only be expected to occur in areas directly affected by RHR operation.
- In MODE 5 and 6, the reactor vessel may be drained down below the top of the coolant loop piping. In this case, the mixing volume is assumed to include the RHR train with the least volume and the portions of the reactor vessel and coolant loops below the minimum water level and between the RHR inlet and outlet connections.

The dilution flow rate used for the calculation is maximized for each mode of operation relative to the capacity of the pumps normally used for makeup in that mode of operation. The expansion of the cooler makeup water as it enters the RCS is accounted for in the calculation and results in a larger volumetric flow rate for dilution.

The NRC staff determined that the licensee's approach to analyzing the dilution event was appropriate, based on the information contained in DPC-NE-3009 and the associated RAI responses. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. Accordingly, the NRC staff considers RAIs 40 and 41 resolved.

3.5.20 Inadvertent Loading of a Fuel Assembly in an Improper Position

This event involves the misloading of one or more fuel assemblies. Misloading is defined in the licensee's methodology as insertion into improper core locations or loading with improper burnable poison rods. The event is considered by the methodology to be an ANS Condition III event for Harris and Robinson, consistent with the plant UFSARs. The concern addressed in the fuel assembly misloading analysis is that a reactivity deviation caused by a misloaded assembly may result in peaking factors that could result in fuel failures that would challenge or exceed dose analysis assumptions. Peak primary and secondary system pressure responses

are considered to be bounded by other events and are not analyzed; this is because the reactor is at steady-state conditions and there is no transient power, pressure, temperature or flow conditions. The primary concern is increased peaking factors, which may lead to a loss of DNBR margin.

As discussed in the licensee's response to RAI 42 (Reference 4), the primary layer of protection against a misloading is administrative controls and checks on fuel assembly loading. However, if these controls fail, additional protection is provided by zero-power physics testing and low--and intermediate-power flux maps. Misload screening criteria are developed on a cycle-specific basis for the low-power flux map, and are designed to ensure that the number of DNB or CFM failures assumed in the plant-specific dose analysis will not be exceeded. Misloads that are undetected by the screening criteria must therefore not invalidate the dose analysis assumptions. The power distributions that would result from undetectable misloadings are compared against the MARP and CFM limits to determine the number of rods expected to fail. If the number of DNB or CFM failures predicted in a given misloading event exceeds the number assumed in the dose analysis, the screening criteria must be revised to ensure that the misloading will be detectable.

The NRC staff notes that it is impossible to review the screening criteria in a report like DPC-NE-3009, because they must be developed on a cycle-specific basis; however, the overall approach of analyzing misloads that would go undetected by screening to ensure that they do not exceed the dose analysis assumptions is acceptable, and the models developed in DPC-NE-3008 and DPC-NE-3009 are appropriate for that analysis. The NRC staff therefore determined that the licensee's overall approach to analyzing fuel misloads is acceptable. Accordingly, the NRC staff considers RAI 42 resolved.

3.5.21 Spectrum of Rod Control Cluster Assembly Ejection Accidents

A CRE is initiated by a failure of the control rod drive mechanism housing, which causes a pressure differential that ejects the control rod from the reactor. The DPC-NE-3009 methodology classifies the CRE as an ANS Condition IV event for Harris and Robinson, consistent with the plant UFSARs. The corresponding acceptance criteria are applied, which are to ensure that the assumptions made in the dose analysis are not exceeded and that the peak primary system pressure remains below 120 percent of the design value. Core cooling analyses are performed to determine the number of fuel pins that exceed the 95/95 DNBR limit and the percentage of fuel that exceeds the CFM limit.

The peak secondary system pressure response is considered to be bounded by the turbine trip or loss of load, depending on the plant, and is not analyzed. For rod ejections that result in an immediate flux-based reactor trip, the same argument as for the uncontrolled rod bank withdrawal from subcritical is applied–the duration of the transient is short enough, and a small enough quantity of energy is deposited, that significant pressurization of the secondary system will not occur. The NRC staff finds this justification to be acceptable. For rod ejections that do not reach the flux-based trips, the reactor trip is delayed and the reactor coolant temperature and pressure increases. However, because the heat sink remains available and is capable of limiting the secondary system pressurization, the event is bounded by the turbine trip and loss of load, where the heat sink is isolated from the SGs while the reactor continues to operate at full power. The NRC staff finds this justification to be acceptable and determined therefore that a secondary system pressure analysis is unnecessary.

3.5.21.1 Acceptance Criteria

The NRC has imposed acceptance criteria designed to avoid gross fuel cladding failure and pellet-cladding mechanical interaction (PCMI). Guidance on rod ejection analysis and the associated original acceptance criteria were published in Regulatory Guide (RG) 1.77, in 1974 (Reference 45); however, since that time, the state of knowledge of fuel rod performance under prompt power excursion conditions has increased significantly. While a revision to this RG is under development (and has been published as Draft Regulatory Guide DG-1327 (Reference 46)), the NRC has published interim acceptance criteria for reactivity initiated accidents (which includes the rod ejection for pressurized water reactors and the rod drop for boiling water reactors) in Appendix B to SRP Section 4.2 (Reference 15).

