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May 8, 2015

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

Serial No. 15-159
NLOS/MLC R0
Docket No. 50-423
License No. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
LICENSE AMENDMENT REQUEST TO ADOPT DOMINION CORE DESIGN AND SAFETY ANALYSIS METHODS AND TO ADDRESS THE ISSUES IDENTIFIED IN WESTINGHOUSE DOCUMENTS NSAL-09-5, REV. 1, NSAL-15-1, AND 06-IC-03.

In accordance with the provisions of 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) is submitting a request for an amendment to revise the Technical Specifications (TS) for Millstone Power Station Unit 3 (MPS3). The proposed changes would revise the TS to enable use of Dominion nuclear safety and reload core design methods for MPS3 and address the issues identified in three Westinghouse communication documents. This effort is analogous to that applied previously to Kewaunee Power Station and approved in Reference 1.

The scope of this license amendment request was presented to the Nuclear Regulatory Commission (NRC) staff during a teleconference on February 3, 2015. In response to feedback provided by the NRC staff, the scope of this submittal was expanded to address the issues identified in three Westinghouse communication documents.

Specifically, the proposed changes accomplish the following objectives:

- Enable use of Dominion nuclear safety and reload core design methods for MPS3
- Approve use of applicable departure from nucleate boiling ratio (DNBR) design limits for VIPRE-D at MPS3
- Update the approved reference methodologies cited in TS 6.9.1.6.b
- Eliminate the base load mode of operation, which is not a feature of the Dominion Relaxed Power Distribution Control (RPDC) power distribution control methodology
- Address the issues identified in Westinghouse NSAL-09-5, Rev. 1 for MPS3
- Address the issue identified in Westinghouse NSAL-15-1 for MPS3
- Address the issue identified in Westinghouse Communication 06-IC-03 for MPS3

The justification for adopting Dominion core design and safety analysis methods to MPS3 involved a systematic review of the several methods to be applied and the necessary analyses and evaluations which establish applicability to MPS3. The Dominion NRC-approved methods to be applied to MPS3 are documented in the following topical reports, with the general topic noted in brackets. Also listed below are those topical reports that DNC is proposing to add to the reference list of applicable methodologies for establishing core operating limits located in TS 6.9.1.6.b.

Methods to be added to the MPS3 COLR are:

- VEP-FRD-42-A [Reload Nuclear Design Methodology] (Reference 2)
- VEP-NE-1-A [Relaxed Power Distribution Control] (Reference 3)

A001
NRR

- DOM-NAF-2-P-A [Core Thermal/Hydraulics Using VIPRE-D] (Reference 4)
- VEP-NE-2-A [Statistical DNBR Evaluation Methodology] (Reference 5)

Supporting Methods

- VEP-FRD-41-P-A [RETRAN NSSS Non-LOCA Analysis] (Reference 6)
- DOM-NAF-1-P-A [Core Management System (CMS) Reactor Physics] (Reference 7)

Attachment 1 provides a description, technical analysis, regulatory evaluation, and environmental considerations for the proposed changes.

Attachment 2 contains marked-up pages to reflect the proposed changes to the TS.

Attachment 3 contains marked-up pages to reflect the changes to the TS Bases. These pages are provided for information only and will be incorporated in accordance with the TS Bases Control Program upon approval of this amendment request.

Attachment 4 presents an evaluation for each of the Dominion methods, as it applies to MPS3. The evaluation includes a description of each method, including its purpose, key features, and dependencies. Also addressed is the identification of any conditions and limitations associated with each method. Such items include limitations in NRC Safety Evaluation Reports, physical limitations (plant systems, features, and conditions), and limitations specified in Dominion topical reports (specific modeling approaches or inherent assumptions). Each method is assessed with respect to these conditions and limitations, and the assessments range from written evaluations to validation and/or benchmark analyses and detailed comparison of results.

Attachment 5 presents the benchmark analyses performed for application of VEP-FRD-41-P-A to MPS3. The benchmark analysis contains analyses of several current Final Safety Analysis Report (FSAR) events, representing a wide range of event types. The benchmark analyses using Dominion's models are compared against the current FSAR analyses with a focus on modeling the key phenomena. The analyses demonstrate acceptable agreement with the Westinghouse FSAR analysis performed for the MPS3 Stretch Power Uprate.

Attachment 6 presents the analysis for application of Dominion's Statistical DNBR analysis to MPS3. The analysis involves development of applicable design limits with the Dominion VIPRE-D code and Westinghouse DNB correlations. These Statistical Design Limits (SDLs) constitute Design Basis Limits for a Fission Product Barrier (DBLFPB), and are required to be submitted for NRC review and approval per 10 CFR 50.59(c)(2)(vii).

The proposed amendment does not involve a Significant Hazards Consideration under the standards set forth in 10 CFR 50.92. The Millstone Facility Safety Review Committee has reviewed and concurred with the determinations herein.

DNC requests approval of the proposed amendment by October 31, 2016 to support long lead planning activities associated with operation of MPS3 Cycle 19.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

Should you have any questions in regard to this submittal, please contact Wanda Craft at (804) 273-4687.

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2016

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 8th day of May, 2015.

My Commission Expires: 12/31/16.

Notary Public

References:

1. Letter from Patrick D. Milano (USNRC) to David A. Christian (Dominion), "Kewaunee Power Station – Issuance of Amendment Re: Nuclear Core Design and Safety Analysis Methods (TAC NO. MD6824)," March 28, 2008 (ADAMS Accession No. ML080630529).

2. Topical Report, VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.

3. Topical Report, VEP-NE-1, Rev. 0.1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," August 2003.

4. Fleet Report, DOM-NAF-2-P-A, Rev. 0.3, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," September 2014.

5. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

6. Topical Report, VEP-FRD-41-P-A, Rev. 0.2, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," March 2015.

7. Topical Report, DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," June 2003.

Attachments:

1. Evaluation of Technical Specifications Changes
2. Marked-up Technical Specifications Pages
3. Marked-up Technical Specifications Bases Pages for Information Only
4. Application of Dominion Nuclear Core Design and Safety Analysis Methods
5. RETRAN Benchmarking Information
6. Development of Statistical Design Limits

Commitments made in this letter: None

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ATTACHMENT 1

EVALUATION OF TECHNICAL SPECIFICATIONS CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Evaluation of Technical Specifications Changes

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1.0 DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) is submitting a request to amend the Technical Specifications (TS) for Millstone Power Station Unit 3 (MPS3). DNC is proposing to change, add, or delete the following TS and Surveillance Requirements (SR):

TS 1.43, 1.44	ALLOWED POWER LEVEL
TS 3/4.2.1.1	Axial Flux Difference
TS 3/4.2.2.1	Heat Flux Hot Channel Factor – $F_Q(Z)$
TS 3.2.3.1.b(5)	RCS Flow Rate (measurement uncertainty)
SR 4.2.3.1.3.a	RCS Flow Rate (precision heat balance)
TS 6.9.1.6.a	Core Operating Limits Report
TS 6.9.1.6.b	

The Bases for TS 3/4.2.1.1 and TS 3/4.2.2.1 are being modified to address the proposed changes and are provided for information only. Changes to the TS Bases will be incorporated in accordance with the TS Bases Control Program (TS 6.18) upon approval of this amendment request.

Pursuant to 10 CFR 50.59(c)(2)(vii), DNC also requests review and approval of the Statistical Design Limits (SDLs) provided within, as they constitute Design Basis Limits for a Fission Product Barrier (DBLFPB).

This license amendment request (LAR) contains proposed changes to the above TS to address the following objectives:

- Enable use of Dominion nuclear safety and reload core design methods for MPS3
- Approve use of applicable departure from nucleate boiling ratio (DNBR) design limits for VIPRE-D at MPS3
- Update the approved reference methodologies cited in TS 6.9.1.6.b
- Eliminate the base load mode of operation, which is not a feature of the Dominion Relaxed Power Distribution Control (RPDC) power distribution control methodology
- Address the issues identified in Westinghouse NSAL-09-5, Rev. 1 for MPS3
- Address the issue identified in Westinghouse NSAL-15-1 for MPS3
- Address the issue identified in Westinghouse Communication 06-IC-03 for MPS3

2.0 PROPOSED TECHNICAL SPECIFICATIONS CHANGES

The proposed TS changes are detailed below. To aid review, deleted text is struck through and added text is italicized and bolded. For more extensive changes, reference is made to the markups in Attachments 2 and 3.

2.1 TS 1.43, 1.44 - Definitions

These definitions involving minimum and maximum allowable power level associated with base load operation are being deleted.

Definitions 1.43 and 1.44 – ALLOWED POWER LEVEL

- Delete definition 1.43 and replace with the word "Deleted."

  1.43 ***Deleted*** ~~APLND is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.~~

- Delete definition 1.44 and replace with the word "Deleted."

  1.44 ***Deleted*** ~~APL^{BL} is the maximum allowable power level when transitioning into base load operation.~~

2.2 TS 3.2.1.1, SR 4.2.1.1 – Axial Flux Difference and TS 3.2.2.1, SR 4.2.2.1 – Heat Flux Hot Channel Factor – $F_Q(Z)$

TS 3.2.1.1 and 3.2.2.1 and associated SR 4.2.1.1 and 4.2.2.1 currently reflect use of the Westinghouse Relaxed Axial Offset Control (RAOC) power distribution control methodology. Also included in these specifications are the provisions for base load operation. In addition, three Westinghouse notices identified issues concerning these TS sections (NSAL-09-5, Rev. 1, Reference 1; 06-IC-03, Reference 2; and NSAL-15-1, Reference 8).

The proposed changes detailed below remove the base load operation mode, revise the specification language, and relocate certain equations to make it independent of specific power distribution control methodology. In addition, the proposed changes revise appropriate TS action statements and bases to address the issues in References 1, 2 and 8.

The proposed changes follow:

TS 3.2.1.1.a, 3.2.1.1.b – Axial Flux Difference

- TS 3.2.1.1.a will be revised as follows:

  a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR) ~~for Relaxed Axial Offset Control (RAOC) operation, or~~.

- Delete TS 3.2.1.1.b which is associated with base load operation and replace with the word "Deleted.".

  b. ***Deleted*** ~~Within the target band about the target flux difference during base load operation, specified in the COLR~~.

TS 3.2.1.1 – ACTION

- TS 3.2.1.1 ACTION a will be revised as follows:

  a. ~~For RAOC operation~~ ***W***~~w~~ith the indicated AFD outside of the applicable limits specified in the COLR,

- Delete TS 3.2.1.1 ACTION b.1 and ACTION b.2 which are associated with base load operation. Replace TS 3.2.1.1 ACTION b with the word "Deleted.".

  b. ***Deleted*** ~~For base load operation above APL~~$^{\cancel{ND}}$ ~~with the indicted AFD outside of the applicable target band about the target flux differences:~~

     ~~1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or~~

     ~~2. Reduce THERMAL POWER to less than APL~~$^{\cancel{ND}}$ ~~of RATED THERMAL POWER and discontinue base load operation within 30 minutes.~~

SR 4.2.1.1.3, 4.2.1.1.4 – Surveillance Requirements

- Delete SR 4.2.1.1.3 and SR 4.2.1.1.4, which are associated with base load operation and replace with the word "Deleted.".

  4.2.1.1.3 ***Deleted*** ~~When in base load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement at the frequency specified in the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.~~

  4.2.1.1.4 ***Deleted*** ~~When in base load operation, the target flux difference shall be updated at the frequency specified in the Surveillance Frequency Control Program by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.1.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.~~

TS 3.2.2.1 – Heat Flux Hot Channel Factor – FQ(Z)

- Delete the Limiting Condition for Operation (LCO) requirements of TS 3.2.2.1 (see markup in Attachment 2) and replace with the following:

3.2.2.1 ***$F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.***

TS 3.2.2.1 – ACTION

- TS 3.2.2.1 ACTION a will be revised as follows:

  a. ~~For RAOC operation~~ ***W***with Specification 4.2.2.1.2.b not being satisfied ~~or for base load operation with Specification 4.2.2.1.4.b not being satisfied~~:

- .TS 3.2.2.1 ACTION b will be revised as follows:

  b. ~~For RAOC operation~~ ***W***with Specification 4.2.2.1.2.c not being satisfied, ~~one~~ ***all*** of the following ACTIONS shall be taken:

- Delete TS 3.2.2.1 ACTION b(1) and replace with the following :

  (1) ~~Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by at least 1% AFD for each percent s limits. Within 8 hours, reset the AFD alarm setpoints to modified limits, or~~

  a. Within 15 minutes, control the AFD to within the new reduced AFD limits specified in the COLR that restores $F_Q(Z)$ to within its limits, and

  b. Reduce the THERMAL POWER by the amount specified in the COLR that restores $F_Q(Z)$ to within its limits within 4 hours, and

  c. Reduce the Power Range Neutron Flux – High Trip Setpoints by ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

  d. Reduce the Overpower ΔT Trip Setpoints by ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

  e. Within 8 hours, reset the AFD Alarm Setpoints to the modified limits, and

  f. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by (1)b. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits."

- Delete TS 3.2.2.1 ACTION b(2) and ACTION c which are associated with base load operation (see markups in Attachment 2) and replace both ACTIONS with the word "Deleted.".

SR 4.2.2.1.2 – Surveillance Requirements

- SR 4.2.2.1.2 will be revised as follows:

  4.2.2.1.2 ~~For RAOC operation,~~ $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

- Delete SR 4.2.2.1.2.c (see markup in Attachment 2) and replace with the following:

  c. ***Verify $F_Q^M(Z)$ satisfies the non-equilibrium limits specified in the COLR.***

- Delete SR 4.2.2.1.2.e (see markup in Attachment 2) and replace with the following:

  e. ***Compliance with the non-equilibrium limits shall be conservatively accounted for during intervals between $F_Q^M(Z)$ measurements by performing either of the following:***

- SR 4.2.2.1.2.e(1) will be revised as follows:

  (1) Increase $F_Q^M(Z)$ by an appropriate factor specified in the COLR and verify that this value satisfies ~~the relationship in~~ Specification 4.2.2.1.2.c, or

- SR 4.2.2.1.2.e(2) will be revised as follows:

  (2) ***Verify*** $F_Q^M(Z)$ ~~shall be measured~~ ***satisfies its limits*** at least once per 7 Effective Full Power Days ~~until two successive maps indicate that the maximum value of~~

  $$\cancel{\frac{F_Q^M(Z)}{K(Z)}}$$

  ~~over the core height (Z) is not increasing~~.

- SR 4.2.2.1.2.f will be revised as follows:

  f. The limits specified in Specifications 4.2.2.1.2.c and 4.2.2.1.2.e above are not applicable in the ~~following~~ core plane regions defined ***in the Bases***:

  ~~(1) Lower core region from 0% to 15%, inclusive.~~

  ~~(2) Upper core region from 85% to 100%, inclusive.~~

- Delete SR 4.2.2.1.3 and SR 4.2.2.1.4 (see markups in Attachment 2) which are associated with base load operation and replace with the word "Deleted."

- SR 4.2.2.1.5 will be changed to delete the reference to SR 4.2.2.1.4 (see Attachment 2 markup).

2.3 TS 3.2.3.1.b(5) - RCS Flow Rate (measurement uncertainty)

- TS 3.2.3.1.b(5) will be revised as follows:

  5) The measured value of RCS total flow rate shall be used since uncertainties ~~of 2.4%~~ for flow measurement have been included in Specification 3.2.3.1.a.

2.4 SR 4.2.3.1.3.a - RCS Flow Rate (precision heat balance)

- SR 4.2.3.1.3.a will be revised as follows:

  a. Verifying by precision heat balance that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR within ~~24 hours~~ ***7 days*** after reaching 90% of RATED THERMAL POWER after each fuel loading, and

2.5 TS 6.9.1.6.a – Core Operating Limits Report

- TS 6.9.1.6.a, Items 7, 8, and 9 will be revised as follows:

  7. AXIAL FLUX DIFFERENCE Limits~~, target band, and APLND~~ for Specification 3/4.2.1.1.

  8. Heat Flux Hot Channel Factor ***Limits***~~, K(z), W(z), APLND, and W(z)$_{BL}$~~ for Specification 3/4.2.2.1.

  9. RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3.***1***.

2.6 TS 6.9.1.6.b – Core Operating Limits Report

- TS 6.9.1.6.b will be revised as follows:

  6.9.1.6.b The analytical methods used to determine the core operating limits ***in Specification 6.9.1.6.a*** shall be those previously reviewed and approved by the NRC ~~in~~ ***and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements).***

- To improve readability, the specifications identified in the TS 6.9.1.6.b references will be reformatted using bullets.

- Delete the specifications in Reference 1 and replace with the specifications in Insert C (see Attachment 2).

- Delete TS 6.9.1.6.b References 2, 3, 11, 12, 13, 14, and 15 (see markups in Attachment 2) and replace each with the word "Deleted."

- In TS 6.9.1.6.b References 4, 5, 6, 8, 9, 10, and 19, Specifications 3.2.1 and 3.2.2 will be revised to Specifications 3.2.1.1 and 3.2.2.1.

- TS 6.9.1.6.b Reference 4 will be revised to delete reference to: Relaxed Axial Offset Control and W(z) surveillance requirements for FQ Methodology.

- TS 6.9.1.6.b References 7, 16, and 17 will be revised to add the specification to which the methodologies apply.

- TS 6.9.1.6.b Reference 18 will be revised to delete reference to Specification 2.1.1 and add a descriptor for Specification 2.2.1.

- Four new methodology references will be added to TS 6.9.1.6.b as References 20, 21, 22, and 23 (see Insert F in Attachment 2)

TS Bases Changes

Several changes to the Bases will be required which reflect the relocation of items from specific TS sections noted in the description above. These Bases changes are provided in Attachment 3 for information only and will be incorporated in accordance with the TS Bases Control Program upon approval of this amendment request.

3.0 TECHNICAL ANALYSIS

Attachment 4 provides justification for application of the Dominion safety analysis and reload core design methods to MPS3, in conjunction with the required analyses in Attachment 5 (RETRAN benchmarking) and Attachment 6 (Development of Statistical Design Limits). The proposed TS changes identified in Section 2.0 conform to application of the Dominion methods and are evaluated for technical adequacy in the following sections.