Fuel failure criteria include:

- For rod ejections initiated from HZP conditions, fuel cladding failure from exposure to high temperatures is to be presumed if peak radial average fuel enthalpy exceeds 170 calories per gram (cal/g) or 150 cal/g, for fuel rods with internal pressures at or below or exceeding system pressures, respectively. This failure criterion corresponds to an exceedance of the DNBR limit for rod ejections initiated from partial or full power conditions.
- Fuel failure from PCMI is presumed if the change in radial average enthalpy exceeds a corrosion-dependent limit provided in SRP Section 4.2 Figure B-1.

Additional criteria must be satisfied to ensure that core coolability can be maintained. These criteria include:

- Peak radial average fuel enthalpy must remain below 230 cal/g.
- Peak fuel temperature must remain below incipient melting conditions.
- Mechanical energy generated as a result of non-molten fuel-to-coolant interaction and fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- Fuel pellet and cladding fragmentation and dispersal and fuel rod ballooning must not cause a loss of coolable geometry.

In response to RAI 43 (Reference 5), the licensee stated that the peak radial average fuel enthalpy would remain below 230 cal/g, and that the peak fuel temperature would remain below melting conditions. The 95/95 DNBR limit will be used to predict the number of pins that experience cladding failure due to DNB for rod ejection initiated from greater than 5 percent of rated thermal power. For HZP cases, the 170 or 150 cal/g criterion discussed above will be used. **[[**

**]]**.

The PCMI failure criterion provided by the NRC is dependent on the fuel cladding corrosion performance. The licensee proposed to demonstrate margin to the PCMI failure criterion based on an acknowledgement that the current fuel in use at Harris and Robinson has AREVA's M5 cladding alloy, which has very good corrosion resistance. In the response to RAI 43 (Reference 5), the licensee provided results of several sample rod ejection calculations **[[**

**]]**. The licensee also then developed prompt fuel enthalpy rise limits, based on Figure B-1 of SRP Section 4.2. The end-of-life oxide thickness calculations used in developing the limits were performed with the COPERNIC code, which has been approved by the NRC for use in analyzing the fuel in use at Harris and Robinson (though approval has not yet been received for use of the code in Harris and Robinson licensing analyses), and assumed **[[** **]]**. These limits ended up as **[[** **]]**.

The values of total fuel enthalpy and prompt enthalpy rise from the demonstration analysis were compared to the core coolability limit of 230 cal/g, the high temperature cladding failure limit of **[[** **]]**. Results are presented in Tables RAI-43-1 and RAI-43-2, and show **[[**

**]]**. Based on the **[[** **]]**, the NRC staff agrees. **[[**

**]]**. Accordingly, the NRC staff considers RAI 43 resolved.

3.5.21.2 Analysis Method

The licensee stated in response to RAI 44 (Reference 5) that the initial application of the analysis will be performed at BOC, EOC, **[[** **]]**. The cycle-specific validation will compare to the limiting cases from this initial analysis to determine the continued acceptability of the analysis and applicability to the cycle in question. The full range of cases will be re-examined for significant changes in fuel design, fuel management, or plant operation, to confirm the acceptability of the current limiting cases or determine new limiting cases (as necessary). This is consistent with SRP Section 4.2 and SRP Section 15.4.8 (Reference 47), which state that a variety of initial power levels and times in life should be considered. The NRC staff considers the licensee's approach acceptable for identifying limiting cases and re-evaluating the limiting cases as needed.

SIMULATE-3K is used to determine the nuclear response to the rod ejection, and calculates the core power level and nodal power distribution. VIPRE-01 is used to evaluate the fuel enthalpy, fuel temperature, and the pin radial peaking that causes the DNBR limit to be exceeded. If a high flux or high flux rate trip is not predicted by SIMULATE-3K, the transient is expected to last considerably longer before being tripped by one of thermal-hydraulic or systems trips (usually OTΔT, OPΔT, or low pressurizer pressure). In that case, RETRAN-3D is used to determine the system response.

3.5.21.2.1 Nuclear Analysis using SIMULATE-3K

As discussed above, SIMULATE-3K is used to evaluate the core response and calculates the core power and nodal power distribution.

Radial geometry is typically modeled using **[[ ]]** radial nodes per fuel assembly. Axial geometry is modeled as appropriate for the given axial characteristics of the fuel. For current fuel designs, 24 axial nodes are considered to be appropriate.

Rods from within control banks C and D are evaluated at HZP conditions, and rods from within control bank D alone are evaluated at HFP conditions. Because the ejected rod worth drives the power excursion, it is conservatively increased in SIMULATE-3K. The effective delayed neutron fraction **[[**

**]]**. The rod is ejected from a fully inserted position to a fully withdrawn position in 0.1 seconds at constant acceleration. This is consistent with the approved methodology documented in DPC-NE-3001.

The initial MTC **[[**

**]]**; alternatively, for BOC cases, the boron concentration is adjusted. Making the MTC less negative (or more positive) will cause less negative (or more positive) reactivity to be inserted when the moderator temperature increases following the rod ejection, and these adjustments are thus conservative.

The DTC is also adjusted to be less negative in such a way that future cores are bounded. To achieve this, **[[**

**]]**. Because the negative reactivity feedback from increased fuel temperature is responsible for limiting the initial power excursion, a less negative DTC will result in less negative reactivity addition and is therefore conservative.

Neither the ejected rod nor the remaining highest worth rod is assumed to fall into the core during the reactor trip. The available rod worth for insertion during a trip is additionally reduced within SIMULATE-3K. As discussed in the licensee's response to RAI 45 (Reference 4), the assumed rod drop time is consistent with the plant TSs.

The limiting single failure is the loss of the highest-indicating excore detector. This is more conservative than the Harris and Robinson AORs, which do not assume any single failures. **[[** **]]**, with a conservative trip delay assumed–this model has been previously used in the NRC-approved DPC-NE-3001 methodology and is thus acceptable to the NRC staff.

The heat conduction model in SIMULATE-3K is used to calculate the temperature distribution within the pin as well as the transport of heat from the fuel, through the gap and cladding, and into the coolant. **[[**

**]]**. In response to RAI 46 (Reference 4), the licensee stated that calculating the Doppler feedback **[[**

**]]**. Reference 8 from DPC-NE-3009, which is the

Studsvik report SSP-04/443 “LWR Core Reactivity Transients; SIMULATE-3K Models and Assessment” (Reference 48), provides various reactivity benchmarks that use **[[** **]]** for calculating Doppler feedback. For the most part, there is good agreement–though the NRC staff notes that they are only computational benchmarks and do not necessarily reflect all of the details of the event that would occur in the reactor. However, the good agreement for a variety of reactivity insertion scenarios coupled with the licensee’s justification gives confidence that basing the Doppler feedback on **[[** **]]** results in an adequate prediction of reactivity. Accordingly, the NRC staff considers RAIs 45 and 46 resolved.

3.5.21.2.2 Pinwise Fuel Temperature and Enthalpy Calculations using VIPRE-01

As discussed above, the final evaluation of fuel enthalpy, fuel temperature, and pin radial peaking for comparison to the DNBR and fuel enthalpy limits is performed in the VIPRE-01 code. A VIPRE-01 model with **[[** **]]** is used for fuel temperature and enthalpy calculations. This **[[** **]]** model is consistent with the model employed in DPC-NE-3001, Revision 1, which was approved by the NRC staff. This model is used to simulate the hot pin response, which is selected, as discussed above, based on **[[** **]]**.

The VIPRE-01 fuel pin heat conduction model is used to calculate the fuel rod temperature and enthalpy. **[[**

**]]**. Both modelling options are consistent with the NRC-approved DPC-NE-3001 and are thus acceptable to the NRC staff.

However, **[[**

**]]**.

The subcooled and bulk voiding models discussed in Section 3.3.2 of this SE will be used in the VIPRE-01 analysis. The **[[** **]]** model is used to calculate the two-phase friction multiplier. **[[**

**]]**. The following correlations are used to describe the boiling curve:

- **[[**

**]]**

These correlations have been previously found to be acceptable to the NRC staff in various other applications and are therefore acceptable in DPC-NE-3009. The CHF correlation used in the DNB evaluation, as discussed in Section 3.3.1 of this SE, is used to define the peak of the boiling curve.

As previously discussed, power distributions from SIMULATE-3K are applied in VIPRE-01 for the DNB, CFM, and fuel enthalpy calculations. For HZP cases, **[[**

**]]**. This is consistent with the previously approved method described in DPC-NE-3001 and is thus acceptable. For HFP cases, **[[**

**]]**.

Engineering hot channel factors are applied to the power distribution. The hot subchannel flow area is reduced by 2.5 percent from the nominal subchannel flow area, and the hot assembly inlet flow is reduced by 5 percent from the nominal assembly flow. These are standard approaches to make the analysis more conservative and are acceptable to the NRC staff.

Direct coolant heating is modeled based on information provided by the fuel vendor. This phenomenon is appropriate to include, and the fuel vendor is the best source of information on the subject. As such, this approach is acceptable to the NRC staff.

As discussed in the response to RAI 48 (Reference 5), the peak pin is analyzed to ensure that the 230 cal/g and CFM criteria are met because no pins are allowed to exceed these criteria. For the HZP case, a fuel pin census is conducted to determine the number of pins that exceed the high temperature cladding failure criterion (150 or 170 cal/g, depending on rod internal pressure). This census is carried out by **[[**

**]]**. These pins are then assumed to fail. The burnup corresponding to a rod internal pressure exceeding the system pressure may be calculated using an NRC fuel performance code, and separate censuses will be performed for pins that are above and below the burnup (corresponding to the two high temperature cladding failure criteria).

If a census evaluation of the prompt fuel enthalpy rise is needed due to changes in the fuel design, fuel management strategy, or plant operation, the licensee stated in the response to RAI 48 that **[[**

**]]**.

The licensee provided a demonstration benchmark in Tables RAI-48-1 and RAI-48-2. The benchmark shows **[[**

**]]**.

Because of the **[[**
**]]**, the NRC staff determined that **[[**

**]]**.

The licensee proposed **[[**

**]]**. Accordingly, based on the foregoing, the NRC staff considers RAI 48 resolved.

3.5.21.2.3 Departure from Nucleate Boiling Evaluation using VIPRE-01 and SIMULATE-3K

<u>Maximum Allowable Radial Peaking Approach</u>

The licensee's overall approach is to determine the radial peaking factors that would result in DNB using VIPRE-01 with the **[[** **]]** model discussed in DPC-NE-2005. With a given **[[**

**]]**. The licensee defines these limits as MARP limits.

For CREs that trip on flux-related trips, **[[**

**]]**.

Once MARP limits have been determined, **[[**

**]]**.

Alternatively, cycle-specific calculations performed with VIPRE-01 may use the core average power, assembly peaking, and pin-power distributions from SIMULATE-3K to directly evaluate the DNBR. In this case, the power distribution from SIMULATE-3K is adjusted for appropriate uncertainties prior to input to VIPRE-01.