3.1 TS 1.43, 1.44 - Definitions

Definitions 1.43 and 1.44 – ALLOWED POWER LEVEL

Definition 1.43 specifies APL^{ND} as the minimum allowable nuclear design power level for base load operation. The value of APL^{ND} is specified in the COLR. This definition is being deleted, since the base load operation mode is not supported by Dominion methods.

Definition 1.44 specifies APL^{BL} as the maximum allowable power level when transitioning into base load operation. This definition is being deleted since the base load operation mode is not supported by Dominion methods.

3.2 TS 3.2.1.1, SR 4.2.1.1 – Axial Flux Difference and TS 3.2.2.1, SR 4.2.2.1 - Heat Flux Hot Channel Factor - $F_Q(Z)$

The proposed changes involve additions, deletions and revisions to existing content in the TS and TS Bases (provided for information only) that are associated with TS 3.2.1.1, TS 3.2.2.1, SR 4.2.1.1, and SR 4.2.2.1. These changes accomplish three key objectives: 1) accommodate implementation of the Dominion Relaxed Power Distribution Control (RPDC) method, 2) removal of base load operation, and 3) provide resolution of issues documented in Westinghouse notification documents NSAL-09-5, Rev. 1 (Reference 1), 06-IC-03 (Reference 2) and NSAL-15-1 (Reference 8). The proposed changes are structured in a manner that is independent of specific power distribution control methodology (RAOC or RPDC). Relocating the specific equations associated with either the Westinghouse or Dominion power distribution control methodologies to the Bases is consistent with the guidance contained in NUREG-1431, Rev. 4 (Reference 3) for Westinghouse plants with Standard Technical Specifications (STS). Evaluation of the specific proposed changes is provided below.

<u>TS 3.2.1.1</u>

The proposed changes to TS 3.2.1.1.a and TS 3.2.1.1 ACTION 'a' delete a reference to RAOC operation, consistent with the intent to make the TS more general regarding specific axial power distribution control methodology. TS 3.2.1.1.b is deleted since it is associated with the base load mode of operation, which is not supported by the Dominion methods.

<u>SR 4.2.1.1</u>

The proposed changes delete SR 4.2.1.1.3 and SR 4.2.1.1.4, related to the base load mode of operation, which is not supported by the Dominion methods.

<u>TS 3.2.2.1</u>

The proposed changes note that $F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR. The replacement sentence is consistent with that used in the North Anna and Kewaunee ITS and NUREG-1431 (Reference 3). The change to TS 3.2.2.1 relocates the $F_Q(Z)$ equations and supporting description to the Bases. The equations for steady state and non-equilibrium $F_Q(Z)$ limits, along with the equation for determining the percent by which $F_Q(Z)$ exceeds its limits, are being relocated to the Bases. The relocation of these items to the Bases is consistent with guidance obtained from the North Anna and Kewaunee ITS, along with NUREG-1431. The definition of these equations within the TS is deemed unnecessary based on the guidance of NUREG-1431, as they are also described in the TS Bases and the COLR.

TS 3.2.2.1 – ACTION, step b

The proposed changes incorporate a modified version of the interim actions identified in NSAL-09-5, Rev. 1 (Reference 1), in the event that SR 4.2.2.1.2.c is not satisfied. This approach was determined by Dominion analysis to most appropriately address the issues in NSAL-09-5, Rev. 1 for MPS3. The allowable operating space that applies to TS 3.2.2.1 – ACTION, step b, is relocated to the COLR. A new table, entitled "Required Operating Space Reductions for $F_Q(z)$ Exceeding its Non-Equilibrium Limits," will be added to the COLR to quantify the required THERMAL POWER and AFD limits associated with different amounts of non-equilibrium $F_Q(Z)$ margin improvement (1%, 2%, etc.). If TS 3.2.2.1 – ACTION, step b is entered, the operating space as defined in the new COLR table will ensure that sufficient margin exists. Including the numerical specification of the operating space in the COLR provides greater assurance that the recommended actions are acceptable without regard to the specific power distribution control methodology. The proposed change can be applied under either the Westinghouse (RAOC) or Dominion (RPDC) power distribution control methodologies for a given reload cycle. Table 3 below presents a sample of the proposed table to be included in the COLR.

Note that the values provided in the sample table below are only intended to provide a representative example of typical reload values. The determination and verification of the required non-equilibrium $F_Q(Z)$ margin improvements and the corresponding required reductions in the Thermal Power Limit and AFD Bands will be performed on a reload specific basis in accordance with approved methodologies of WCAP-10216-P-A or VEP-NE-1-A listed in Technical Specification 6.9.1.6.b.

Table 3

Required Operating Space Reductions For $F_Q(Z)$ Exceeding Its Non-Equilibrium Limits

Required Non-Equilibrium $F_Q(Z)$ Margin Improvement (%)	Required THERMAL POWER Limit (% RTP)	Required Negative Band AFD Reduction (% AFD)	Required Positive Band AFD Reduction (% AFD)
≤ 1%	≤ 97%	≥ 2%	≥ 4%
> 1 and ≤ 2%	≤ 95%	≥ 3%	≥ 5%
> 2 and ≤ 3%	≤ 93%	≥ 4%	≥ 6%
> 3%	≤ 50%	N/A	N/A

MPS3 is currently operating with compensatory actions in accordance with the Westinghouse recommendations contained in NSAL-09-5, Rev. 1 (Reference 1) and 06-IC-03 (Reference 2). Recommendations for TS changes were included in References 1 and 2. Westinghouse subsequently submitted WCAP-17661 (Reference 4) for NRC review and approval which is the proposed resolution for the issues described in NSAL-09-5, Rev. 1 for Westinghouse plants. Following review of WCAP-17661 and the requirements for implementation, and based on the

differences between Westinghouse and Dominion methodologies, Dominion has decided to pursue the proposed TS changes that more closely align with NSAL-09-5, Rev. 1 and 06-IC-03, rather than those in WCAP-17661.

SR 4.2.2.1.2.c

The proposed change notes that $F_Q^M(Z)$ will satisfy non-equilibrium limits as specified in the COLR. The limit equations and associated description are already described in the COLR, making this information redundant in the TS. However, to be consistent with the guidance of NUREG-1431, the equations are being relocated to the TS Bases.

SR 4.2.2.1.2.e

Westinghouse NSAL-15-1 (Reference 8) identified a technical concern which is applicable to MPS3 SR 4.2.2.1.2.e. Under some conditions, the required actions in this step may not provide assurance that the non-equilibrium $F_Q(Z)$ LCO limit will be met between the performance of the required 31 Effective Full Power Day (EFPD) core power distribution surveillances. These conditions pertain to scenarios in which the value of $F_Q^M(Z)$ has decreased since its previous determination, however, due to the W(z) values, the margin to the non-equilibrium $F_Q(Z)$ LCO limit (SR 4.2.2.1.2.c) may have decreased. The net result of this scenario could result in the unexpected loss of $F_Q(Z)$ margin to, or exceeding, the SR 4.2.2.1.2.c limits between 31 EFPD surveillance intervals. The proposed change conservatively addresses the issue identified in NSAL-15-1 by revising SR 4.2.2.1.2.e to apply the burnup dependent penalty factor, specified in the COLR, to all power distribution surveillances in which confirmation of SR 4.2.2.1.2.c is required, regardless of the trend in the value of $F_Q^M(Z)$.

This revised surveillance requirement will ensure the bounding analytical penalty factor is always accounted for in the verification of SR 4.2.2.1.2.c. In the event that the value of $F_Q^M(Z)$ computed from SR 4.2.2.1.2.e(1) does not satisfy SR 4.2.2.1.2.c, SR 4.2.2.1.2.e(2) will continue to allow for $F_Q(Z)$ surveillances to be performed within 7 EFPD intervals in lieu of requiring the application of the burnup dependent penalty to satisfy SR 4.2.2.1.2.c. It is intended that SR 4.2.2.1.2.e(2) will be exited when SR 4.2.2.1.2.e(1) satisfies SR 4.2.2.1.2.c and the surveillance interval can be returned to 31 EFPD. A discussion of the technical intent of SR 4.2.2.1.2.e is provided in the TS Bases markups which are provided for information only in Attachment 3.

SR 4.2.2.1.2.f

The proposed change notes that the core plane regions for which the limits in SR 4.2.2.1.2.c and 4.2.2.1.2.e are not applicable will be defined in the Bases. This change is consistent with the recommendation in Westinghouse Notice 06-IC-03 (Reference 2) and NUREG-1431, Rev. 4 (Reference 3). SR 4.2.2.1.2.f states that the upper and lower 15% of the core are not applicable for surveillance of the transient F_Q limit. The reason for this is that, historically, the chances of the limiting F_Q margin occurring in the top and bottom of the core is low and accurate measurements in those regions are more difficult. The intent of 06-IC-03 was to inform utilities that it is probable for the minimum F_Q margin to occur near the top or bottom of the core. In response to the information

in 06-IC-03, the proposed change increases the core plane regions for which the limits apply and reduces the "not applicable" portion at top and bottom to 8%. Reload methodology confirms transient F_Q margin over the entire core height and the "not applicable" region can be adjusted larger or smaller, as necessary, to ensure peak transient F_Q is not in this region. Moving the "not applicable" region to the Bases allows this adjustment.

3.3 TS 3.2.3.1.b(5) - RCS Flow Rate (measurement uncertainty)

TS 3.2.3.1.b(5) currently states the numerical value (2.4%) of the flow measurement uncertainty that is included in the reactor coolant system (RCS) flow rate limits presented in TS 3.2.3.1.a. The key aspect of TS 3.2.3.1.b(5) is to indicate that the measured value of RCS flow rate be used for comparison to its limit since measurement uncertainty has already been accounted for in establishing the limit value. In general, the value of the uncertainty may depend on plant instrumentation or evaluation methodology, and is not material to the fundamental concept of TS 3.2.3.1.b(5).

As defined in 10 CFR 50.36, RCS flow uncertainty does not meet any of the categories for items required to be in TS. TS categories are: 1) Safety limits, limiting safety system settings, and limiting control settings, 2) Limiting conditions for operation, 3) Surveillance requirements, 4) Design features, 5) Administrative controls, 6) Decommissioning, 7) Initial notification, or 8) Written Reports. The parameter that does meet the requirements of 10 CFR 50.36 related to the RCS flow uncertainty is the RCS Total Flow Rate contained in TS 3.2.3.1.a. The RCS Total Flow Rate is used in the Final Safety Analysis Report (FSAR) Chapter 15, "Accident Analyses," initiated from nominal conditions (statistical events) as it accounts for the RCS flow uncertainty. The thermal design flow rate is used in the FSAR Chapter 15, "Accident Analyses," initiated from deterministic conditions (nominal biased for uncertainty). Thus, the difference between the thermal design flow rate and the RCS total flow rate bounds the RCS flow uncertainty. This is consistent with reload evaluation methods of WCAP-9272-P-A and VEP-FRD-42-A as well as the statistical DNB methodologies of WCAP-11397-P-A and VEP-NE-2-A. In addition, the RCS flow uncertainty does not appear in the Standard Technical Specifications for Westinghouse plants documented in NUREG-1431, Volume 1, Revision 4.0 (Reference 3). Therefore, it is acceptable to remove the value of the RCS flow uncertainty from the MPS3 TS.

3.4 SR 4.2.3.1.3.a - RCS Flow Rate (precision heat balance)

This change provides operational flexibility to enable the operators and reactor engineers to focus on key activities immediately following a refueling outage. The relaxation of the time requirement to perform the precision heat balance from 24 hours to 7 days after reaching 90% of RATED THERMAL POWER is reasonable to: establish stable operating conditions, install the test equipment, perform the test, and analyze the results. The 7 day period is consistent with that approved for Surry Power Station (Reference 5) and less than the 30 day period approved for North Anna Power Station (Reference 6). Any gross changes in core flow rate from the previous cycle would be detected well before the unit reaches 90% power. Thus, the relaxation in the time does not pose any undue risk to the health and safety of the public.

3.5 TS 6.9.1.6.a – Core Operating Limits Report

The proposed changes to Items 7 and 8 delete the terminology associated with specific power distribution control methodology and base load operation. These changes conform with the changes described in LAR Section 2.2 and evaluated in LAR Section 3.2 where the proposed changes to TS 3/4.2.1.1 and 3/4.2.2.1 eliminate base load operation and relocate from TS the equations and terminology for either RAOC or RPDC transient multiplication factors (e.g., W(Z)). The proposed change to Item 9 aligns the associated TS subsection with that specified in the COLR (TS 3/4.2.3.1 versus TS 3/4.2.3).

3.6 TS 6.9.1.6.b – Core Operating Limits Report

Statement Preceding TS 6.9.1.6.b Reference List

The proposed change to the statement preceding the reference list in TS 6.9.1.6.b provides clarification that the reference methodologies used for a reload core will be specifically identified in the cycle-specific COLR. The language for the proposed change is taken from the Surry TS and is consistent with the "Reviewer's Note" in TS 5.6.3.b (Page 5.6-2) of the Westinghouse STS (NUREG-1431, Rev. 4, Reference 3). The addition of this wording requires that the cycle-specific COLR identify the full reference citation of the topical reports used to support that cycle. This will involve citations that include the full title, revision, date, and supplements, as applicable.

Addition of Approved Dominion Methodologies to the TS 6.9.1.6.b Reference List

MPS3 TS 6.9.1.6.b currently states that the analytical methods used to determine core operating limits shall be those previously reviewed and approved by the NRC. This TS also provides a list of NRC-approved analytical methods for MPS3. In order for Dominion to use the analytical methods described in Attachment 4, NRC-approved methodologies VEP-FRD-42, VEP-NE-1, VEP-NE-2, and DOM-NAF-2, require addition to the list in TS 6.9.1.6.b. In addition, the MPS3-specific application of the core thermal hydraulic analysis methodology, DOM-NAF-2, also requires NRC approval of the Westinghouse RFA-2 fuel-specific DNBR statistical design limits (SDLs).

Attachment 4 describes the Dominion nuclear core design and safety analysis methods that will be applied to MPS3. Items a, b, e and f below are added to the list in TS 6.9.1.6.b (see markup in Attachment 2).

The Dominion nuclear core design methods that will be applied to MPS3 are as follows:

a) VEP-FRD-42-A, "Reload Nuclear Design Methodology."

b) VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

c) DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations."

The Dominion safety analysis methods that will be applied to MPS3 are as follows:

d) VEP-FRD-41-P-A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code."

e) VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

f) DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D ComputerCode."

The addition of the analytical methods specified above to TS 6.9.1.6.b, permits the use of Dominion analysis methodologies to perform nuclear core design and safety analyses for MPS3.

Reformatting of the TS 6.9.1.6.b Reference List

The language in TS 6.9.1.6.a establishes a requirement to identify in the COLR each of the applicable TS 6.9.1.6.b references used in a specific reload core. This provides positive identification of the NRC-approved methods used to establish the core operating limits for each reload core.

The specifications listed under each reference in TS 6.9.1.6.b will be reformatted using bullets to improve readability. In addition, minor changes to these specifications are also being made to conform to the usage that is appropriate for either Westinghouse or Dominion references.

In TS 6.9.1.6.b, Reference 1, TS 2.1.1 is substituted in the list for TS 2.1.1.1—Departure from Nuclear Boiling Ratio and 2.1.1.2—Peak Fuel Centerline Temperature, to align with Item 1 in TS 6.9.1.6.a for which the reference is applicable. In addition, TS 3.2.5 and 3.3.5 are added to the list since they did not previously appear in Reference 1 of TS 6.9.1.6.b.

References 1 and 4 are being retained in TS 6.9.1.6.b since these methodologies are applicable to Westinghouse for establishing core operating limits and may be used for a specific core during the transition to Dominion methods.

TS 6.9.1.6.b, References 5, 6, 7, 8, 9, 10, 16, 17, 18 and 19, are Westinghouse topical reports which document methodologies that are independent of the scope for nuclear safety analysis and core design methods which Dominion is applying. These references remain applicable for use with either Westinghouse or Dominion reload core design methods.

Dominion proposes to delete TS 6.9.1.6.b, References 2, 3, 11, 12, 13, 14 and 15. These references do not describe a methodology that establishes core operating limits and therefore should be removed from the reference list. More discussion is provided below.

The references used in a cycle-specific COLR will be a subset of the TS 6.9.1.6.b methodologies that are applicable to the specific reload cycle. If Westinghouse reload methods are used, then Westinghouse reload methods shall be listed in the cycle-specific COLR. If Dominion reload methods are used, then Dominion reload methods shall be listed.

Deletion of References from TS 6.9.1.6.b

A methodology in the COLR reference list is intended to satisfy these criteria: 1) it is used to determine core operating limits, and 2) it has been previously approved by the NRC. This interpretation is consistent with the intent and discussion included in Generic Letter 88-16 (Reference 7). GL 88-16 additionally states:

> "Generally, the methodology for determining cycle-specific parameter limits is documented in an NRC-approved Topical Report or in a plant-specific submittal."

MPS3 TS 6.9.1.6.b currently states that the analytical methods used to determine core operating limits shall be those previously reviewed and approved by the NRC. According to the intent of GL-88-16, TS 6.9.1.6.b references should also be those that describe a methodology that is applicable for use in establishing core operating limits. TS 6.9.1.6.b currently contains various references, only some of which are used to establish core operating limits. Therefore, DNC proposes to amend TS 6.9.1.6.b by deleting References 2, 3, 11, 12, 13, 14 and 15 since these references represent historical correspondence for which no use is specified in the cycle-specific COLR.

The language in TS 6.9.1.6.a establishes a requirement to identify in the COLR each of the applicable TS 6.9.1.6.b references used in a specific reload core. A Dominion and Westinghouse methodology for each TS 6.9.1.6.a item has been identified in Attachment 2. With removal of the aforementioned references, an approved methodology reference is available for each of the items in TS 6.9.1.6.a. Citation of specific methods from the references in TS 6.9.1.6.b in the cycle-specific COLR provides positive identification of the NRC-approved methods used to establish the core operating limits for each reload core.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Criteria

10 CFR 50.36, Technical Specifications, paragraphs (c)(2), (c)(3), and (c)(5) states that technical specifications will include limiting conditions for operations, surveillance requirements, and administrative controls. Limiting conditions for operations are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Administrative controls 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.

4.2 No Significant Hazards Consideration

DNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by addressing the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The Dominion analysis methods do not make any contribution to the potential accident initiators and thus do not increase the probability of any accident previously evaluated. The use of the approved Dominion analysis methods will not increase the probability of an accident because plant systems, structures, and components (SSC) will not be affected or operated in a different manner, and system interfaces will not change.