DNB Evaluation Initial and Boundary Conditions

The DNBR cases are evaluated at BOC and EOC at HFP to determine the limiting case. HZP conditions are not limiting with respect to DNB and are not analyzed.

**[[**

**]]**

In the MARP analysis, **[[**

**]]**. In the cycle-specific evaluation, as discussed above, the pin power distribution is taken directly from SIMULATE-3K.

VIPRE-01 Models

The fuel pin heat conduction model is used in VIPRE-01. **[[**

**]]**. The NRC staff determined, based on the licensee's discussion and the results from the Robinson demonstration analysis, that the use of the **[[** **]]** is acceptable.
**[[** **]]**

**[[**

**]]**. These models were found to be acceptable in the NRC staff's generic review of VIPRE-01 and are therefore acceptable to the NRC staff for use in DPC-NE-3009.

3.5.21.2.4 System Thermal-Hydraulic Calculations using RETRAN-3D

If a flux-based trip does not occur, the length of the transient increases significantly, allowing primary system thermal-hydraulic variables (especially primary system temperature) to start to change before the reactor trip. Therefore, when a flux-based trip is not predicted in SIMULATE-3K, a system analysis is performed in RETRAN-3D to determine the overall system thermal-hydraulic response. Peak primary system pressure may also be evaluated in RETRAN-3D.

In the RETRAN-3D analysis, the failure of the control rod drive mechanism housing that causes the CRE is assumed to create a hole in the reactor vessel head the size of the control rod drive shaft. In response to RAI 49 (Reference 4), the licensee provided some details on how this hole

is modeled. **[[**

**]]**. The NRC staff therefore found the modeling of the hole in the RCS caused by the ejection of the control rod to be acceptable.

Core power is taken from the SIMULATE-3K solution. For the core cooling system thermal-hydraulic calculation, initial conditions are specified at SCD conditions. The availability of primary and secondary systems and components (e.g., pressurizer heaters, sprays, etc.) is evaluated to identify conservative treatment. For the primary system pressure analysis, initial and boundary conditions are assumed that maximize both the primary and secondary system pressures (which also serves to minimize the heat transfer from the primary system to the secondary system, exacerbating the primary system pressure response).

3.5.21.3 Harris Control Rod Ejection Demonstration Analysis (DPC-NE-3009)

The NRC staff reviewed the specific assumptions for the Harris CRE accident demonstration analysis presented in Section 6.7 of the methodology report. The assumed initial and boundary conditions were found to conform to the method, as discussed in Section 3.5.21 of this SE. **[[**

**]]**. In comparison to the AOR presented in the Harris UFSAR, some parameters from the demonstration analysis (ejected rod worth, effective delayed neutron fraction) are more conservative, while others (DTC and MTC) are not. This is not because the parameters in the demonstration analysis are insufficiently conservative, however; it is because the AOR parameters are overly and unrealistically conservative - an EOC MTC of 0 pcm/°F, for example, is not a reasonable assumption.

The licensee also stated that, for the BOC HZP case, “the high boron concentration necessary to achieve a positive MTC was found to suppress the power excursion.” The case was therefore analyzed with an MTC of 0 pcm/°F, rather than the TS upper limit of 5 pcm/°F. In response to RAI 50 (Reference 5), the licensee provided sensitivity studies to demonstrate that the case with an MTC of 0 pcm/°F was conservative. Two studies were performed; **[[**

**]]**. Based on the results of the licensee’s sensitivity studies, the NRC staff determined that the licensee’s approach to modeling the MTC conditions for BOC rod ejection cases is acceptable. Accordingly, the NRC staff considers RAI 50 resolved.

In the Harris demonstration analysis, the licensee modeled combinations of HZP and HFP initial conditions and BOC and EOC times in life. All four cases were found to quickly trip on flux trips, negating the need for a systems analysis, and the DNBR and the CFM were not evaluated in the Harris demonstration analysis. For these reasons, the power curves are essentially all that is of interest to the NRC staff. These curves behave as expected. The only distinctions between the BOC and EOC HFP cases are (1) the EOC case has slightly higher peak power because of the slight increase in ejected rod worth and slight decrease in effective delayed neutron fraction relative to the BOC case, and (2) the power in the EOC case begins to decrease immediately after reaching the peak due to negative reactivity feedback from the more negative DTC and MTC. In the BOC HZP case, the ejected rod worth and delayed neutron fraction combine to add exactly one dollar of reactivity to make the reactor prompt critical, causing a sharp increase in core power to approximately 25 percent before negative Doppler temperature effects halt the power rise. In the EOC HZP case, the insertion of 1.44$ of reactivity causes a massive power spike to 484 percent power. These are consistent with the results that would be expected for rod ejections from these times in core life.

3.5.21.4 Robinson Control Rod Ejection Demonstration Analysis (DPC-NE-3009)

NRC staff reviewed the specific assumptions for the Robinson CRE accident demonstration analyses presented in Section 6.8 of DPC-NE-3009. The assumed initial and boundary conditions were found to conform to the method, as discussed in Section 3.5.21 of this SE. As in the Harris rod ejection demonstration analyses, **[[**

**]]**.

The limiting case from the AOR appears to be the BOC/HFP case. Compared to the demonstration analysis, the MTC and DTC and the ejected rod worth are conservative, while the delayed neutron fraction is not. However, the analyses have completely different results, as the AOR trips on the high flux trip that is disabled by the single worst failure in the demonstration analysis. Because the demonstration analysis fails to trip on the high flux trip, the transient is calculated out for an extended period of time after the rod ejection, and includes a system response analysis using RETRAN-3D that is based on the power response from SIMULATE-3K. The RETRAN-3D analysis inputs are consistent with the methodology, which uses SCD conditions.

In general, the transients proceed as expected. Because the Robinson ejected rod worths are higher and the DTCs less negative than the Harris demonstration analyses, the Robinson demonstration analyses result in higher power excursions before the transient is turned around (for three of the four cases). Otherwise, analysis results are very similar. In the BOC/HFP transient, which does not trip on a flux trip and is simulated with RETRAN-3D, the reactor power level jumps immediately to a 118 percent power level but decreases and settles to about 110 percent power.

As shown in the licensee's response to RAI 51 (Reference 5), the primary system temperature slowly increases and pressurizer pressure slowly decreases until the reactor trips on OTΔT at around 34 seconds after the transient initiation. The MDNBR occurs just before the reactor trip, as would be expected for a transient of this type.

The Robinson demonstration analysis also contained a sensitivity study on the effect of gap conductance modeling. This study was discussed in some detail in Section 3.5.21.2.3 of this

SE. The NRC staff concluded that the use of the **[[**
**]]** was acceptable based on the results of the study. Accordingly, the NRC staff considers RAI 51 resolved.

3.5.22 Inadvertent Operation of the Emergency Core Cooling System

The inadvertent operation of the emergency core cooling system (IOECCS) event is initiated by a spurious actuation of the ECCS, resulting in delivery of highly borated water to the RCS. The negative reactivity insertion may decrease core power and primary pressure, which, when combined with the RCS inventory increase from the ECCS injection, may result in overfilling the pressurizer and relieving liquid through pressurizer PORVs and safety relief valves (SRVs). At Robinson, the high pressure SI pumps have a shutoff head that is much lower than the trip setpoint pressure, which is lower than the RCS pressure (1500 psia, 1800 psia, and 2250 psia, respectively); as such, the IOECCS is not analyzed at Robinson, consistent with the current licensing basis.

The IOECCS at Harris is classified as an ANS Condition II event, which is consistent with the plant UFSAR. Acceptance criteria are primarily to ensure that the event does not generate a more serious condition without the occurrence of other independent faults, as well as to ensure that the DNBR remains above the 95/95 limit. The event is mitigated by a low pressurizer pressure trip.

Peak primary pressure is bounded by the turbine trip. Because the relief capacity of the pressurizer PORVs and safety valves exceeds the capacity of the ECCS, the peak primary system pressure is limited to the pressurizer safety valve lift setpoint. Because the turbine trip is not subject to these same limitations, the post-turbine trip flow through the safeties exceeds the capacity of the valves and the primary pressure continues to rise until the reactor trips. The NRC staff finds this justification to be acceptable and determined that no primary pressure analysis is necessary for the IOECCS.

Peak secondary pressure is also bounded by the turbine trip. The IOECCS event does not result in a significant power excursion, and a heat sink remains available after the reactor is tripped. The turbine trip is more limiting with respect to secondary pressure because the SGs are isolated from the heat sink before the reactor is tripped. The NRC staff finds this justification to be acceptable and determined that no secondary pressure analysis is necessary for the IOECCS.

The CVCS malfunction that increases RCS inventory is also considered under the umbrella of events considered under the IOECCS. This event does actually apply to Robinson; however, the arguments that peak primary and secondary system pressures are bounded by other events are the same as those above, except that the loss of load event is limiting rather than the turbine trip (for reasons discussed in Section 3.5.5 and 3.5.6 of this SE). The NRC staff finds this justification to be acceptable, and determined that peak primary and secondary system pressure analyses are not necessary for the CVCS malfunction analysis at Robinson.

The MDNBR is not actually analyzed; Duke proposed that it would be bounded by the inadvertent operation of a pressurizer relief or safety valve transient, which results in a similar but more severe pressure decrease than IOECCS. If this argument were to be invalidated, the MDNBR would be performed using the SCD methodology. A similar disposition to the DNB evaluation and discussion of how the analysis would be performed if necessary was found

acceptable in the NRC staff's review of the DPC-NE-3002, Revision 0 (Reference 33)[9] report for Catawba and McGuire. The NRC staff thus finds this to be acceptable. Consistent with the Harris UFSAR, the primary system pressure results inherently satisfy the acceptance criterion of keeping the system pressure below 110 percent of design pressure because the PORV and SRV relief capacity far exceeds the ECCS capacity to fill the pressurizer. This is also acceptable.