   Since the applicable safety analysis and nuclear core design acceptance criteria will be satisfied when the Dominion analysis methods are applied to MPS3, the use of the approved Dominion analysis methods does not increase the potential consequences of any accident previously evaluated. The use of the approved Dominion methods will not result in a significant impact on normal operating plant releases, and will not increase the predicted radiological consequences of postulated accidents described in the FSAR. The proposed resolution of Westinghouse notification documents NSAL-09-5, Rev. 1, 06-IC-03 and NSAL-15-1 is intended to address deficiencies identified within the existing MPS3 Technical Specifications to return them to their as designed function and does not result in actions that would increase the probability of any accident previously evaluated.

   Therefore, the proposed amendment does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The use of Dominion analysis methods and the Dominion statistical design limit (SDL) for fuel departure from nucleate boiling ratio (DNBR) and fuel critical heat flux (CHF) does not impact any of the applicable core design criteria. All pertinent licensing basis limits and acceptance criteria will continue to be met. Demonstrated adherence to these limits and acceptance criteria precludes new challenges to SSCs that might introduce a new type of accident. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the Dominion methods does not involve

any alteration to plant equipment or procedures that might introduce any new or unique operational modes or accident precursors. The proposed resolution of Westinghouse notification documents NSAL-09-5, Rev. 1, 06-IC-03 and NSAL-15-1 does not involve the alteration of plant equipment or introduce unique operational modes or accident precursors.

Therefore, the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Nuclear core design and safety analysis acceptance criteria will continue to be satisfied with the application of Dominion methods. Meeting the analysis acceptance criteria and limits ensure that the margin of safety is not significantly reduced. Nuclear core design and safety analysis acceptance criteria will continue to be satisfied with the application of Dominion methods. In particular, use of VIPRE-D with the proposed safety limits provides at least a 95% probability at a 95% confidence level that DNBR will not occur (the 95/95 DNBR criterion). The required DNBR margin of safety for MPS3, which is the margin between the 95/95 DNBR criterion and clad failure, is therefore not reduced. The proposed resolution of Westinghouse notification documents NSAL-09-5, Rev. 1, 06-IC-03 and NSAL-15-1 does not propose actions that would result in a significant reduction in margin to safety.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above information, DNC concludes that the proposed license amendment involves no significant hazards consideration under the criteria set forth in 10 CFR 50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

4.3 Precedents

The requested licensing actions and proposed changes to the MPS3 TS are fundamentally the same as the following approved changes associated with previous application of Dominion methods:

1. Letter from Patrick D. Milano (USNRC) to David A. Christian (Dominion), "Kewaunee Power Station – Safety Evaluation for Topical Report DOM-NAF-5 (TAC NO. MD2829)," August 30, 2007 (ADAMS Accession No. ML072290373).

2. Letter from Patrick D. Milano (USNRC) to David A. Christian (Dominion), "Kewaunee Power Station – Issuance of Amendment Re: Nuclear Core Design and Safety Analysis Methods (TAC NO. MD6824)," March 28, 2008 (ADAMS Accession No. ML080630529).

3. Letter from V. Sreenivas (USNRC) to David A. Heacock (Dominion), "North Anna and Surry Power Stations Units 1 and 2, Issuance of Amendments Regarding Addition of an Analytical Methodology to the North Anna and Surry Core Operating Limits Reports and an Increase in to the Surry Minimum Temperature for Criticality (TAC NOS. MF 2364, MF2365, MF2366, and MF2367," August 12, 2014 (ADAMS Accession No. ML14169A359).

Precedent 1 and 2 are the NRC approvals for application of Dominion methods to Kewaunee Power Station (KPS). The MPS3 application applies the same methods used in the KPS submittal. Precedent 3 is the NRC approval for the application of the Dominion VIPRE-D code with the ABB-NV and WLOP DNB correlations for North Anna and Surry, in the form of Appendix D to the Dominion Fleet Report DOM-NAF-2-A. The current MPS3 request also includes application of this methodology.

5.0 ENVIRONMENTAL CONSIDERATIONS

DNC has reviewed the proposed license amendment for environmental considerations. The proposed license amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Westinghouse Nuclear Safety Advisory Letter, NSAL-09-5, Rev. 1, "Relaxed Axial Offset Control FQ Technical Specification Actions," September 23, 2009.

2. Westinghouse Notice "06-IC-03, F_Q and F_{xy} Surveillance Zone Issue," February 21, 2006.

3. NUREG-1431, Revision 4, Vol. 1 and 2, "Standard Technical Specifications – Westinghouse Plants."

4. WCAP-17661-P, Revision 1, "Improved RAOC and CAOC FQ Surveillance Technical Specifications," November 2013.

5. Letter from K. Cotton (USNRC) to D. A. Heacock (Dominion), "Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Request for Technical Specification Revisions Related to the Core Operating Limits Report (TAC Nos. ME2591 and ME2592)," dated October 19, 2010; corrected by Letter from K. Cotton (USNRC) to D. A. Heacock (Dominion), "Surry Power Station, Unit Nos. 1 and 2, Correction to Amendments Regarding Technical Specification Revisions Related to the Core Operating Limits Report (TAC Nos. ME2591 and ME2592)," October 21, 2010.

6. Letter from S. Monarque (USNRC) to D. A. Christian (VEPCO), "North Anna Power Station, Units 1 and 2 – Issuance of Amendments Re: Conversion to Improved Technical Specifications (TAC NOS. MB0799 and MB0800)," April 5, 2002.

7. Generic Letter No. 88-16, "Removal of Cycle-specific Parameter Limits from Technical Specifications," October 3, 1988.

8. Westinghouse Nuclear Safety Advisory Letter, NSAL-15-1, "Heat Flux Hot Channel Factor Technical Specification Surveillance," February 3, 2015.

ATTACHMENT 2

MARKED-UP TECHNICAL SPECIFICATIONS PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

VENTING

1.39 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

SPENT FUEL POOL STORAGE PATTERNS:

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 AND 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

(a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.

(b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

~~ALLOWED POWER LEVEL~~

Deleted

1.43 ~~APL^{ND} is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.~~

1.44 ~~APL^{BL} is the maximum allowable power level when transitioning into base load operation.~~

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR) ~~for Relaxed Axial Offset Control (RAOC) operation, or~~

Deleted

b. ~~Within the target band about the target flux difference during base load operation, specified in the COLR.~~

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*.

ACTION:

W

a. ~~For RAOC operation~~ with the indicated AFD outside of the applicable limits specified in the COLR,

1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux--High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

Deleted

b. ~~For base load operation above APL^{ND} with the indicted AFD outside of the applicable target band about the target flux differences:~~

~~1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or~~

~~2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue base load operation within 30 minutes.~~

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

* See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS

4.2.1.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at the frequency specified in the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE:

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

Deleted

4.2.1.1.3 ~~When in base load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement at the frequency specified in the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.~~

Deleted

4.2.1.1.4 ~~When in base load operation, the target flux difference shall be updated at the frequency specified in the Surveillance Frequency Control Program by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.1.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.~~

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

$F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

3.2.2.1 ~~$F_Q(Z)$ shall be limited by the following relationships:~~

$$\cancel{F_Q(Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5}$$

$$\cancel{F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5}$$

~~F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) provided in the CORE OPERATING LIMITS REPORT (COLR).~~

~~Where:~~ $\cancel{P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}}$, ~~and~~

~~K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.~~

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit: W

a. ~~For RAOC operation~~ with Specification 4.2.2.1.2.b not being satisfied ~~or for base load operation with Specification 4.2.2.1.4.b not being satisfied~~:

(1) Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, and

(2) Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by item (1) above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

b. ~~For RAOC operation~~ W with Specification 4.2.2.1.2.c not being satisfied, ~~one~~ all of the following ACTIONS shall be taken:

INSERT A

(1) ~~Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by at least 1% AFD for each percent $F_Q(Z)$ exceeds its limits. Within 8 hours, reset the AFD alarm setpoints to these modified limits, or~~

Deleted

(2) ~~Verify that the requirements of Specification 4.2.2.1.3 for base load operation are satisfied and enter base load operation.~~

~~Where it is necessary to calculate the percent that $F_Q(Z)$ exceeds the limits for item (1) above, it shall be calculated as the maximum percent over the core height (Z), consistent with Specification 4.2.2.1.2.f, that $F_Q(Z)$ exceeds its limit by the following expression:~~

$$\left[\frac{\dfrac{F_Q^M(Z) \times W(Z)}{RTP}}{\dfrac{F_Q}{P} \times K(Z)} - 1\right] \times 100 \text{ for } P > 0.5$$

$$\left[\frac{\dfrac{F_Q^M(Z) \times W(Z)}{RTP}}{\dfrac{F_Q}{0.5} \times K(Z)} - 1\right] \times 100 \text{ for } P \leq 0.5$$

Deleted

c. ~~For base load operation with Specification 4.2.2.1.4.c not being satisfied, one of the following ACTIONS shall be taken:~~

~~(1) Place the core in an equilibrium condition where the limit in 4.2.2.1.4.c is satisfied, and remeasure $F_Q^M(Z)$, or~~

INSERT "A"

a. Within 15 minutes, control the AFD to within the new reduced AFD limits specified in the COLR that restores $F_Q(Z)$ to within its limits, and

b. Reduce the THERMAL POWER by the amount specified in the COLR that restores $F_Q(Z)$ to within its limits within 4 hours, and

c. Reduce the Power Range Neutron Flux – High Trip Setpoints by $\geq$ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

d. Reduce the Overpower ΔT Trip Setpoints by $\geq$1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

e. Within 8 hours, reset the AFD Alarm Setpoints to the modified limits, and

f. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by (1)b. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

LIMITING CONDITION FOR OPERATION (Continued)

~~(2) Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds its limit shall be calculated as the maximum percent over the core height (Z), consistent with Specification 4.2.2.1.4.f, by the following expression:~~

$$\left[\left[\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(Z)}\right] - 1\right] \times 100 \text{ for } P \geq APL^{ND}$$

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 ~~For RAOC operation,~~ $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Evaluate the computed heat flux hot channel factor by performing both of the following:

(1) Determine the computed heat flux hot channel Factor, $F_Q^M(Z)$ by increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and

(2) Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e., 0-100% inclusive.

SURVEILLANCE REQUIREMENTS (Continued)

Verify $F_Q^M(Z)$ satisfies the non-equilibrium limits specified in the COLR.

c. ~~Satisfying the following relationship~~:

$$\cancel{F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \quad \text{for } P > 0.5}$$

$$\cancel{F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{W(Z) \times 0.5} \quad \text{for } P \leq 0.5}$$

~~where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and $W(Z)$ is the cycle-dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, $K(Z)$, and $W(Z)$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.6.~~

d. Measuring $F_Q^M(Z)$ according to the following schedule:

(1) Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined,*** or

(2) At the frequency specified in the Surveillance Frequency Control Program, whichever occurs first.

e. ~~With the maximum value of~~

Compliance with the non-equilibrium limits shall be conservatively accounted for during intervals between $F_Q^M(Z)$ measurements by performing either of the following:

$$\cancel{\frac{F_Q^M(Z)}{K(Z)}}$$

~~over the core height (Z) increasing since the previous determination of $F_Q^M(Z)$, either of the following ACTIONS shall be taken:~~

(1) Increase $F_Q^M(Z)$ by an appropriate factor specified in the COLR and verify that this value satisfies ~~the relationship in~~ Specification 4.2.2.1.2.c, or

***During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map outlined.

SURVEILLANCE REQUIREMENTS (Continued)

Verify

satisfies its limits

(2) $F_Q^M(Z)$ ~~shall be measured~~ at least once per 7 Effective Full Power Days ~~until two successive maps indicate that the maximum value of~~

$$\cancel{\frac{F_Q^M(Z)}{K(Z)}}$$

~~over the core height (Z) is not increasing~~.

f. The limits specified in Specifications 4.2.2.1.2c and 4.2.2.1.2e above are not applicable in the ~~following~~ core plane regions:

defined in the Bases

~~(1) Lower core region from 0% to 15%, inclusive.~~

~~(2) Upper core region from 85% to 100%, inclusive.~~

Deleted

4.2.2.1.3 ~~Base load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:~~

~~a. Prior to entering base load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.1.2 for at least the previous 24 hours. Maintain base load operation surveillance (AFD within the target band limit about the target flux difference of Specification 3.2.1.1) during this time period. Base load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.1.4. APL^{BL} is defined as the minimum value of:~~

$$\cancel{APL^{BL} = \frac{F_Q^{RTP} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \times 100\%}$$

~~over the core height (Z) where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is F_Q^{RTP}. $W(Z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.~~

~~b.~~ ~~During base load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.1.3.a shall be satisfied before reentering base load operation.~~

Deleted

4.2.2.1.4 ~~During base load operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:~~

~~a.~~ ~~Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.~~

~~b.~~ ~~Evaluate the computed heat flux hot channel factor by performing both of the following:~~

~~(1)~~ ~~Determine the computed heat flux hot channel factor, $F_Q^M(Z)$, by increasing the measured $F_Q^M(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and~~

~~(2)~~ ~~Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e., 0 - 100% inclusive.~~

~~c.~~ ~~Satisfying the following relationship:~~

$$\cancel{F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)_{BL}} \quad \text{for } P > APL^{ND}}$$

~~where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and $W(Z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.~~

~~d.~~ ~~Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:~~

~~(1)~~ ~~Prior to entering base load operation after satisfying Section 4.2.2.1.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative THERMAL POWER having been maintained above APL^{ND} for the 24 hours prior to mapping, and~~

~~(2)~~ ~~At the frequency specified in the Surveillance Frequency Control Program.~~

SURVEILLANCE REQUIREMENTS (Continued)

~~e. With the maximum value of~~

$$\cancel{\frac{F_Q^M(Z)}{K(Z)}}$$

~~over the core height (Z) increasing since the previous determination of $F_Q^M(Z)$, either of the following ACTIONS shall be taken:~~

~~1) Increase $F_Q^M(Z)$ by appropriate factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.1.4.e, or~~

~~2) $F_Q^M(Z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive maps indicate that the maximum value of~~

$$\cancel{\frac{F_Q^M(Z)}{K(Z)}}$$

~~over the core height (Z) is not increasing.~~

~~f. The limits specified in 4.2.2.1.4.e and 4.2.2.1.4.e are not applicable in the following core plane regions:~~

~~1) Lower core region 0% to 15%, inclusive.~~

~~2) Upper core region 85% to 100%, inclusive.~~

4.2.2.1.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specifications 4.2.2.1.2 ~~or 4.2.2.1.4~~, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate ≥ 363,200 gpm and greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR), and

b. $F^N_{\Delta H} \leq F^{RTP}_{\Delta H} \quad [1.0 + PF_{\Delta H}(1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the COLR,

4) $PF_{\Delta H}$ = The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties ~~of 2.4%~~ for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate to within the limits specified above and in the COLR and $F^N_{\Delta H}$ to within the above limit, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that the RCS total flow rate is restored to within the limits specified above and in the COLR and $F^N_{\Delta H}$ is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At the frequency specified in the Surveillance Frequency Control Program.

4.2.3.1.3 The RCS total flow rate shall be determined to be within the acceptable range by:

a. Verifying by precision heat balance that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR within ~~24 hours~~ 7 days after reaching 90% of RATED THERMAL POWER after each fuel loading, and

MONTHLY OPERATING REPORTS

6.9.1.5 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Reactor Core Safety Limit for Specification 2.1.1.
2. Overtemperature ΔT and Overpower ΔT setpoint parameters for Specification 2.2.1.
3. SHUTDOWN MARGIN for Specifications 3/4.1.1.1.1, 3/4.1.1.1.2, and 3/4.1.1.2.
4. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.
5. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5.
6. Control Rod Insertion Limits for Specification 3/4.1.3.6.
7. AXIAL FLUX DIFFERENCE Limits~~, target band, and~~ ~~APL^{ND}~~ for Specification 3/4.2.1.1.
8. Heat Flux Hot Channel Factor Limits~~, K(z), W(z), APL^{ND}, and $W(z)_{BL}$~~ for Specification 3/4.2.2.1.
9. RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3.1
10. DNB Parameters for Specification 3/4.2.5.
11. Shutdown Margin Monitor minimum count rate for Specification 3/4.3.5.
12. Boron Concentration for Specification 3/4.9.1.1.

CORE OPERATING LIMITS REPORT (Cont.)

INSERT B

in Specification 6.9.1.6.a

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC ~~in:~~

INSERT C

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary). (Methodology for Specifications ~~2.1.1.1--Departure from Nucleate Boiling Ratio, 2.1.1.2--Peak Fuel Centerline Temperature, 3.1.1.3--Moderator Temperature Coefficient, 3.1.3.5--Shutdown Bank Insertion Limit, 3.1.3.6--Control Bank Insertion Limits, 3.2.1--AXIAL FLUX DIFFERENCE, 3.2.2--Heat Flux Hot Channel Factor, 3.2.3--Nuclear Enthalpy Rise Hot Channel Factor, 3.1.1.1.1, 3.1.1.1.2, 3.1.1.2 -- SHUTDOWN MARGIN, 3.9.1.1--Boron Concentration.)~~

Deleted

2. ~~T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.~~

Deleted

3. ~~NUREG-800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981 Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Revision 2, July 1981.~~

4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1 ~~AXIAL FLUX~~ .1 DIFFERENCE ~~[Relaxed Axial Offset Control]~~ and 3.2.2 ~~Heat~~ Flux Hot Channel Factor ~~[W(z) surveillance requirements for F_Q Methodology]~~.)

5. WCAP-12945-P-A, "CODE QUALIFICATION DOCUMENT FOR BEST ESTIMATE LOCA ANALYSIS," (W Proprietary). (Methodology for Specification 3.2.2 .1 --Heat Flux Hot Channel Factor.)

6. WCAP-16009-P-A, "REALISTIC LARGE-BREAK LOCA EVALUATION METHODOLOGY USING THE AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD (ASTRUM)," (W Proprietary). (Methodology for Specification 3.2.2 .1 --Heat Flux Hot Channel Factor.)

7. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," (W Proprietary). INSERT D

8. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," (W Proprietary). (Methodology for Specification 3.2.2 .1 - Heat Flux Hot Channel Factor.)

9. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Methodology for Specification 3.2.2 .1 - Heat Flux Hot Channel Factor.)

10. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Methodology for Specification 3.2.2 .1 - Heat Flux Hot Channel Factor.)

INSERT "B"

and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements).

INSERT "C"

- 2.1.1 Reactor Core Safety Limits
- 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
- 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
- 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
- 3.1.1.3 Moderator Temperature Coefficient
- 3.1.3.5 Shutdown Rod Insertion Limit
- 3.1.3.6 Control Rod Insertion Limits
- 3.2.1.1 AXIAL FLUX DIFFERENCE
- 3.2.2.1 Heat Flux Hot Channel Factor
- 3.2.3.1 RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
- 3.9.1.1 REFUELING Boron Concentration
- 3.2.5 DNB Parameters
- 3.3.5 Shutdown Margin Monitor

INSERT "D"

Methodology for Specification:

- 3.1.1.3 - Moderator Temperature Coefficient

CORE OPERATING LIMITS REPORT (Cont.)

Deleted

11. ~~Letter from V. L. Rooney (USNRC) to J. F. Opeka, "Safety Evaluation for Topical Report, NUSCO-152, Addendum 4, 'Physics Methodology for PWR Reload Design,' TAC No. M91815," July 18, 1995.~~

Deleted

12. ~~Letter from E. J. Mroczka to the USNRC, "Proposed Changes to Technical Specifications, Cycle 4 Reload Submittal - Boron Dilution Analysis," B13678, December 4, 1990.~~

Deleted

13. ~~Letter from D. H. Jaffe (USNRC) to E. J. Mroczka, "Issuance of Amendment (TAC No. 77924)," March 11, 1991.~~

Deleted

14. ~~Letter from M. H. Brothers to the USNRC, "Proposed Revision to Technical Specification, SHUTDOWN MARGIN Requirements and Shutdown Margin Monitor OPERABILITY for MODES 3, 4, and 5 (PTSCR 3-16-97), B16447, May 9, 1997.~~

Deleted

15. ~~Letter from J. W. Anderson (USNRC) to M. L. Bowling (NNECO), "Issuance of Amendment - Millstone Nuclear Power Station, Unit No. 3 (TAC No. M98699)," October 21, 1998.~~

16. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis."

INSERT E

17. WCAP-10054-P-A, ~~Addendum~~ 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation ~~Model."~~

18. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature DT Trip Functions," (Westinghouse Proprietary Class 2). (Methodology for Specifications ~~2.1.1 and~~ 2.2.1 -- Overtemperature ΔT and Overpower ΔT Setpoints.)

19. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

INSERT F

INSERT "E"

Methodology for Specification:

- 3.2.2.1 - Heat Flux Hot Channel Factor

INSERT "F"

20. VEP-FRD-42-A, "Reload Nuclear Design Methodology."
    Methodology for Specifications:
    - 2.1.1 Reactor Core Safety Limits
    - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
    - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
    - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
    - 3.1.1.3 Moderator Temperature Coefficient
    - 3.1.3.5 Shutdown Rod Insertion Limit
    - 3.1.3.6 Control Rod Insertion Limits
    - 3.2.2.1 Heat Flux Hot Channel Factor
    - 3.2.3.1 Nuclear Enthalpy Rise Hot Channel Factor
    - 3.3.5 Shutdown Margin Monitor
    - 3.9.1.1 REFUELING Boron Concentration

21. VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

    Methodology for Specifications:
    - 3.2.1.1 AXIAL FLUX DIFFERENCE
    - 3.2.2.1 Heat Flux Hot Channel Factor

22. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

    Methodology for Specifications:
    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

23. DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."

    Methodology for Specifications:
    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

ATTACHMENT 3

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES

(FOR INFORMATION ONLY)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

AXIAL FLUX DIFFERENCE (Continued)

~~At power levels below APL^{ND}, the limits on AFD are defined in the COLR consistent with the Relaxed Axial Offset Control (RAOC) operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.~~

~~At power levels greater than APL^{ND}, two modes of operation are permissible: (1) RAOC, the AFD limit of which are defined in the COLR, and (2) base load operation, which is defined as the maintenance of the AFD within COLR specifications band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. To allow operation at the maximum permissible power level, the base load operating procedure restricts the indicated AFD to relatively small target band (as specified in the COLR) and power swings (APL^{ND} ≤ power ≤ APL^{BL} or 100% RATED THERMAL POWER, whichever is lower). For base load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the base load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the base load procedure. After the waiting period, extended base load operation is permissible.~~

The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are~~: (1)~~ outside the allowed delta-I power operating space (as specified in the COLR ~~for RAOC operation), or (2) outside the allowed delta-I target band (for base load operation~~). These alarms are active when power is greater than ~~(1)~~ 50% of RATED THERMAL POWER ~~(for RAOC operation), or~~

~~AXIAL FLUX DIFFERENCE (Continued)~~

~~(2) APL^{ND} (for base load operation). Penalty deviation minutes for base load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed~~.

3/4.2.2 AND 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ±12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

The $F^N_{\Delta H}$ as calculated in Specification 3.2.3.1 is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

The RCS total flow rate and $F^N_{\Delta H}$ are specified in the CORE OPERATING LIMITS REPORT (COLR) to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow rate, that is based on 10% steam generator tube plugging, is retained in the Technical Specifications.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset the effect of rod bow and any other DNB penalties that may occur. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor, $F_Q(Z)$, is measured periodically in accordance with the Surveillance Frequency Control Program using the incore detector system. These measurements are generally taken with the core at or near steady state conditions. Using the measured three dimensional power distributions, it is possible to derive $F_Q^M(Z)$, a computed value of $F_Q(Z)$. ~~However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations.~~

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~~To account for these possible variations, the steady state limit of $F_Q(Z)$ is adjusted by an elevation dependent factor appropriate to either RAOC or base load operation, W(Z) or $W(Z)_{BL}$, that accounts for the calculated worst case transient conditions. The W(Z) and $W(Z)_{BL}$, factors described above for normal operation are specified in the COLR per Specification 6.9.1.6. Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion. Evaluation of the steady state $F_Q(Z)$ limit is performed in Specification 4.2.2.1.2.b and 4.2.2.1.4.b while evaluation nonequilibrium limits are performed in Specification 4.2.2.1.2.c and 4.2.2.1.4.c.~~

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors for RCS total flow rate and for $F^N_{\Delta H}$ have been taken into account in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. To perform the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated in accordance with the Surveillance Frequency Control Program. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Any fouling which might bias the RCS flow rate measurement can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

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$F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be limited by the following relationships:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP}}{p} K(Z) \; for \; P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \; for \; P \leq 0.5$$

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) provided in the CORE OPERATING LIMITS REPORT (COLR).

Where: $P = \frac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$, and

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

Evaluation of the steady state $F_Q(Z)$ limit is performed in Specification 4.2.2.1.2.b.

To account for possible variations in the value of $F_Q(Z)$ that are present during non-equilibrium situations, the steady state limit of $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions. The elevation dependent factors for normal operation are specified in the COLR per Specification 6.9.1.6. Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

$F_Q^M(Z)$ shall be evaluated to determine if the non-equilibrium limits described by the following relationships are satisfied:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} * K(Z)}{P * N(Z)} \; for \; P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} * K(Z)}{N(Z) * 0.5} \; for \; P \leq 0.5$$

Where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty.

F_Q^{RTP} is the F_Q limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR,

N(Z) is the cycle-dependent function that accounts for power distribution transients encountered during normal operation. N(Z) is specified in the COLR; and

P is the fraction of RATED THERMAL POWER defined as

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$$P = \frac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$$

Evaluation of the non-equilibrium $F_Q(Z)$ limit is performed in Specification 4.2.2.1.2.c.

To account for possible increases in the value of $F_Q(Z)$ between required 31 EFPD surveillances, the $F_Q^M(Z)$ is adjusted by an appropriate factor, typically 2%, that bounds the maximum expected increase in $F_Q(Z)$ over the interval. The appropriate factor, which may be defined as a function of burnup, is specified in the COLR per Specification 6.9.1.6. Specification 4.2.2.1.2.e allows for two options to conservatively account for compliance with the non-equilibrium $F_Q(Z)$ between the required 31 EFPD surveillance interval for $F_Q(Z)$:

1) Increase F_Q^M (Z) by the appropriate factor specified in the COLR and verify that F_Q^M (Z) satisfies Specification 4.2.2.1.2.c, or

2) Verify that F_Q^M (Z) satisfies Specification 4.2.2.1.2.c and perform the subsequent $F_Q(Z)$ surveillance at least once within 7 EFPD. The 7 EFPD surveillance interval may be discontinued when Option 1 above satisfies Specification 4.2.2.1.2.c.

Where it is necessary to calculate the percent that $F_Q(Z)$ exceeds the non-equilibrium limits, it shall be calculated as the maximum percent over the core height (Z) for the appropriate core planes, that $F_Q(Z)$ exceeds its limit by the following expression:

$$\left[\left[\frac{F_Q{}^M(Z) * N(Z)}{\frac{F_Q{}^{RTP}}{P} * K(Z)}\right] - 1\right] * 100\ for\ P > 0.5$$

$$\left[\left[\frac{F_Q{}^M(Z) * N(Z)}{\frac{F_Q{}^{RTP}}{0.5} * K(Z)}\right] - 1\right] * 100\ for\ P \leq 0.5$$

The core plane regions applicable to an $F_Q(Z)$ evaluation exclude the following measured in percent of core height:

a. Lower core region, from 0% to 8% inclusive,
b. Upper core region, from 92% to 100% inclusive,

The THERMAL POWER specified in the ACTION 3.2.2.1.b.(1) is normally the RATED THERMAL POWER. For example, if $F_Q^M(Z)$ exceeds its limit, then THERMAL POWER must be reduced to less than or equal to the percentage of RATED THERMAL POWER specified in the COLR.

ATTACHMENT 4

APPLICATION OF DOMINION NUCLEAR CORE DESIGN AND SAFETY ANALYSIS METHODS

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

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1 INTRODUCTION

Millstone Power Station Unit 3 (MPS3) became part of the Dominion nuclear fleet following Dominion's acquisition of Millstone Power Station in 2001. In addition to MPS3, the Dominion nuclear fleet presently includes Millstone Power Station Unit 2 (MPS2), North Anna Power Station (NAPS), Surry Power Station (SPS) and Kewaunee Power Station (KPS). Dominion nuclear core design and safety analysis methods were developed for application to the original Dominion nuclear power stations (SPS and NAPS) in the 1980's and to KPS in 2007. Over the years, these analysis methods have been successfully applied in numerous analytical, operational, and regulatory support activities.

This attachment documents justification for application of Dominion nuclear core design and safety analysis methods to MPS3. This attachment:

a) Describes Dominion nuclear core design and safety analysis methods, and
b) Documents assessments of the applicability of the Dominion nuclear core design and safety analysis methods to MPS3.

Section 2.0 identifies the analysis methods that are in the scope of application considered herein. The following methods are outside the scope of this review:

a) Containment response and containment integrity analysis methods
b) Radiological analysis methods
c) Fuel rod design and analysis methods (Note: transient fuel rod thermal response for specific transient events is in scope. The transient fuel rod thermal response is to be calculated using the approved RETRAN hot-spot model as described in Reference 15. With this exception, the responsibility for fuel rod design calculations is to reside with the fuel vendor using the approved methods described in the Core Operating Limits Report.)
d) Small break and large break loss of coolant accident (LOCA) analysis methods
e) Control Rod Ejection analysis methods

The reload design and safety analysis process performed by the current MPS3 fuel supplier (Westinghouse) is essentially the same process as the Dominion process, but it is performed with approved Westinghouse design and analysis methods. The current MPS3 Core Operating Limits Report (COLR) references the approved Westinghouse design and analysis methods for MPS3. These Westinghouse design and analysis methods will remain applicable to MPS3, and Dominion intends to retain the Westinghouse methods in the COLR after approval of the enclosed LAR to facilitate orderly transition to Dominion analyses.

Section 3.0 describes the various in-scope design and analysis methodologies, and documents assessments of the applicability of those methodologies to MPS3. Section 4.0 presents the conclusions derived from the methods applicability assessments.

As described herein, Dominion nuclear core design and safety analysis methods have been determined to be applicable to MPS3, and can be employed in design and licensing analyses for MPS3. The applicability of certain methods (e.g., VEP-FRD-41-P-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code"; and DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management

System Reactor Physics Methods for Application to North Anna and Surry Power Stations") required validation analyses. The results of the validation analysis for VEP-FRD-41-P-A are contained in Attachment 5 and for DOM-NAF-1-P-A are contained in Section 3.3 of this attachment. In addition, application of VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," involves a plant-specific and fuel-specific application analysis to define appropriate Departure from Nucleate Boiling Ratio (DNBR) Statistical Design Limits (SDLs). The Statistical DNBR analysis report is provided in a separate attachment. The accompanying LAR requests addition of Dominion Topical Reports as reference methodologies in the MPS3 Core Operating Limits Report (COLR) and in Technical Specification (TS) 6.9.1.6.b. Other conforming Technical Specification changes are incorporated into the LAR to reflect use of Dominion methods.

This attachment documents the application of the identified Dominion nuclear core design and safety analysis methods to MPS3. Applicability to MPS3 has been demonstrated by addressing specific aspects of the methods as documented in the individual topical reports which are noted in Section 2.0.

2 DOMINION NUCLEAR CORE DESIGN AND SAFETY ANALYSIS METHODOLOGIES

2.1 DOMINION METHODS TO BE APPLIED TO MPS3

Dominion currently applies its nuclear core design and safety analysis methods to its nuclear power stations, while the fuel vendor is responsible for fuel design analyses and reload fuel performance assessments. Dominion has performed reload design and safety analyses for approximately 90 reload cores at Surry, North Anna and Kewaunee using both vendor and Dominion-developed tools. Dominion will apply the Dominion nuclear core design and safety analysis methods to MPS3 in the same manner it applies these methods to the other plants in the fleet. The MPS3 fuel vendor retains responsibility for licensing the fuel design, for performing fuel rod design analysis, and for reload fuel performance assessment. For MPS3, the fuel vendor also performs certain specific safety analyses, e.g. small break and large break LOCA analyses.

Dominion has established a process for control and maintenance of its NRC-approved nuclear core design and safety analysis methodologies. Section 2.3 of Topical Report VEP-FRD-42-A, Rev. 2, "Reload Nuclear Design Methodology," (Reference 2) refers to this process. This process was further defined in responses to Requests for Additional Information (RAIs) on Dominion's Reload Nuclear Design Methodology (Reference 3). The NRC reviewed the Dominion analysis methods control and maintenance process and found it acceptable, as discussed in their Safety Evaluation Report (SER) for Dominion's Reload Nuclear Design Methodology (Reference 4). The Dominion analysis methods applied to MPS3 are to be controlled and maintained using these approved processes.

The Dominion nuclear core design methods within the scope of this review are:

a) VEP-FRD-42-A, "Reload Nuclear Design Methodology" (Reference 2)
b) VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications" (Reference 5)
c) DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations" (Reference 6)

The Dominion safety analysis methods within the scope of this review are:

d) VEP-FRD-41-P-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code" (Reference 1)
e) VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 7)
f) DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code" (Reference 8)

Each of the above methods is assessed for applicability to MPS3 in Section 3.0 using the Applicability Assessment Methodology described in Section 2.2.

2.2 APPLICABILITY ASSESSMENT METHODOLOGY

Dominion analysis methods are to be applied to MPS3 in a manner consistent with the conditions and limitations described in the Dominion topical reports and in applicable NRC Safety Evaluation Reports (SERs). Any differences from the methods described in the Dominion Topical Reports that are required for application of the methods to MPS3 are identified and addressed through the Applicability Assessment Methodology described herein.

The following systematic evaluation process is applied herein to assess the application of candidate methodologies to MPS3:

a) Each method is described, including its purpose, key features, and dependencies. Descriptions include:

- Key phenomena/conditions predicted by the method
- General calculation approach or assumptions
- Types of reactor conditions for which the method is used

b) Conditions and limitations associated with each method are identified, including:

- Regulatory limitations in NRC Safety Evaluation Reports (SER)
- Physical limitations (e.g., plant systems, plant features & conditions)
- Limitations in Dominion Topical Reports (e.g., specific modeling approaches or inherent assumptions)

c) Each method is assessed with respect to the identified Conditions and Limitations. The assessment effort ranges from written evaluations, to validation and benchmark analyses and detailed comparisons of results.

d) The results of the applicability assessment are documented for each method.

3 APPLICABILITY ASSESSMENTS

3.1 APPLICABILITY ASSESSMENT OF RELOAD NUCLEAR DESIGN METHODS – VEP-FRD-42-A, "RELOAD NUCLEAR DESIGN METHODOLOGY"

3.1.1 Description of Methodology

The Dominion reload nuclear design methodology includes calculational and process elements that are employed in the design and evaluation of reload nuclear cores. The major activities of the methodology are: 1) determination and fulfillment of cycle energy requirements; 2) determination of a core loading pattern; and 3) a reload safety evaluation that confirms acceptable behavior for the reload core under predicted design basis accident conditions.

The Dominion reload nuclear design methods, as documented in VEP-FRD-42-A (Reference 2), consist of the following elements:

a) Analytical Models (e.g., Studsvik Core Management System (CMS) Models, VEPCO RETRAN Models, Core Thermal-Hydraulics VIPRE-D Models)

b) Analytical Methods (e.g., Core Depletions, Core Reactivity Parameters and Coefficients, Core Reactivity Control, Safety Analysis, Statistical DNB)

c) Reload Design Process (e.g., Core Loading Pattern Design & Optimization, Key Parameter Treatment in Nuclear Design Analyses)

d) Reload Safety Evaluation Process

e) Nuclear Design Report, Operator Curves & Core Follow Data

The Dominion methodology for designing a reload core is an iterative process. The process involves determining a core loading pattern which provides the required total cycle energy and then demonstrating through analysis or evaluation that the plant will continue to meet all applicable safety criteria after considering the changes associated with the reload core.