A pressurizer overfill analysis is therefore the primary analysis performed for the IOECCS event. However, because some liquid relief is anticipated, the acceptance criterion for the pressurizer overfill analysis is not that there will be no liquid relief through the PORVs and SRVs. Rather, the analysis seeks to ensure that the liquid relief is of sufficiently high temperature to ensure that the PORV and SRV operability is not degraded. The analysis evaluates the minimum fluid conditions to ensure that the valves remain operable while discharging subcooled water.

HFP conditions are assumed and the analysis is conducted with EOC core physics. Initial and boundary conditions are selected to maximize the primary system depressurization rate and minimize the fluid temperature at the entrance to the pressurizer PORVs and SRVs. As discussed in the licensee's response to RAI 53 (Reference 4), boron injection is modeled using the RETRAN-3D general transport model (as previously described in Section 5.1.4.1 of DPC-NE-3009 and Section 3.2.2.3.5 of this SE). Maximum boron concentration and boron worth are assumed, as are minimum boron transport delays (i.e., flushing volumes are minimized). These assumptions will serve to maximize the reduction in RCS pressure and coolant temperatures, and are therefore acceptable to the NRC staff.

No single failure has been identified by the licensee that would adversely affect the consequences of the event. No single failure is discussed in the Harris UFSAR, so this is consistent with the plant licensing basis and is thus acceptable.

The NRC staff determined that the licensee's approach to analyzing the IOECCS event was appropriate, based on the information contained in the DPC-NE-3009 report and associated RAI responses. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. Accordingly, the NRC staff considers RAI 53 resolved.

3.5.23 Inadvertent Opening of a Pressurizer Relief or Safety Valve

The inadvertent opening of a pressurizer relief or safety valve event is initiated by the failure of a pressurizer PORV or safety valve in the fully open position. This causes an RCS depressurization that leads to a loss of DNB margin, and is usually mitigated by low pressurizer pressure or OTΔT trips. The DPC-NE-3009 methodology classifies the transient as an ANS Condition II event for Harris and an ANS Condition IV event for Robinson, consistent with the plant UFSARs. The appropriate acceptance criteria are applied, which are that the DNBR must remain above the 95/95 limit (Condition II) or that the release of radioactive material must not result in releases that exceed regulatory limits (Condition IV). Since this is a depressurization event, it stands to reason that peak primary and secondary system pressures are bounded by other events and are not analyzed.

[9] DPC-NE-3002, Revision 4b in the reference documents the NRC approval of DPC-NE-3002, Revision 0.

The MDNBR is determined using the SCD methodology. The SCD-based analysis is performed at BOC/HFP conditions, which is intended to maximize the core power prior to the reactor trip. At Robinson, the potential for fuel damage due to core uncovery following the trip is addressed by the LOCA analysis. At Harris, where the event is classified as an ANS Condition II transient, core uncovery is not allowed by the acceptance criterion that the event may not generate a more serious plant condition without the occurrence of other independent faults.

As discussed in the licensee's response to RAI 55 (Reference 4), the SCD method is only applicable to the short term phase of the analysis (up until around the time of the reactor trip). The long term phase of the analysis, where there is the potential for core uncovery, is addressed in the plant licensing bases either by the LOCA analysis (at Robinson) or using small break LOCA (SBLOCA) analysis methods to demonstrate that the long-term phase of the event does not result in fuel failure (at Harris). In this second case, the SBLOCA analysis was performed with conservative initial conditions and no core uncovery or DNB was allowed to occur. If a long-term analysis is needed, it will be performed using the licensing basis SBLOCA methods and would assume conservative initial conditions, consistent with the Harris licensing basis. The NRC staff finds the overall analysis approach for the short-term phase of the transient to be acceptable, but notes that the long-term phase of the transient is considered to be outside of the scope of this review. Any future use of the licensing basis SBLOCA methods for reviewing the inadvertent opening of a pressurizer relief or safety valve would need to be reviewed by the licensee to determine whether further NRC review is required.

The NRC staff reviewed the event initial and boundary conditions and found them to be consistent with the SCD methodology proposed for analyzing the MDNBR. A variety of system parameters are evaluated to determine the limiting condition, including pressurizer level, the availability of pressurizer spray and heaters, the availability of the main steam PORVs, and the availability of the steam dumps. As discussed in the licensee's response to RAI 54 (Reference 4), the flow capacity of the safety valve bounds that of the PORV and it is therefore modeled in the analysis. Maximum flow is considered by adjusting the effective area of the valve to achieve maximum flow capacity at rated conditions. The valve failure itself is modeled as a rapid opening of the valve to the fully open position.

Critical flow through the valve is modeled using a combination of the extended Henry-Fauske model for subcooled critical flow and Moody for saturated critical flow – this is consistent with the requirements of 10 CFR Part 50 Appendix K for LOCAs, and is therefore acceptable to the NRC staff.

The NRC staff determined that the licensee's approach to analyzing the inadvertent operation of a PORV or safety valve event was appropriate, based on the information contained in the DPC-NE-3009 report and associated RAI responses. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, and single failure assumptions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. Accordingly, the NRC staff considers RAIs 54 and 55 resolved.