Reload safety evaluation and analysis criteria are established using a bounding analysis concept. This approach employs a list of key analysis parameters with the limiting direction of each parameter identified. This allows reload core characteristics to be compared with the parameter values assumed in the reference analyses for various transients and accidents. For a proposed core reload design, if all key analysis parameters are conservatively bounded, then the reference safety analysis applies, and no further analysis is necessary. If one or more key analysis parameters are not bounded, then the affected events may be handled in a number of ways. If the parameter only slightly exceeds the limit, or the affected transients are relatively insensitive to the parameter, a quantitative evaluation is performed to conservatively estimate the effect, but the current limit is not changed. If the deviation is large and/or is expected to have a more significant or not easily quantifiable effect on the event, the event is reanalyzed. If event reanalysis produces unsatisfactory results, then the loading pattern may be changed or changes may be made in operational requirements (e.g., Technical Specifications or Core Operating Limits Report (COLR) changes) to ensure that plant operation will satisfy the applicable safety analysis criteria for the proposed loading pattern.

Topical Report WCAP-9272, "Westinghouse Reload Safety Evaluation," (Reference 9) describes the Westinghouse methodology used for reload safety evaluation. WCAP-9272 forms the basis for Dominion's reload methodology as described in Topical Report VEP-FRD-42-A. The Westinghouse methodology defines the specific key parameters for use in accident analyses and provides limiting directions for consideration in reload evaluations.

The reload core design is evaluated by comparing the reload core parameters against the assumptions in the current safety analyses. Safety analysis (accident analysis) is the study of nuclear reactor behavior under accident conditions. The accident analyses consider all relevant aspects of the plant and core including the operating procedures and limits on controllable plant parameters and the engineered safety, shutdown, and containment systems.

There are two stages in the typical safety analysis process. First, steady state nuclear calculations are performed for the core conditions assumed in the accident analysis. The nuclear parameters derived from these calculations are called the core physics related key analysis parameters and serve as input to the second stage. The second stage is the actual dynamic accident analysis, which yields the accident results that are applicable for these key analysis parameter values. The accident analyses are transient calculations that usually model the core nuclear kinetics and those parts of the plant systems that have a significant impact on the events under consideration.

The Millstone Unit 3 Final Safety Analysis Report (FSAR) documents acceptable plant safety via detailed results of accident analyses performed with the bounding values of key analysis parameters. Plant safety is demonstrated if accident analysis results meet the applicable acceptance criteria. The reload core design is evaluated by comparing the core physics related key analysis parameters against the assumptions in the current safety analyses. The reload evaluation process is complete if the acceptance criteria delineated in the FSAR are satisfied with the reload core implemented. If an accident reanalysis is necessary, more detailed analysis methods and/or Technical Specifications changes may be required to meet the acceptance criteria. Such changes are to be processed in accordance with the applicable regulatory processes.

In summary, the overall reload evaluation process includes the following steps:

a) Determine bounding key analysis parameters, which constitute the current limits for reload cores.
b) Perform (or confirm) accident analysis using the bounding key analysis parameters and conservative assumptions.
c) Establish a proposed core loading pattern that provides the required total cycle energy.
d) Determine, for the proposed core loading pattern, the value for each key analysis parameter.
e) Compare key reload analysis parameters to current limits.
f) Evaluate whether an accident reanalysis is needed based on the effect the reload key analysis parameters may have on the accident.
g) Perform reanalysis of specific affected accidents, change operating limits, or revise the loading pattern, as necessary.

Key attributes of the reload nuclear design methodology include the following elements:

a) An analysis framework in which safety analyses establish acceptable limit values for reload core key analysis parameters, while nuclear and fuel design codes confirm each core's margin to these limits
b) The use of bounding key parameter values in reference safety analyses
c) Recurrent validation of nuclear design analytical predictions through comparison with reload core measurement data
d) Representation of key fuel features via detailed inputs in core design and safety analysis models
e) Fuel modeling using approved critical heat flux correlations demonstrated to be applicable and within the range of qualification and identified in the plant COLR section of the TS

3.1.2 Conditions and Limitations

There are specific and inherent conditions and limitations that are associated with application of the methods documented in VEP-FRD-42-A to MPS3:

a) Regulatory limitations in NRC Safety Evaluation Reports (SER)
   i. Inherent limitation for use at North Anna and Surry Power Stations
   ii. Prior to its use for fuel types other than Westinghouse and Framatome ANP Advanced Mark-BW fuel, confirm that the impact of the fuel design and its specific features can be accurately modeled with the VEPCO nuclear design and safety analysis codes and methods. Should the changes necessary to accommodate another fuel product require changes to the reload methodology of Topical Report VEP-FRD-42-A, these proposed changes are required to be submitted for prior NRC review and approval.

b) Physical limitations (e.g., plant systems, plant features & conditions)
   None identified (The methods of VEP-FRD-42-A are not dependent on such physical limitations)

c) Limitations in Dominion Topical Reports (e.g., specific modeling approaches or inherent assumptions)
   None identified

3.1.3 Assessment

The Dominion reload nuclear design methods and the current MPS3 reload nuclear design methods are both based on the Westinghouse Topical Report WCAP-9272. Thus, the Dominion and MPS3 reload nuclear design methods are similar since they have a common basis in the Westinghouse reload safety evaluation methods.

MPS3 and the other Dominion Westinghouse nuclear units use Westinghouse designs for nuclear steam supply system (NSSS) and reactor protection system (RPS). MPS3 and the other Dominion Westinghouse units have many design and operating similarities. The specific differences in NSSS, RPS and fuel features for MPS3 are all capable of being reflected via modeling inputs in the analytical methods of VEP-FRD-42-A. These differences do not impact the execution of the key VEP-FRD-42-A methodology elements for the design and evaluation of reload cores.

Verification of the boration capabilities for Millstone Unit 3 will be addressed during each reload

cycle to confirm the boron requirements are within TS, TRM, and FSAR limits. The method of verifying the boration requirements on a cycle-specific basis uses fundamental boration equations. The Westinghouse BORDER code (WCAP-14441) automates the verification of the boration requirements that historically was performed manually using these fundamental boration equations. As part of the implementation of Dominion methods, Dominion will verify the boration requirements for Millstone Unit 3 on a reload basis using the same constituent equations employed in WCAP-14441. The Dominion reload process is designed to address boration capability for routine plant changes, such as core reloads, and infrequent plant changes, such as plant uprating that result in a change to core operating conditions and initial core reactivity.

The reload safety evaluation and analysis process for Dominion and MPS3 uses a bounding analysis concept. The method that is used for Dominion and MPS3 employs a list of key analysis parameters and limiting directions of the key analysis parameters for various transients and accidents.

The key analysis parameters and the limiting directions of those key parameters for the various transients and accidents were evaluated. The key analysis parameters and their limiting direction for MPS3 are the same as the key analysis parameters and limiting direction for the other Dominion Westinghouse units. The design basis accidents and transients in the safety analyses were also evaluated. The design basis transients and accidents for MPS3 are similar to the design basis transients and accidents for the other Dominion Westinghouse units. Based on the key analysis parameters assessment, and the design basis transients and accidents that are evaluated in the reload safety evaluation process, the Dominion reload safety evaluation and analysis methods have been determined to be applicable to MPS3.

3.1.4 Summary

Dominion reload nuclear design methods documented in VEP-FRD-42-A are concluded to be applicable to MPS3 and can be applied to MPS3 licensing analysis for reload core design and reload safety evaluation.

3.2 APPLICABILITY ASSESSMENT OF RELAXED POWER DISTRIBUTION CONTROL METHODS – VEP-NE-1-A, "RELAXED POWER DISTRIBUTION CONTROL METHODOLOGY AND ASSOCIATED FQ SURVEILLANCE TECHNICAL SPECIFICATIONS"

3.2.1 Description of Methodology

The relaxed power distribution control (RPDC) method is the Dominion method for axial power distribution control. RPDC involves a variable axial flux difference (delta-I) band power distribution control strategy that uses a widened full power delta-I band, and provides for an increasing delta-I band with decreasing power. The widened delta-I band is based on maintaining an approximately constant analysis margin to the design bases limits at all power levels.

RPDC also involves the formulation of Technical Specification surveillance and COLR limits for Total Peaking Factor (FQ). The FQ surveillance uses the measured core axial position-dependent FQ (FQ(Z)) augmented by a non-equilibrium operation multiplier (N(Z)) in order to verify compliance with the peaking factor limits. This FQ surveillance is a required element of the RPDC method.

The objective of the RPDC analysis is to determine acceptable delta-I bands that maintain margin to all the applicable design bases criteria and at the same time provide enhanced delta-I operating margin. Because the RPDC delta-I band is an analysis output quantity rather than a fixed input, power shapes that adequately bound the potential delta-I range must be generated. These power shapes must include the effect of combinations of the key parameters such as burnup, control rod position, xenon distribution, and power level. Dominion has developed the methodology to generate the large number of power shapes required for RPDC analyses.

After the power shapes have been created, proposed delta-I bands are chosen such that all shapes within the delta-I bands satisfy the COLR FQ(Z) limit. For the normal operation analysis, power levels spanning the 50% to 100% range are investigated to establish the RPDC delta-I limits. Further verification of the proposed delta-I bands is performed via two different limiting shape evaluations, one based on Loss of Coolant Accident (LOCA) FQ considerations and the other based on a Loss of Flow Accident (LOFA) thermal/hydraulic evaluation. Delta-I bands can be narrowed to satisfy the requirements of these evaluations, if necessary.

Condition II or Abnormal Operation events, which may be the result of system malfunctions or operator errors, are also analyzed for RPDC. This RPDC analysis examines the more limiting of these Condition II events and verifies on a cycle-to-cycle basis that the Over-Power Delta-T (OPΔT) and the Over-Temperature Delta-T (OTΔT) setpoints are conservative. The OPΔT and OTΔT setpoints were designed primarily to provide transient and steady state protection against fuel centerline melt and DNB, respectively.

The RPDC methodology takes advantage of the large amount of margin to the design bases limits available at reduced power levels and provides delta-I limits at all power levels that are less restrictive than under Constant Axial Offset Control (CAOC) operation. The RPDC methodology may be summarized as follows:

a) A full range of normal-operation power shapes is obtained by combining the key parameters upon which each shape is dependent: xenon distribution, core burnup, boron concentration, core power level and control rod position. Reasonable increments spanning the entire range of values are considered for each of these parameters. A xenon "free oscillation" method is used to create the many and varied axial xenon distributions required for this analysis.

b) Proposed delta-I bands are selected such that the COLR FQ(Z) limit is met for all power shapes within the proposed bands. These power shapes are then analyzed to determine which shapes result in an approach to the LOFA limits.

c) Final normal operation delta-I bands are established such that the LOCA and the LOFA limits are satisfied.

d) Conditions that yield shapes within the final normal operation delta-I bands are used as initial conditions for the bounding Condition II accident simulations.

e) The resultant transient shapes are analyzed and the OPΔT and OTΔT trip function/setpoints are verified to ensure that margin to fuel design limits is maintained.

f) N(Z) functions (non-equilibrium power distribution multiplier) are formulated based on the maximum composite calculated Condition I FQ(Z) x P (i.e., local FQ times total core thermal power) to support the implementation of FQ Technical Specifications surveillance.

All neutronic calculations are performed with NRC-approved codes and methods. All DNBR calculations are performed using NRC-approved thermal-hydraulic code(s), correlation(s), and methods.

3.2.2 Conditions and Limitations

There are no specific conditions or limitations specified in the RPDC SER (Reference 5) for use of this methodology. Commonality amongst the Dominion, Westinghouse, CE, and Exxon versions of this methodology is noted throughout the RPDC SER. The following comments from the RPDC SER are relevant to the assessment of use of the RPDC method for MPS3:

a) Approved methods were used for analyses supporting RPDC.

b) Justification was provided for the uncertainties assigned.

c) Impact of cycle specific variations on the delta-I power domain, OPΔT and OTΔT trip setpoints, and other safety analyses will be evaluated on a reload basis.

d) RPDC is an acceptable methodology for application to reload cores that are similar to those of the Surry (SPS) and North Anna (NAPS) reactors.

The following methodology items must be evaluated to determine if the values stated are applicable to MPS3 cores:

e) The appropriate plant-specific value for the maximum calculated temperature reduction during an EOC cooldown transient is to be determined for MPS3. A cooldown of 20°F was shown to be bounding for North Anna.

f) A dilution time of 15 minutes is assumed after the control rods pass the insertion limit for the boron dilution event.

3.2.3 Assessment

Conditions (a)-(c) cited in Section 3.2.2 are met for use of RPDC at MPS3 because the same methods, uncertainty parameters, and analyses will be employed as are currently employed for North Anna RPDC analyses.

The specific MPS3 uncertainty for calculated FQ (FQU, condition (b)) has been determined as part of the MPS3 validation analysis for the CMS methods (see applicability assessment for DOM-NAF-1-P-A methodology in Section 3.3).

Condition (d) is satisfied due to the many similarities between MPS3, SPS, and NAPS. All three stations use Westinghouse designs for the nuclear steam supply system (NSSS) and reactor protection system (RPS). In addition, MPS3 reload cores and those at NAPS are essentially identical since both use the RFA-2 fuel design (see comparison in Table 3.3.1).

The applicability of the RPDC methodology to MPS3 is also supported by the fact that the Westinghouse Relaxed Axial Offset Control (RAOC) method is currently used by MPS3. There are a number of similarities between Dominion's RPDC method and the Westinghouse RAOC power distribution control method (Reference 10). Table 3.2.1 presents a detailed side by side comparison of the Westinghouse RAOC and Dominion RPDC methodologies:

Table 3.2.1: Comparison of Key Elements of Dominion RPDC and Westinghouse RAOC Methodology

Category	Element	Westinghouse RAOC	Dominion RPDC	Comparison
Technical Specification / COLR Limits				
	Operating Limits	Axial Flux Difference (AFD) limits versus reactor power	Axial Flux Difference (AFD) limits versus reactor power	Same
FQ Surveillance				
	Non-equilibrium conditions	RAOC applies a cycle specific W(z) factor to the FQ limit to account for non-equilibrium operation.	RPDC applies a cycle specific N(z) factor to the FQ limit to account for non-equilibrium operation.	Same
Condition I Analysis				
	Xenon Distributions	RAOC methodology populates a xenon shape library using a xenon reconstruction model. The reconstruction model is dependent on several parameters whose ranges are determined by xenon transient analysis. Parameters for a given xenon shape are retained only if Delta-I control can be maintained for those shapes within a tentative limit.	The RPDC xenon shape library is built by reducing Doppler feedback in the base neutronics model and allowing a divergent xenon oscillation to occur. Actual xenon distributions are sampled and saved from this transient. No consideration is given during the transient calculation to whether or not the shapes are obtainable during normal operation	Both methodologies generate axial xenon distributions that cover essentially the same delta-I space.

Table 3.2.1: Comparison of Key Elements of Dominion RPDC and Westinghouse RAOC Methodology

Category	Element	Westinghouse RAOC	Dominion RPDC	Comparison
	Power Levels	Minimum of three power levels, 100%, 50%, and an intermediate power are required.	A range of power levels between 50% and 100% power with small enough increments to ensure an adequate number of power distributions are being analyzed (typically 10% power intervals).	Essentially the same
	Control Rod Positions	Range of control rod positions from ARO to power dependent Rod Insertion Limits (RIL)s.	Range of control rod positions from ARO to power dependent Rod Insertion Limits (RIL)s.	Same
	Burnups	BOL, MOL, EOL	BOL, MOL, EOL	Same
FQ Analysis				
	Loss of Coolant Accident (LOCA)	Each power shape generated for Condition I is analyzed to determine if LOCA constraints are met or exceeded. For each power level the results of this analysis will indicate a tentative range of delta-I in which there are no violations of the LOCA limits.	The FQ*Power for each shape is compared to the LOCA FQ*Power*K(z) limit at each power level to determine which axial shapes approach the LOCA limit, thereby establishing a preliminary allowable delta-I versus power band.	Essentially the same
	Loss of Flow Accident (LOFA)	Normal operation power distributions are evaluated relative to the assumed limiting normal operation power distribution used in the accident analysis	The entire set of axial power distributions from the normal operation analysis are evaluated against the design axial power distribution for the LOFA analysis with the applicable thermal-hydraulic code(s) and correlation(s).	Essentially the same

Table 3.2.1: Comparison of Key Elements of Dominion RPDC and Westinghouse RAOC Methodology

Category	Element	Westinghouse RAOC	Dominion RPDC	Comparison
Condition II Analysis				
	Analyzed Accidents	Cooldown Accident, Control Rod Withdrawal, Boration / Dilution	Cooldown Accident, Control Rod Withdrawal, Boration / Dilution	Same
	Shape Selection	Initial statepoints for Condition II analysis are limited to the Condition I axial power distributions that fit within tentative delta-I bands.	Initial statepoints for Condition II analysis are limited to the Condition I axial power distributions that fit within tentative delta-I bands.	Same

With regard to the applicability of specific values in the RPDC methodology to MPS3 (conditions (e)-(f)):

The cooldown transient assumption of 30°F (condition (e)) currently assumed for the Westinghouse method at MPS3 shall be used unless a MPS3-specific analysis can demonstrate that a plant trip will occur prior to reaching 30°F.

The at-power 15 minute dilution time prior to operator action (condition (f)) value is the same as that stated in MPS3 FSAR Section 15.4.6.1 (CVCS malfunction event description, for Mode 1), and the same as that currently used for MPS3.

3.2.4 Summary

The Dominion RPDC method is determined to be applicable to MPS3 and can be applied to MPS3 licensing analysis for nuclear core design and reload safety evaluation. The enclosed License Amendment Request (LAR) to add VEP-NE-1-A to Section 6.9.1.6.b of the MPS3 Technical Specifications includes Technical Specification changes necessary for conformance with the RPDC methodology.

3.3 APPLICABILITY ASSESSMENT OF CORE MANAGEMENT SYSTEM METHODS – DOM-NAF-1-P-A, "QUALIFICATION OF THE STUDSVIK CORE MANAGEMENT SYSTEM REACTOR PHYSICS METHODS FOR APPLICATION TO NORTH ANNA AND SURRY POWER STATIONS"

3.3.1 Description of Methodology

The Dominion reactor physics methods include the Studsvik Core Management System (CMS) core modeling code package. The primary computer codes in the CMS package are CASMO-4 and SIMULATE-3. The CASMO-4 computer code is the fuel assembly lattice code. CASMO-4 is a multi-group, two-dimensional transport theory code used for depletion and branch calculations for a single fuel assembly. The SIMULATE-3 code is a two-group, 3-dimensional nodal code based on the modified coarse mesh (nodal) diffusion theory calculation technique, coupled with thermal hydraulic and Doppler feedback. The general CMS calculation approach is to model the fuel assembly using the CASMO two-dimensional lattice physics code, and then to construct the three-dimensional SIMULATE reactor core model using lattice physics cross section data.