3.5.24 Steam Generator Tube Rupture

A SG tube rupture (SGTR) event is initiated by the double-ended guillotine rupture of a single SG u-tube. As described in the Harris UFSAR, the event essentially proceeds as follows. The SGTR causes the pressurizer pressure to slowly decrease. The flow through the ruptured SG tube causes a steam flow/feedwater flow mismatch in the affected SG. Continued loss of

coolant inventory eventually leads to a reactor trip, either on low pressurizer pressure or OTΔT. The reactor trip also trips the turbine, and with offsite power available, opens the steam dump valves to the condenser; if offsite power is not available, the steam dumps would not open and SGs would eventually pressurize to the point of lifting SG PORVs or safety valves. SI, initiated by low pressurizer pressure, follows soon after the reactor trip. The SI actuation terminates normal feedwater supply to the SGs and initiates AFW. Borated water injection from the SI helps absorb decay heat, limiting the necessary steam relief. The RCS pressure decreases until the SI flow equals the flow from the primary to secondary systems through the ruptured tube. Operator actions are necessary to recover from the SGTR, and include identifying the ruptured SG, isolating it from the intact SGs and the feedwater system, cooling and depressurizing the RCS using the intact SGs, and terminating SI.

The event is defined as an ANS Condition IV event for Harris and Robinson, consistent with the plant licensing bases. Peak primary and secondary pressure are bounded by the loss of load or turbine trip, depending on the plant. The primary system response is a depressurization transient and is inherently non-limiting for peak primary system pressure. The secondary system response is characterized by increasing inventory in the ruptured SG. Though there is some amount of post-trip heat transfer, it is managed by the use of the SG PORVs or safety valves. Thus, the loss of load and turbine trip are more limiting. The NRC staff finds this justification to be acceptable and determined that no peak primary or secondary pressure analysis is necessary.

The Robinson UFSAR states that the thermal-hydraulic response to the SGTR is bounded by the inadvertent operation of a pressurizer PORV or safety valve transient. The Harris UFSAR, on the other hand, provides a lengthy description of the need to perform a SGTR analysis to ensure margin to SG overfill and to ensure that the system thermal-hydraulic response does not violate the assumptions of the dose analysis.

Section 5.6.2 of DPC-NE-3009 provides a method of analysis to ensure that the dose limits are not exceeded, by determining the number of fuel pins that exceed the 95/95 DNBR limit. In this method, the MDNBR is determined using the SCD methodology. The analysis is performed at BOC/HFP conditions, to maximize the fuel pin surface heat flux, and assumes a LOOP on the turbine trip (coincident with the reactor trip). No single active failure was identified to adversely affect the consequences of the event.

The licensee's response to RAI 56 (Reference 4) provided a methodology to analyze the SG margin to overfill (MTO) and the thermal-hydraulic portion of the related offsite dose assessment, for consistency with the Harris UFSAR. In the MTO method, power and SG level are biased high, and RCS coolant temperature, pressurizer pressure, decay heat, and SG tube plugging will all be evaluated with sensitivity studies to determine the limiting direction. Reactor trips are expected on low pressurizer pressure and OTΔT, and the SI signal is expected on low pressurizer pressure; all trip and SI setpoints are adjusted to conservatively account for uncertainties. As in the DPC-NE-3009, Section 5.6.2 analysis, a LOOP is expected following the reactor trip.

The break location is assumed to be at the top of the tube sheet on the cold side of the SG, to maximize the primary to secondary system flow rate. AFW flow is maximized by modeling both motor driven AFW pumps and the turbine driven AFW pump, and temperature is minimized. SI flow is maximized by modeling flow from two HHSI pumps. Several single failures will be considered and evaluated to determine the most limiting condition: failure of an intact SG PORV

to open, failure of the turbine-driven AFW pump speed controller, and failure of an AFW flow control valve to close on demand. The operator actions currently credited in the Harris UFSAR Section 15.6.3 analysis are credited in the new methodology.

The NRC staff determined that the licensee's approach to analyzing the SGTR event was appropriate, based on the information contained in the DPC-NE-3009 report and associated RAI responses. The acceptance criteria, initial and boundary conditions, system availability, plant modeling, single failure assumptions, and operator actions are consistent with the Harris and Robinson licensing bases or are otherwise acceptable. Accordingly, the NRC staff considers RAI 56 resolved.

3.6 Conclusion Regarding DPC-NE-3008 and DPC-NE-3009

The NRC staff reviewed the DPC-NE-3008 and DPC-NE-3009 reports and found the that RETRAN-3D models, VIPRE-01 models, and safety analysis methodologies presented in the reports are acceptable for performing UFSAR Chapter 15 non-LOCA safety analyses to demonstrate compliance with the general and plant-specific design criteria discussed in Section 2.0 of this SE at Harris and Robinson. DPC-NE-3008 and DPC-NE-3009 are therefore approved on a plant-specific basis for Harris and Robinson and are acceptable for incorporation into the Harris and Robinson UFSARs and TSs.

The NRC staff's approval of DPC-NE-3008 and DPC-NE-3009 is contingent on the following limitations and conditions:

1. **[[**

   **]]**. In this case, the licensee should also evaluate the use of this correlation to determine if NRC approval is required.