CMS reactor physics codes are used to model the core physics characteristics of the reload core including depletion/isotopic effects, reactivity, reactivity coefficients, power distribution, and shutdown margin. Dominion uses CMS reactor physics models in licensing applications, including calculations for core reload design, core operation, and key core parameters for reload safety analyses. CMS models are applied in the analyses for relaxed power distribution control, for startup physics testing (including control rod worth determination using the boron dilution and rod swap measurement techniques), and to provide physics constants for measurement of core power distributions.

CMS models are used to analyze the reactor core in all modes of operation including refueling shutdown, cold shutdown, 0% to 100% reactor power, and conditions associated with design basis transients. CMS models are applied over the entire fuel cycle from beginning to end of cycle.

3.3.2 Conditions and Limitations

a) The DOM-NAF-1-P-A title and several statements in its SER refer to use of CMS for North Anna (NAPS) and Surry (SPS) Power Stations.
b) Benchmarking data was provided for 15x15 (SPS) and 17x17 (NAPS) fuel designs.
c) In Section 5.0, "Conditions and Limitations," the SER lists two conditions that would require further validation and NRC approval:
    i. Use of mixed oxide fuel
    ii. Introduction of significantly different or new fuel designs

3.3.3 Assessment

Referring to the Conditions and Limitations listed in Section 3.3.2:

Condition (a) is not a technical limitation. Condition (b) is met since a MPS3 17x17 fuel design benchmark has been performed for MPS3 (see below). For condition (c), part i, MPS3 does not use mixed oxide fuel; condition (c), part ii is met since the MPS3 fuel is the same as NAPS, and that

"significantly different" was further clarified by the NRC review staff to mean a change to the actual fuel lattice structure (for example using 20 x 20 lattice instead of 17x17 or modeling BWR fuel would be examples of changes that would require further approval).

The MPS3 fuel assembly lattice is a 17x17 fuel lattice and the current MPS3 fuel design is the Westinghouse RFA-2 fuel design. The MPS3 fuel lattice and fuel design are not significantly different from the SPS (15x15) and NAPS (17x17) fuel designs, in fact both MPS3 and NAPS currently use RFA-2 fuel. Table 3.3.1 demonstrates the similarities of the current MPS3 and NAPS fuel designs.

Table 3.3.1: Fuel Assembly and Component Design Parameters

Component	MPS3 RFA-2	NAPS RFA-2
Fuel Assembly		
Array	17 x 17	17 x 17
Envelope dimension	8.426 in.	8.426 in.
Pitch (Rod)	0.496 in.	0.496 in.
No. guide tubes	24	24
No. instrument tubes	1	1
No. spacer grids in active fuel	10	10
Fuel Rods		
Fuel Pellet Diameter	0.3225 in.	0.3225 in.
Fuel Pellet Material	Sintered UO_2	Sintered UO_2
Fuel/Clad Diametric Gap	0.0065 in.	0.0065 in.
Fuel Cladding		
O.D.	0.374 in.	0.374 in.
I.D.	0.329 in.	0.329 in.
Material	Optimized ZIRLO	Optimized ZIRLO
Spacers (Top & Bottom/Mid)		
Material	Inconel/ZIRLO	Inconel/ZIRLO
Guide Tube		
O.D. (above dashpot)	0.482 in.	0.482 in.
Material	ZIRLO	ZIRLO
Axial Variation		
Axial Blankets	2.6 w/o annular pellets	Full enrichment annular pellets

Using the methods and processes delineated in DOM-NAF-1-P-A, a nine cycle benchmark of MPS3 (cycles 8 through 16) has been performed. Nuclear Reliability Factors (NRF) for the key reload key physics parameters have been determined and the accuracy of the CMS models has been demonstrated through comparisons with reactor measurements and through comparisons with higher order Monte Carlo neutron transport calculations. This benchmark is consistent with the assessment performed for the Dominion Surry and North Anna units as described in DOM-NAF-1-P-A.

The capability of the CMS models to support the MPS3 Startup Physics Test Program has also been demonstrated through the reactor measurement comparisons.

Table 3.3.2 presents the MPS3 Nuclear Reliability Factors determined from the above benchmark along with the North Anna and Surry NRFs for comparison. The only parameter for MPS3 which has a different NRF is the upper Differential Control Rod Bank Worth, which is larger for MPS3 than for North Anna and Surry.

Table 3.3.2: Nuclear Reliability Factor Comparison

Parameter	Millstone-3 NRF		North Anna / Surry NRF	
	Upper	Lower	Upper	Lower
Integral Control Rod Bank Worth (Individual Banks)	1.10	0.90	1.10	0.90
Integral Control Rod Bank Worth (Total of all banks)	1.10	0.90	1.10	0.90
Differential Control Rod Bank Worth	1.20	0.80	1.15	0.80
Critical Boron Concentration	+50 ppm	-50 ppm	+50 ppm	-50 ppm
Differential Boron Worth	1.05	0.95	1.05	0.95
Isothermal and Moderator Temperature Coefficient	+2 pcm/F	-2 pcm/F	+2 pcm/F	-2 pcm/F
Doppler Temperature Coefficient	1.10	0.90	1.10	0.90
Doppler Power Coefficient	1.10	0.90	1.10	0.90
Effective Delayed Neutron Fraction	1.05	0.95	1.05	0.95
Prompt Neutron Lifetime	1.05	0.95	1.05	0.95
FΔH	1.04	N/A	1.04	N/A
FQ	1.05	N/A	1.05	N/A

3.3.4 Summary

The Dominion CMS Core Physics methodology is determined to be applicable to MPS3 and can be applied to MPS3 licensing analysis for nuclear core design and reload safety evaluation.

3.4 APPLICABILITY ASSESSMENT OF RETRAN METHODS – VEP-FRD-41-P-A, "VEPCO REACTOR SYSTEM TRANSIENT ANALYSES USING THE RETRAN COMPUTER CODE"

3.4.1 Description of Methodology

3.4.1.1 Background

Dominion has the capability to perform system transient analyses. This capability, coupled with core thermal/hydraulic analysis capability, encompasses the non-LOCA licensing analyses required for the Condition I, II, III, and IV transients and accidents addressed in the Final Safety Analysis Report (FSAR). In addition, the capability for performing best-estimate analyses for plant operational support applications has also been developed.

The purposes of having transient and accident safety analysis capability are to: 1) maintain in-house cognizance and expertise in the system transient analysis area; 2) support plant operation; and 3) provide a basis for the reload core safety analysis and licensing process. The principal analysis tool is the RETRAN computer code (Reference 1 and 12), which determines the time-dependent (transient) thermal-hydraulic response of a Nuclear Steam Supply System (NSSS). The RETRAN computer code calculates: 1) general system parameters as a function of time; and 2) boundary conditions for input into more detailed calculations of Departure from Nucleate Boiling (DNB) or other thermal and fuel performance margins. The theory and numerical algorithms, the programming details, and the user's input information for the RETRAN computer code have been documented by its developers, Computer Simulation and Analysis Incorporated (CSA) and the Electric Power Research Institute (EPRI), in Volumes I through III of Reference 1 and 12. Volume IV of Reference 1 and 12 provides the results of the extensive verification and qualification of the RETRAN code. The verification activity consisted of qualification of the code by comparison of code results with separate effects experiments, with systems effects tests, and with integrated system responses based on actual plant data or FSAR results.

In conjunction with both an analysis tool and system models, the development of a non-LOCA licensing analysis capability requires conservative analysis assumptions and input data. For performing licensing calculations at MPS3, the major Dominion analysis assumptions are consistent with those documented in the units' FSAR and the required equipment described in the units' Technical Specifications.[1] If a change in analysis assumptions is required by a plant modification, core reload, or a related change, the change will be evaluated via the 10 CFR 50.59 process. Depending on the results of that assessment, either the analysis is submitted to the NRC for approval or a normal update of the appropriate section of the FSAR is prepared.

[1] An example where these may differ is the current MPS3 FSAR rod withdrawal from subcritical analysis assumption of two reactor coolant pumps in operation versus the three required to be in operation by the Technical Specifications. In this instance, Dominion's methodology would assume three pumps in operation.

3.4.1.2 Licensing Applications

Dominion's system transient analysis capability is intended for both best-estimate (e.g. training simulator validation) and licensing applications (e.g. core reload analysis). Since core reloads are the most common and expected reason for accident reanalysis, Dominion's system transient methodology is discussed in that context.

Transient analyses form an integral part of evaluations performed to verify the acceptability of a reload core design from the standpoints of safety, economics, and operational flexibility. The reload process consists of design initialization, design of the core loading pattern, and detailed characterization of the core loading pattern by the nuclear designer. The latter process determines the values of core physics related key analysis parameters. These key parameters are provided to the safety analyst who uses them in conjunction with current plant operating configurations and limits to evaluate the impact of the core reload on plant safety.

In performing this evaluation, it is necessary to ensure that those key parameters that influence accident response are maintained within the bounds or "limits" established by the parameter values used in the reference analysis (i.e. the currently applicable licensing calculation). The reference analysis (and the associated parameter limits) may be updated from time to time in support of a core reload or to evaluate the impact of some other plant parameter change.

For cases where a parameter is outside of these previously defined limits, an evaluation of the impact of the change on the results for the appropriate transients must be made. This evaluation may be based on known sensitivities to changes in the various parameters in cases where a parameter change is small or the influence on the accident results is weak. For cases where larger parameter variations occur, or for parameters that have a strong influence on accident results, explicit reanalysis of the affected transients is required and performed. Past analytical experience has allowed the correlation of the various accidents with those parameters that have a significant impact on them.

If a reanalysis is performed, the results are compared to the appropriate analysis acceptance criteria. The reload evaluation process is complete if the acceptance criteria are met, and internal documentation of the reload evaluation is provided for the appropriate Dominion safety review. If the analysis acceptance criteria are not met, more detailed analyses and/or Technical Specifications changes may be required to meet the acceptance criteria. Analysis changes are evaluated in accordance with the requirements of 10 CFR 50.59.

3.4.1.3 System Model Application

The production of a conservative, reliable safety analysis of a given anticipated or postulated transient is accomplished by combining a system transient model with appropriate transient-specific input. A system transient model is designed to provide an accurate representation of the reactor plant and those associated systems and components that significantly affect the course of the transient. Transient-specific input ensures that the dynamic response of the system to the postulated abnormality is predicted in a conservative manner, and includes: a) initial conditions; b) core reactivity parameters such as Doppler and moderator temperature coefficients, and control rod insertion and reactivity

characteristics; and c) assumptions concerning overall systems performance. Important system performance assumptions include the availability of certain system components (such as pressurizer spray or relief valves) and control and protection system characteristics (setpoints, instrument errors, and delay times).

RETRAN affords the modeling flexibility to develop an infinite number of representations for a given nuclear plant. At Dominion, several standard plant models are assembled and maintained for performance of the entire spectrum of system transient analyses. RETRAN makes use of an input structure that allows modification of the base deck input for specific cases by use of override cards. Thus, specific transient cases may be executed without altering the base plant models.

The base models are designed to provide a basic system description comprised of those parameters that would not ordinarily change from cycle to cycle. Thus, such parameters as system volumes and flow areas, characteristics of various relief and safety valves, and primary coolant pump characteristics form part of the base models.

Dominion's RETRAN Topical Report VEP-FRD-41-P-A, (Reference 1) describes Dominion's history with the use and application of RETRAN, dating from the early days of code development. Provided is a detailed description of the three-loop base models that have been developed for Dominion's Surry and North Anna plants. In addition, Reference 1 provides a description of benchmarks to other industry codes and plant transient data used to validate Dominion's modeling selections. These benchmarks show that Dominion's RETRAN methodology and modeling selections provide conservative results for a diverse set of actual plant transients. Furthermore, Dominion's RETRAN methodology and models were also shown to be applicable to Westinghouse two-loop designs through benchmarking in Topical Report DOM-NAF-5-A (Reference 14).

3.4.1.4 RETRAN History at MPS3

RETRAN is the computer code currently used at MPS3 for FSAR system transient analyses. The NRC approved methodology is described in Reference 13. While, there are differences between Dominion's RETRAN methodology (Reference 1) and the methodology described in Reference 13, the use of RETRAN at MPS3 demonstrates the code's ability to analyze system transients for 4-loop Westinghouse NSSS plants.

3.4.1.5 Development of the MPS3 Model

The MPS3 RETRAN model was constructed based on the following governing principles:

a) The MPS3 RETRAN model is consistent with the approved methods used in VEP-FRD-41-P-A (Reference 1). There are a few minor differences:

    i. The MPS3 model explicitly models the safety injection accumulators.

ii. The MPS3 model has separate volumes for the steam generator inlet and outlet plenums, where the Reference 1 models lump these volumes with the hot leg and pump suction leg respectively.

b) The MPS3 RETRAN model complies with the restrictions and limitations of use presented in the RETRAN-3D Safety Evaluation Report (SER) for RETRAN-3D in RETRAN-02 mode or the RETRAN-02 SER (Reference 12).VEP-FRD-41-P-A (Reference 1) outlines compliance with both the RETRAN-02 SER and RETRAN-3D SER conditions and limitations.

c) The model is consistent with the input parameters used for the MPS3 FSAR Chapter 15 transient analysis.

d) The MPS3 RETRAN model is essentially consistent with the other Dominion RETRAN models (similar nodalization and node numbers). Adopting a consistent noding and numbering scheme facilitates use of the models by analysts who are familiar with the Surry and North Anna plant models. As noted in (a) above, there are a few minor differences.

3.4.2 Conditions and Limitations

Reference 1 provides a detailed discussion of the conformance of Dominion's Surry and North Anna RETRAN models to the restrictions, limitations and conditions of use imposed by NRC staff in the generic RETRAN code Safety Evaluation Reports (SERs) issued for the RETRAN code Topical Report. This includes the restrictions, limitations, and conditions imposed by the NRC for the use of RETRAN-02 and differences in the restrictions, limitations, and conditions described in the RETRAN-3D SER. This includes the limitations and conditions associated with the use of RETRAN-3D in RETRAN-02 mode. Based on the principles cited above for development of the MPS3 model, the Reference 1 assessment is applicable to the MPS3 model. Table 3.4.1 lists the MPS3-specific evaluations of the NRC RETRAN code restrictions and limitations for which further explanation was warranted for application of VEP-FRD-41-P-A to MPS3.

Table 3.4.1: MPS3-Specific Evaluation of the USNRC Generic RETRAN Code Restrictions & Limitations

RETRAN-02/RETRAN-3D SER Restrictions, Limitations, and Conditions	VEP-FRD-41-P-A Evaluation	MPS3 Disposition
a) Conservative usage of 1-D kinetics must be demonstrated	Rod ejection performed with point kinetics per Dominion Topical Report VEP-NFE-2-A.	Dominion does not propose to apply VEP-NFE-2 methods for the MPS3 rod ejection analysis at this time. Rod ejection analyses will continue to be done with approved Westinghouse methods.
e) Metal-water reaction will have to be justified for specific analyses	Model only used for rod ejection hot pin model. Justification/ benchmarking provided in Dominion Topical Report VEP-NFE-2-A.	Dominion will apply hot pin model with metal-water reaction to MPS3 analyses of rod withdrawal from subcritical and locked reactor coolant pump rotor events. Justification/benchmarking is in Dominion Topical Report VEP-NFE-2-A (Ref 15).

3.4.3 Assessment

RETRAN is approved for application to MPS3 and is used for the current MPS3 safety analyses of record (Reference 13).

MPS3 and other Dominion units (Surry and North Anna) use Westinghouse designs for nuclear steam supply system (NSSS) and reactor protection system (RPS). SPS, NAPS and MPS3 have many design and operating similarities.

The design basis transients and accidents for MPS3 are similar to the design basis transients and accidents for the other Dominion Westinghouse units.

The reactor thermal hydraulic conditions of the design basis transients and accidents are similar between MPS3 and the other Dominion units, and are within the qualification and capability of the RETRAN code, as demonstrated by the current MPS3 FSAR non-LOCA safety analysis methodology in Reference 13.

Validation of the Dominion MPS3 RETRAN model involves comparison of Dominion RETRAN calculations to the MPS3 analysis of record for selected transients using the following strategy:

a) Identify unique classes of events (RCS heatup, RCS cooldown/depressurization, reactivity excursion, loss of RCS flow, and loss of secondary heat sink).

b) Select transients that represent a range of transient responses generated by these events.

c) Perform demonstration analyses of selected events to validate the capability to model key phenomena.

d) Verify that applicability assessment criteria are met:

   i. Key phenomena are appropriately modeled and predicted

   ii. Predicted results are technically sound and are in reasonable agreement with the MPS3 FSAR analyses of record (or differences are understood and assessed as acceptable)

   iii. General trends in key parameters are consistent with FSAR analyses of record

3.4.4 Summary

Dominion's RETRAN methods (Reference 1) are determined to be applicable to MPS3 and can be applied to MPS3 licensing analysis for reload core design and safety analysis. The applicability of these methods is documented in a separate attachment which provides:

a) A base model noding diagram and region descriptions
b) Results of benchmarking comparisons to the analyses of record for selected transients

3.5 APPLICABILITY ASSESSMENT OF STATISTICAL DNBR EVALUATION METHODS – VEP-NE-2-A, "STATISTICAL DNBR EVALUATION METHODOLOGY"

3.5.1 Description of Methodology

Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," (Reference 7) describes Dominion's methodology for statistically treating several important uncertainties in Departure from Nucleate Boiling Ratio (DNBR) analysis. Previously, these uncertainties were treated in a conservative deterministic fashion, with each parameter assumed to be simultaneously and continuously at the worst point in its uncertainty range. The Statistical DNBR Evaluation Methodology determines a plant-specific and fuel-specific statistical design limit (SDL) for DNBR analysis. The SDL combines the correlation uncertainty with the uncertainties in key DNBR analysis input parameters. The Statistical DNBR Evaluation Methodology allows thermal hydraulic evaluations to be performed using nominal operating conditions as opposed to deterministic initial conditions (nominal conditions plus evaluated uncertainty).