2. When using RETRAN-3D's temperature transport delay and general transport models, the Courant limit should be observed and the results must be carefully evaluated to ensure there are no anomalous results or numerical diffusion. Documentation of this evaluation shall be included in the calculation note along with the analysis.

3. The EPRI-1 correlation may not be used if a CHF correlation specifically developed for the fuel in use at the plant is applicable to the conditions being modeled (e.g., in the case of DPC-NE-3009 for Harris and Robinson at the present time, the HTP correlation must be used within its range of applicability instead of EPRI-1).

4. Credit for plant mitigating systems beyond those explicitly discussed in the methodology must be evaluated by the licensee to determine whether prior NRC approval is necessary.

5. The NRC's acceptance of a 10 minute AFW isolation time for the feedwater line break at Robinson is based on the similarity of the early portion of the transient response to the main SLB. If system changes are made that result in a response that differs significantly from the main SLB, the NRC staff would no longer consider the 10 minute time to isolate AFW based on the SLB analysis to be appropriate.

6. Margins to the CRE PCMI failure acceptance criteria should be re-examined **[[**

   **]]**. If the margin is found to be substantially degraded from its current state, the licensee shall explicitly consider fuel cladding corrosion in the evaluation against the CRE PCMI failure acceptance criteria.

7. The methodology presented an approach for determining conservative factors for use with the **[[**

   **]]**.

3.7 Technical Specification Changes

In addition to the review of DPC-NE-3008 and DPC-NE-3009, the licensee requested incorporation of these reports into the Harris and Robinson TSs.

Robinson TS 5.6.5.b is amended to include the following new COLR references:

32. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated [Month xx, xxxx].
33. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated [Month xx, xxxx].

Harris TS 6.9.1.6.2 is amended to include the following new COLR references:

t. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated [Month xx, xxxx].

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

u. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated [Month xx, xxxx].

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

The proposed TS changes are consistent with Generic Letter 88-16 on the use of plant-specific methodologies in generating core operating limits, in that they appropriately identify the NRC staff's SE for each COLR reference. The NRC staff notes that the numbering of the COLR references differs from that initially proposed reflecting the approval of other COLR references in the interim. Similarly, a "-A" was added to each report number to reflect that the reports are approved and to be consistent with other COLR references. As such, and because DPC-NE-3008 and DPC-NE-3009 were determined to be acceptable on a plant-specific basis for Harris and Robinson as discussed above, the NRC staff determined that the proposed TS changes are acceptable in accordance with 10 CFR 50.36.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official and the South Carolina State official were notified of the proposed issuance of the amendment on March 1, 2018. The State officials had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (82 FR 20496). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 6.0 CONCLUSION

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

## 7.0 REFERENCES

1 Repko, R.T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Application to Revise Technical Specifications to Adopt Methodology Report DPC-NE-3008-P Revision 0, "Thermal Hydraulic Models for Transient Analysis," November 19, 2015 (Agencywide Documents Access and Management (ADAMS) Accession No. ML15323A351 (public), ML15323A352 (non-public)).

2 Elnitsky, J., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Supplemental Information for License Amendment Request Regarding Methodology Report DPC-NE-3008-P," October 3, 2016 (ADAMS Accession No. ML16278A080 (public), ML16278A082 (non-public)).

3 Elnitsky, J., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3008, Revision 0," November 10, 2016 (ADAMS Accession Number ML16315A286).

4 Donahue, J., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3009, Revision 0," October 9, 2017 (ADAMS Accession No. ML17282A023 (public), ML17282A024 (non-public)).

5 Donahue, J., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3009, Revision 0 (Part 2)," October 30, 2017 (ADAMS Accession No. ML17303B205 (public), ML17303B209 (non-public)).

6 Donahue, J., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Response to Second Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-3009, Revision 0," December 19, 2017 (ADAMS Accession No. ML17354A002).

7 Duke Energy Carolinas, LLC, "Thermal-Hydraulic Transient Analysis Methodology," DPC-NE-3000, Revision 5a, October 2012 (ADAMS Accession No. ML16032A004 (public), ML16032A005 (non-public)).

8 Duke Power Company, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," DPC-NE-3001, Revision 0, December 2000 (ADAMS Accession No. ML010080327).

9 Barillas, M., U.S. Nuclear Regulatory Commission, email to R.T. Repko, Duke Energy Progress, LLC, "Acceptance of Requested Licensing Action Regarding Duke Energy Progress, Inc. Application to Revise Technical Specifications to Adopt Methodology Report DPC-NE-3008-P, Revision 0, 'Thermal-Hydraulic Models for Transient Analysis' (CACs MF7112/MF7113)," January 8, 2016 (ADAMS Accession No. ML16011A064).

10 Galvin, D.J., U.S. Nuclear Regulatory Commission, letter to J. Elnitsky, Duke Energy Progress, LLC., "Requests for Additional Information Regarding Application to Adopt DPC-NE-3008-P, Revision 0, "Thermal-Hydraulic Models for Transient Analysis" (CAC Nos. MF7112 and MF7113)," October 24, 2016 (ADAMS Accession No. ML16216A061 (public), ML16216A071 (non-public)).

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