In the performance of in-house DNB thermal-hydraulic evaluations, design limits and safety analysis limits are used to define the available retained DNBR margin for each application. The difference between the safety analysis (self-imposed) limit and the design limit is the available retained margin.

For deterministic DNB analyses, the DDL is set equal to the applicable code/correlation limit. For statistical DNB analyses, the design DNBR limit is set equal to the applicable SDL.

The Statistical DNBR Evaluation Methodology is applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWSC, which is initiated from zero power), and to the complete Loss of Flow, the Locked Rotor Accident, the Single Rod Cluster Control Assembly Withdrawal at Power, and feedwater system pipe break. The events modeled statistically are limited by the SDLs evaluated in the implementation of the Statistical DNBR Evaluation Methodology for MPS3, which is attached to this license amendment request (LAR) for NRC review and approval. In addition, there are events that are evaluated with deterministic models. These events are initiated from bounding operating conditions considering the nominal value and the appropriate uncertainty value, and require the application of uncertainty to the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and $F_{\Delta H}^{E}$ (engineering component), etc. The events modeled deterministically are limited by the deterministic design limits (DDLs) stated in DOM-NAF-2-P-A (Reference 8).

3.5.2 Conditions and Limitations

Topical Report VEP-NE-2-A was reviewed and generically approved by the NRC in May 1987. The fuel-specific and plant-specific implementation of the VEP-NE-2-A methodology must be submitted to the NRC for review and approval. The plant-specific analysis for implementation of the Statistical DNBR Evaluation Methodology at Millstone Power Station Unit 3 is attached to this LAR.

The NRC SER for VEP-NE-2-A listed the following conditions that must be met by any plant-specific implementation of this generic methodology:

a) The selection and justification of the Nominal Statepoints used to perform the plant-specific implementation must be included in the submittal.

b) The justification of the distribution, mean and standard deviation for all the statistically treated parameters must be included in the submittal.

c) The justification of the value of model uncertainty must be included in the plant-specific submittal.

d) For the relevant critical heat flux (CHF) correlations, justification of the 95/95 DNBR limit and the normality of the M/P distribution, its mean and standard deviation must be included in the submittal, unless there is an approved Topical Report documenting them.

3.5.3 Assessment

A DNBR evaluation method involving the statistical treatment of uncertainties is currently approved for application to MPS3 with the Westinghouse RFA-2 fuel design using the Westinghouse WRB-2M correlation (Reference 16). This DNB evaluation method (called the Revised Thermal Design

Procedure, RTDP) is used for the current safety analyses of record for MPS3. The RTDP is similar to Dominion's Statistical DNBR Evaluation Methodology.

The plant-specific, fuel-specific implementation of the Dominion Statistical DNBR Evaluation Methodology to MPS3 cores is included in a separate attachment. The attachment provides the specific justification for Conditions (a), (b) and (c) cited above. The implementation of this methodology to MPS3 cores will result in a Statistical Design Limit (SDL) that is plant-specific and fuel-type specific. Appendices C and D to Topical Report DOM-NAF-2-P-A have been approved by NRC (qualification of the VIPRE-D/WRB-2M code/correlation pair and qualification of the VIPRE-D/ABB-NV & VIPRE-D/WLOP code/correlation pairs, respectively), meeting Condition (d) above.

3.5.4 Summary

Statistical DNB evaluation methods are determined to be applicable to MPS3 and can be applied to MPS3 licensing analysis for reload core design and safety analysis. A plant-specific and fuel-specific application for Millstone Power Station Unit 3 cores containing Westinghouse RFA-2 fuel assemblies is included as an attachment to this LAR.

3.6 APPLICABILITY ASSESSMENT OF VIPRE-D METHODS - DOM-NAF-2-P-A, "REACTOR CORE THERMAL-HYDRAULICS USING THE VIPRE-D COMPUTER CODE"

3.6.1 Description of Methodology

The basic objective of core thermal-hydraulic analysis is the accurate calculation of reactor coolant conditions to verify that the fuel assemblies constituting the reactor core can safely meet the limitations imposed by departure from nucleate boiling (DNB) considerations. DNB, which could occur on the heating surface of the fuel rod, is characterized by a sudden decrease in the heat transfer coefficient with a corresponding increase in the fuel rod surface temperature. DNB is a concern in reactor design because of the possibility of fuel rod failure resulting from the increased fuel rod surface temperature. In order to preclude potential DNB-related fuel damage, a design basis is established and is expressed in terms of a minimum departure from nucleate boiling ratio (MDNBR). The departure from nucleate boiling ratio (DNBR) is the ratio of the predicted heat flux at which DNB occurs (i.e. the critical heat flux, CHF) and the local heat flux of the fuel rod. By imposing a DNBR design limit, adequate heat transfer between the fuel cladding and the reactor coolant is assured. If the MDNBR is greater than the design limit, there is adequate thermal margin. Thus, the establishment and qualification of DNBR design limits, for use with the VIPRE-D code, enables the accurate calculation of DNBR in order to assess and quantify core thermal margin.

VIPRE-D is the Dominion version of the computer code VIPRE (Versatile Internals and Components Program for Reactors), developed for EPRI (Electric Power Research Institute) by Battelle Pacific Northwest Laboratories in order to perform detailed thermal-hydraulic analyses to predict CHF and DNBR. VIPRE-D, which is based upon VIPRE-01, was adapted by Dominion for the specific analysis needs of the various Dominion nuclear power stations and their different fuel designs. The main enhancement made to VIPRE-01 to obtain VIPRE-D is the addition of several vendor proprietary CHF correlations. Additional customizations were made in VIPRE-D's input and output to integrate it into Dominion's thermal hydraulic methodologies.

Fleet Report DOM-NAF-2-P-A, "Reactor Thermal Hydraulics using the VIPRE-D Computer Code," (Reference 8) describes Dominion's use of the VIPRE-D computer code and the justification of all input, default parameters, and the specific modeling choices selected by Dominion. DOM-NAF-2-P-A demonstrates that the VIPRE-D core thermal-hydraulics methodology is appropriate for pressurized water reactor (PWR) licensing applications. The DNB model and analyses supporting Millstone 3 performed by Dominion have been done in accordance with the methodology of DOM-NAF-2-P-A. Consistent with the methodology of DOM-NAF-2-P-A, VIPRE-D has not been used to perform post-CHF modeling. Post-CHF modeling is performed using the RETRAN computer code (see Section 3.4). In addition, the various appendices to Topical Report DOM-NAF-2-P-A, document the qualification of several CHF correlations with the Dominion VIPRE-D computer code, as well as their associated code/correlation deterministic design limits (DDLs). Fleet Report DOM-NAF-2-P-A, including several appendices, has received generic NRC approval (Reference 8) and, as such, the VIPRE-D core thermal hydraulics methodology may be used for any of Dominion's nuclear facilities.

3.6.2 Conditions and Limitations

The conditions and limitations associated with the implementation of Fleet Report DOM-NAF-2-P-A can be split into three groups. The first group of conditions and limitations is related to the general use of the VIPRE-01 code, and its maintenance, and were imposed by the NRC in the SER for VIPRE-01 (References 17 and 18).

1.A. The application of VIPRE-D is limited to PWR licensing calculations modeling heat transfer regimes up to CHF. VIPRE-D is not used for post-CHF calculations or for BWR calculations.

1.B. VIPRE-D analyses only use DNB correlations that have been reviewed and approved by the NRC. The VIPRE-D DNBR calculations are within the NRC-approved thermal hydraulic parameter ranges of the DNB correlations. The correlation DNBR design limits associated with approved CHF correlations are derived or verified using fluid conditions predicted by the VIPRE-D code. Each CHF correlation is qualified or verified in the appropriate appendices to Fleet Report DOM-NAF-2-P-A.

1.C. Any plant-specific, fuel-specific application of the DOM-NAF-2-P-A methodology strictly uses the modeling choices approved in Topical Report DOM-NAF-2-P-A, which describes the intended uses of VIPRE-D for PWR licensing applications, and provides justification for Dominion's specific modeling assumptions, including the choice of two-phase flow models and correlations, heat transfer correlations and turbulent mixing models.

1.D. The Courant number, which is based on flow velocity, time step, and axial node size, is set greater than 1.0 in VIPRE-D transient calculations whenever a subcooled void model is used to ensure numerical stability and accuracy.

1.E. The code is maintained within Dominion's 10CFR50 Appendix B Quality Assurance program.

A second set of conditions and limitations is related to Dominion planned uses and applications for VIPRE-D, and are imposed by the NRC in the SER for DOM-NAF-2-P-A (Reference 8). According to this group of conditions and limitations, VIPRE-D can be used for:

2.A. Analysis of 14x14, 15x15 and 17x17 fuel in PWR reactors.

2.B. Analysis of DNBR for statistical and deterministic transients in the Final Safety Analysis Report (FSAR). Additional DNBR transients that are plant-specific may be analyzed in a plant specific application.

2.C. Steady state and transient DNB evaluations.

2.D. Development of reactor core safety limits (also known as core thermal limit lines, CTLs).

2.E. Providing the basis for reactor protection setpoints.

2.F. Establishing or verifying the deterministic code/correlation DNBR design limits of the various DNB correlations in the code. Each one of these DNBR limits is documented in an appendix to the original DOM-NAF-2-P-A Fleet Report.

The third and final set of conditions and limitations is related to the plant-specific and fuel-specific application of the VIPRE-D methodology:

3.A. Changes to the Technical Specifications (TS) to add Fleet Report DOM-NAF-2-P-A and applicable approved appendixes to the plant Core Operating Limit Report.

3.B. A plant-specific and fuel-specific Statistical Design Limit(s) for the relevant code/correlation pairs, to be used in statistical evaluations, which is evaluated following the Statistical DNBR Evaluation Methodology.

3.C. Any TS changes related to Over-Temperature Delta-T (OTΔT), Over-Power Delta-T (OPΔT), enthalpy rise factor (FΔH) or other reactor protection function, as well as revised reactor core safety limits.

3.D. Changes to the list of FSAR transients for which the code/correlations and limits apply.

3.6.3 Assessment

MPS3 is a standard 4-loop PWR that uses 17x17 fuel. This is one of the approved applications of the DOM-NAF-2-P-A methodology according to conditions 1.A and 2.A.

Conditions 1.B and 2.F are met by the use of the WRB-2M, W-3, ABB-NV and WLOP CHF correlations (which are NRC-approved), with the design limits and ranges of applicability listed in the appendices of Reference 8. The inclusion of Appendix C of DOM-NAF-2-P-A to the COLR list of references for Millstone Unit 3 licenses Dominion to use the W-3 CHF correlation for the RFA-2 fuel at MPS3; however, Dominion plans to use the ABB-NV and WLOP CHF correlations instead of the W-3 CHF correlation for the RFA-2 fuel at MPS3.

The application of DOM-NAF-2-P-A is approved for all NRC-approved PWR fuel types (Reference 8). Should Dominion elect to load in the MPS3 core a fuel product that uses a CHF correlation not previously qualified with VIPRE-D, a submittal would be made to the NRC in accordance with DOM-NAF-2-P-A to qualify the new CHF correlation with the VIPRE-D computer code and provide the associated code/correlation deterministic design limit.

Dominion has developed VIPRE-D models for MPS3 cores containing Westinghouse RFA-2 fuel. These models use all the modeling inputs approved in Fleet Report DOM-NAF-2-P-A, including two-phase flow models and correlations, heat transfer correlations and turbulent mixing models, thus meeting conditions (1.C) and (1.D). Should Dominion elect to load in the MPS3 core a different fuel product, Dominion would develop new VIPRE-D models for MPS3 cores containing the new fuel product, and these models would strictly follow all the modeling guidelines specified in Topical report DOM-NAF-2-P-A, thus meeting conditions (1.C) and (1.D). VIPRE-D and the MPS3 models have been generated and are maintained in accordance with Dominion's 10CFR50 Appendix B Quality Assurance program; thus meeting condition (1.E).

These models are used to evaluate the DNB-related design basis transients and accidents. The MPS3 transients and accidents are representative of the ones listed in Table 2.1-1 of Fleet Report DOM-NAF-2-P-A. The reactor thermal hydraulic conditions of the design basis transients and accidents are similar between MPS3 and the other Dominion units and within the qualification and capability of the VIPRE-D code. Therefore, conditions (2.B), (2.C), (2.D) and (2.E) are met.

These models are also used to evaluate the plant-specific and fuel-specific Statistical Design Limit (SDL) within the context of the Statistical DNBR Evaluation Methodology, which is included as a

separate attachment. These models are also used to verify the MPS3 setpoint functions, core thermal limit lines and FSAR statepoint and transient analyses. No changes to the MPS3 reactor protection system setpoints and core thermal limit lines are necessary to support the implementation of DOM-NAF-2-P-A. The list of FSAR transients for which the code/correlations apply is contained in Attachment C. Conditions (3.A), (3.B), (3.C), and (3.D) are thus met for application of DOM-NAF-2-P-A methods to MPS3.

3.6.4 Summary

The DOM-NAF-2-P-A core thermal hydraulic analysis methodology, including the applicable appendices, can be used for the thermal hydraulic evaluation of Millstone Power Station Unit 3 cores. The methods therein are determined to be applicable to MPS3 and can be applied to MPS3 licensing analysis for reload core design and safety analysis.

4 CONCLUSIONS

Dominion nuclear core design and safety analysis methods were assessed for applicability to MPS3. The Dominion reload nuclear design methods, as documented in the Dominion Topical Reports below, were determined to be applicable to MPS3, and can be employed in the licensing design and evaluation of reload cores for MPS3. The bases for this conclusion are provided in the Section 3.0 methodology applicability assessments.

- VEP-FRD-42-A, "Reload Nuclear Design Methodology" (Reference 2)
- VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications" (Reference 5)
- DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations" (Reference 6)
- VEP-FRD-41-P-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code" (Reference 1)
- VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 7)
- DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code" (Reference 8)

The applicability of the CMS and RETRAN methods to MPS3 have been further demonstrated through detailed validation analyses (see Section 3.3 and Attachment B). In addition, the plant-specific and fuel-specific application analysis to define the DNBR Statistical Design Limit (SDL) has been completed to support the applicability of the Statistical DNBR methods to MPS3 (see Attachment C).

MPS3 and other Dominion units (Surry and North Anna) use Westinghouse designs for nuclear steam supply system (NSSS) and reactor protection system (RPS). MPS3 has many design and operating similarities with the other Dominion Westinghouse units. MPS3 plant-specific considerations and features were evaluated and the differences from the methods as described in the Dominion Topical Report required for the application to MPS3 were identified in Section 3.0. The identified differences do not affect the conclusions on applicability of the methods to MPS3.

Dominion analysis methods will be applied to MPS3 consistent with the conditions and limitations described in the Dominion Topical Reports and in applicable NRC Safety Evaluation Reports (SER). The conditions and limitations for each method were addressed in Section 3.0. The conditions and limitations will be met when the method is applied to MPS3.

5 REFERENCES

1. Topical Report VEP-FRD-41, Rev. 0.2-P-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code," March 2015.

2. Topical Report VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.

3. Letter from E. S. Grecheck, Virginia Electric and Power Company, to USNRC, "Response to Request for Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report," December 2, 2002 (ADAMS Accession No. ML023430343).

4. Letter from Scott Moore (NRC) to D. A. Christian (VEPCO), "Acceptance of Topical Report VEP-FRD-42, Revision 2, Reload Nuclear Design Methodology, North Anna and Surry Power Stations, Units 1 and 2," June 11, 2003 (ADAMS Accession No. ML031621014).

5. Topical Report VEP-NE-1, Rev. 0.1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," August 2003.

6. Topical Report DOM-NAF-1, Rev. 0.0-P-A "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," June 2003.

7. Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

8. Fleet Report DOM-NAF-2-P-A, Rev. 0.3, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," September 2014.

9. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Proprietary), July 1985.

10. WCAP 10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-FQ Surveillance Technical Specification," (Proprietary), February 1994.

11. EPRI Report, NP-1850(A), "RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems."

12. EPRI NP-7450-CCM-A, "RETRAN-3D, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Revision 9," March 2014.

13. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," (Proprietary), April 1999.

14. Topical Report DOM-NAF-5, Rev.0.2-A, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," January 2011.

15. Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient," December 1984.

16. Letter from J. G. Lamb (NRC) to D. A. Christian (VEPCO), "Millstone Power Station, Unit No. 3 - Issuance of Amendment Re: Stretch Power Uprate (TAC No. MD6070)," August 12, 2008 (ADAMS Accession No. ML081610585).

17. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4," May 1, 1986.

18. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), "Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, 'VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores,' (TAC No. M79498)," October 30, 1993.

ATTACHMENT 5

RETRAN BENCHMARKING INFORMATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

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1.0 Introduction and Summary

1.1 Introduction

Topical report VEP-FRD-41-P-A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," (Reference 1) details the Dominion methodology for Nuclear Steam Supply System (NSSS) non-LOCA transient analyses. This methodology encompasses the non-LOCA licensing analyses required for the Condition I, II, III, and IV transients and accidents addressed in the Final Safety Analysis Report (FSAR). The VEP-FRD-41-P-A methods are also used in support of reload core analysis. In addition, this capability is used to perform best-estimate analyses for plant operational support applications. The material herein supports the applicability assessment of the VEP-FRD-41-P-A methods to Millstone Power Station Unit 3 (MPS3) for the stated applications.

1.2 Summary

This attachment provides a description of the RETRAN base model for MPS3 and results of benchmarking analyses using this model. The MPS3 model was developed in accordance with the methods in VEP-FRD-41-P-A, with certain noding changes noted below. This assessment confirms the conclusion that the Dominion RETRAN methods, as documented in topical report VEP-FRD-41-P-A, are applicable to MPS3 and can be applied to MPS3 licensing analysis for reload core design and safety analysis. Dominion analyses of MPS3 will employ the modeling in VEP-FRD-41-P-A, as augmented with the noding changes listed below. Thus, VEP-FRD-41-P-A, as augmented, is the Dominion methodology for analyses of non-LOCA NSSS transients for MPS3.

The MPS3 RETRAN base model contains the following alterations in noding with respect to the modeling that is documented in VEP-FRD-41-P-A.

a) The MPS3 model explicitly models the safety injection (SI) accumulators.
b) The MPS3 model has separate volumes for the steam generator inlet and outlet plenums.
c) The MPS3 model includes cooling paths between downcomer and upper head.

2.0 MPS3 RETRAN Model

The MPS3 RETRAN-3D Base Model and associated model overlays are developed using Dominion analysis methods described in the Dominion RETRAN topical report (Reference 1). The Dominion analysis methods are applied consistent with the conditions and limitations described in the Dominion topical report and in the applicable NRC Safety Evaluation Reports (SERs).

The MPS3 Base Model noding diagram for a representative loop is shown on Figure 2-1. Volume numbers are circled, junctions are represented by arrows, and the heat conductors are shaded. This model simulates all four reactor coolant system (RCS) loops and has a single-node steam generator (SG) secondary side, consistent with Dominion methodology. The SG primary nodalization includes 10 steam generator tube volumes and conductors. There is a multi-node SG secondary overlay that can be added to the Base Model for sensitivity studies although none of the analysis results presented herein utilize this overlay.

In addition to the base MPS3 model, an overlay deck is used to create a split reactor vessel model to use when analyzing Main Steam Line Break (MSLB) events, consistent with Dominion methodology. This overlay adds volumes to create a second, parallel flow path through the active core from the lower plenum to the upper plenum such that RCS loop temperature asymmetries can be represented. This noding is consistent with the method described in VEP-FRD-41-P-A. A noding diagram of the split reactor vessel is shown on Figure 2-2.

The base MPS3 model noding is virtually identical to the Surry (SPS) and North Anna (NAPS) models with the exception of some minor noding differences listed as follows.

a) The MPS3 model explicitly models the SI accumulators.
b) The MPS3 model has separate volumes for the SG inlet and outlet plenums.
c) The MPS3 model includes cooling paths between downcomer and upper head.

The SI accumulators are part of the MPS3 model because injection from the accumulators occurs in the current FSAR analysis for MSLB. The use of separate volumes for the inlet and outlet should have little effect on transient response since the fluid temperature in these volumes is generally the same as the connecting RCS piping. The cooling paths are included to appropriately model upper head T-cold conditions.

The Dominion models, including the MPS3 model, have some differences compared to the vendor RETRAN model that was used to perform the current FSAR analyses. Table 2-1 and the subsequent text discussion provide an overview of these differences. Additional details

concerning differences between the Dominion MPS3 and FSAR RETRAN models are discussed in the benchmarking analyses in Section 4.

A description of the Dominion RETRAN methodology is provided in Reference 1, where specific model details are discussed in Sections 4 and 5 of that reference.

Table 2-1 RETRAN Model Comparison of Key Characteristics

Parameter	Dominion	FSAR
Code Version:	RETRAN-3D in "02 mode"	RETRAN-02
Noding:		
Reactor Vessel	Single flow path (special split core overlay for MSLB only)	Multiple parallel flow paths
Steam Generator	Single node secondary. Five axial levels (10 nodes) for SG tubes primary side. Local Conditions Heat Transfer model available for loss of heat sink events.	Multi-node secondary.
Reactivity Model		
Doppler Feedback	Doppler temperature coefficient that is a function of T_{FUEL}.	Doppler-only power coefficient and a Doppler temperature coefficient effect driven by moderator temperature.
Moderator Feedback	Moderator temperature coefficient	Moderator density coefficient
Decay Heat	ANS 1979 Standard U-235 with 1500 day burn. Q = 190 MeV/fission. 1.0 Decay Heat Multiplier Bounds additional 2σ uncertainty	ANS 1979 Standard Bounds additional 2σ uncertainty

Figure 2-1 MPS3 Base Model Nodalization Diagram

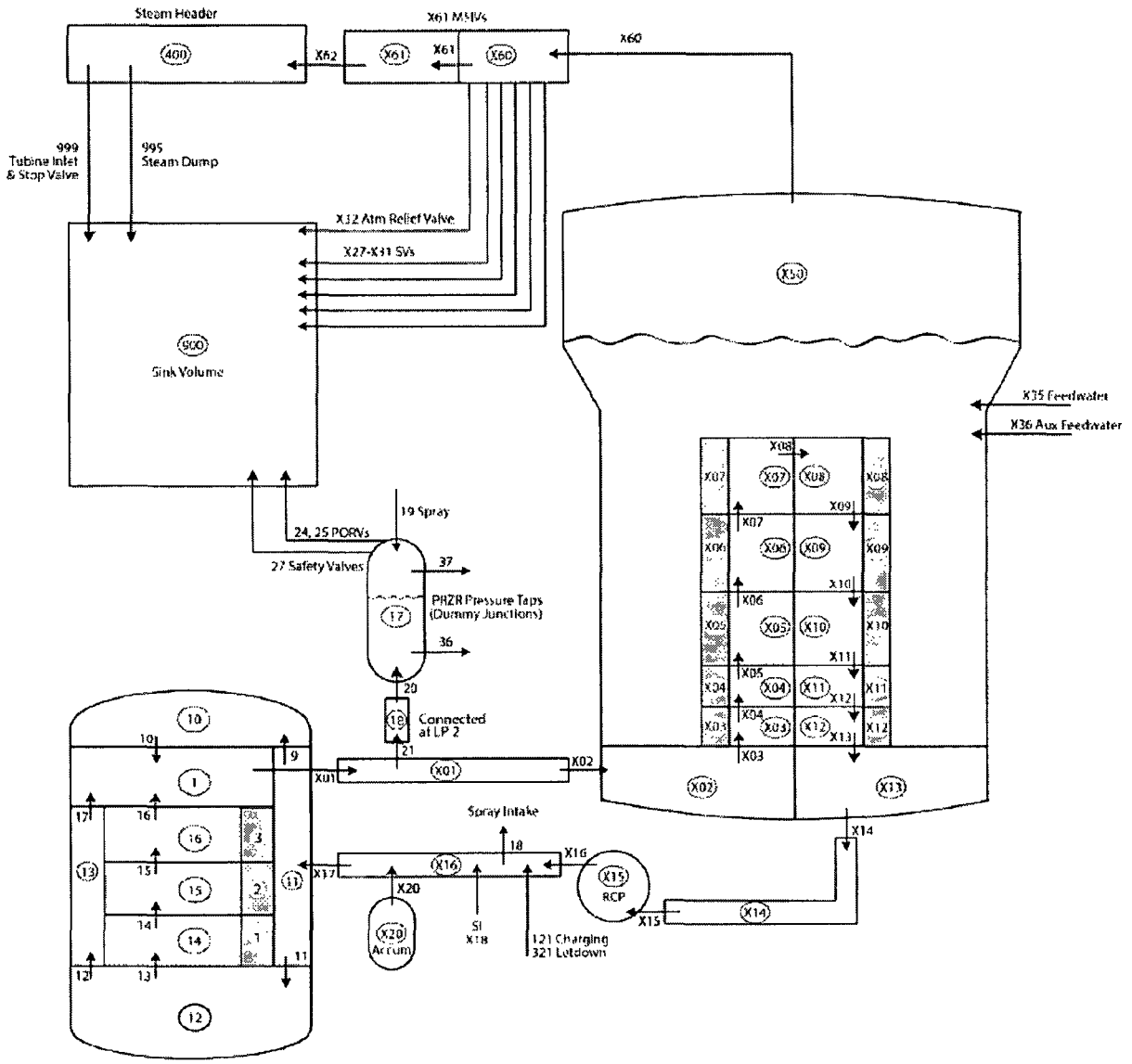
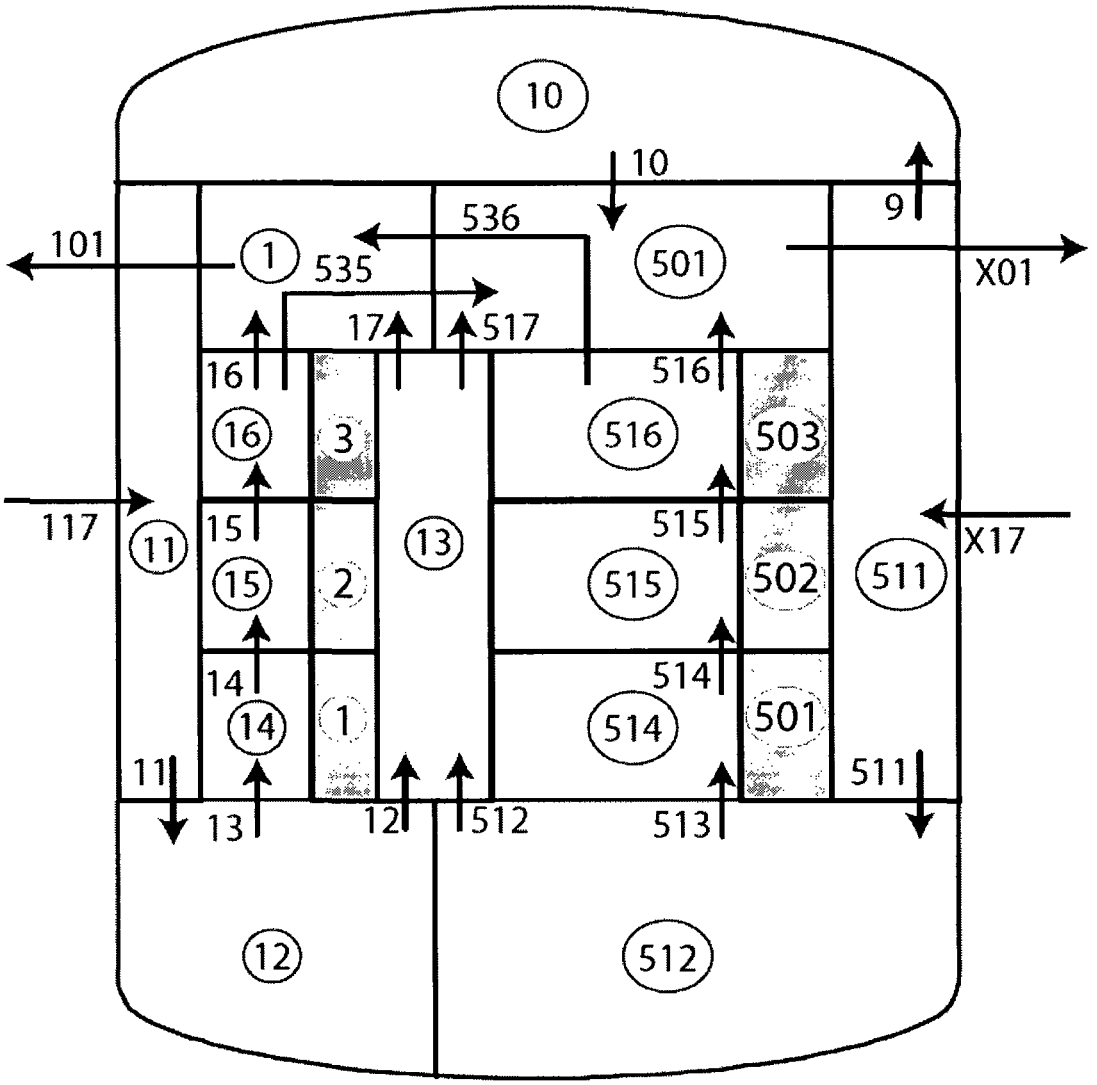


Figure 2-2 MPS3 Split Vessel Nodalization



3.0 Method of Analysis

Validation of the Dominion MPS3 RETRAN method involves comparison of RETRAN analyses to the MPS3 FSAR analysis of record (AOR) for select events. The Dominion analyses presented herein are not replacements for the existing AORs. These events represent a broad variation in behavior (e.g. RCS heatup, RCS cooldown/depressurization, reactivity excursion, loss of heat sink, etc.), and demonstrate the ability to appropriately model key phenomena for a range of transient responses. The transients selected for comparison with their corresponding MPS3 FSAR section are provided in Table 3-1. For each transient, an analysis is performed using the Dominion MPS3 RETRAN model and compared with the current FSAR analysis. Initial conditions and inputs are established for each benchmark to provide an adequate comparison of specific transient behavior.

Table 3-1 Transients Analyzed for FSAR Comparison

Transient	MPS3 FSAR Section
Main Steam Line Break	15.1.5
Loss of Load/Turbine Trip	15.2.3
Loss of Normal Feedwater	15.2.7
Locked Rotor	15.3.3
Control Rod Withdrawal at Power	15.4.2

4.0 Benchmarking Analysis Results

A summary for each transient comparison is presented in the following sections. Included in each section is an input summary identifying key inputs and assumptions along with differences from FSAR assumptions. A comparison of the results for key parameters is provided with an explanation of key differences between the Dominion and FSAR cases.

4.1 Loss of Load/Turbine Trip

The Loss of Load/Turbine Trip (LOL) event is defined as a complete loss-of-steam load and turbine trip from full power without a direct reactor trip, resulting in a primary fluid temperature rise and a corresponding pressure increase in the primary system. This transient results in degraded steam generator heat transfer, reactor coolant heatup and pressure increase following a manual turbine trip.

The LOL transient scenario presented here was developed to analyze primary RCS overpressurization. It is initiated by decreasing both the steam flow and feedwater flow to zero immediately after a manual turbine trip. The input summary is provided in Table 4.1-1.

Table 4.1-1 LOL Input Summary

Parameter	Value	Notes
Initial Conditions		
NSSS Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design
Vessel T_{AVG} (F)	576.5	Low Tavg plus uncertainty
Pressurizer Pressure (psia)	2200	Includes -50 psia uncertainty
Pressurizer Level (%)	71.6	Nominal + 7.6%
SG Level (%)	50.0	Nominal
SG tube plugging (%)	10	Maximum
Pump Power (MW/Pump)	5.0	Maximum
Assumptions/Configuration		
Reactor trip	-	only Hi Pzr Pressure is active
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
AFW flow	-	Not credited
Reactivity Parameters		
Doppler Reactivity Feedback	Least Negative	
Moderator Feedback	Most Positive	

Results - LOL

Pressure in the RCS increases during a LOL due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open as well as the main steam safety valves (MSSV). The pressurizer pressure response is shown on Figure 4.1-1, RCP outlet pressure in Figure 4.1-2, and the peak RCS pressure values are listed in Table 4.1-2. The Dominion case predicts a pressurizer pressure and RCP outlet pressure response that agrees very well with the FSAR results past the point of peak RCS pressure.

Following the initial decrease in primary system pressure, the FSAR pressure levels out where the Dominion case results continue to decrease. The difference is due to differing secondary safety valve modeling in the vendor model, specifically in that the Dominion model includes the modeling of blowdown in the main steam safety valves and the vendor model does not. Hence, more energy is removed through the secondary system in the Dominion case once the main steam safety valves actuate than is removed from the secondary system in the vendor model.

Figure 4.1-3 shows the power response is nearly identical both before and after the reactor trip on high pressurizer pressure and control rod insertion. The Dominion case trips slightly earlier than the FSAR data because of the higher RCS pressurization rate.

The Dominion model vessel inlet temperature, Figure 4.1-4, and coolant average temperature, Figure 4.1-5, agrees in trend and rate of increase although the response lags the FSAR response before the inlet temperature peaks at a slightly lower value. This indicates that the FSAR steam generator heat transfer degrades sooner than what is predicted by Dominion model and is attributed to the difference expected between the use of a multi-node steam generator (MNSG) in the FSAR model and the single-node steam generator (SNSG) model employed in the Dominion model. Overall, both the Dominion model and FSAR models exhibit similar trends in the temperature responses and the differences have no effect on peak RCS pressure.

Table 4.1-2 LOL RCS Overpressure Results

Parameter	Dominion	FSAR
Sequence of Events:		
High Pressurizer Pressure Setpoint Reached	5.8	6.2
Peak RCS Pressure (sec)	9.2	9.9
Peak RCS Pressure (psia)	2717.19	2729.41

Figure 4.1-1 LOL - Pressurizer Pressure

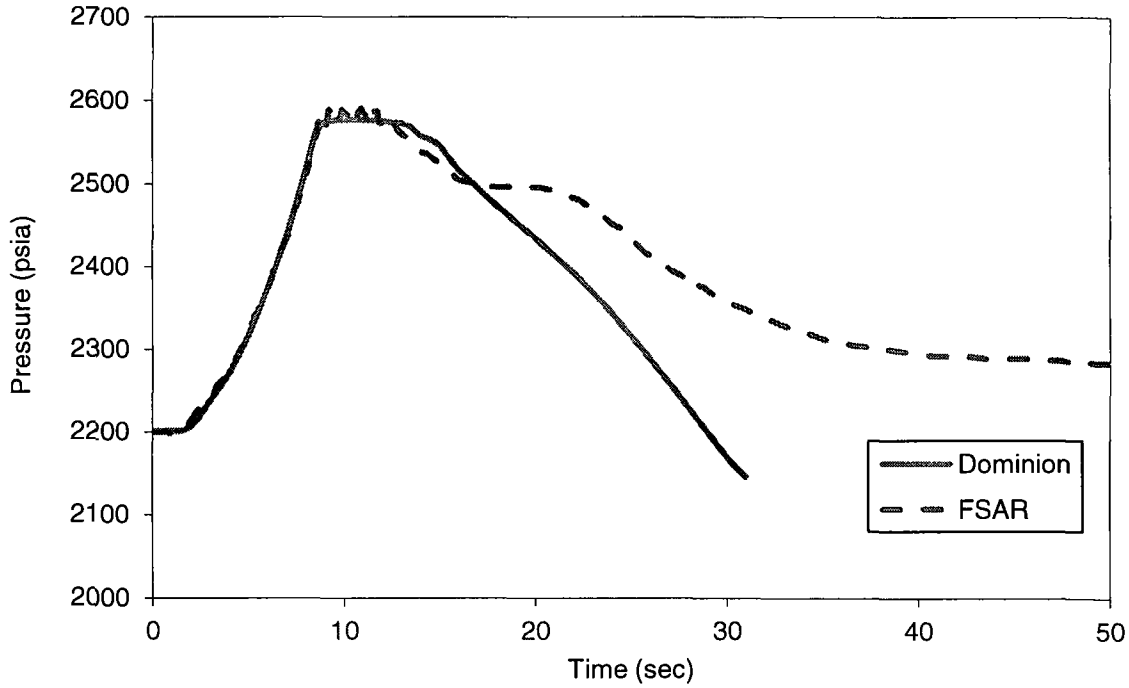


Figure 4.1-2 LOL – RCP Outlet Pressure

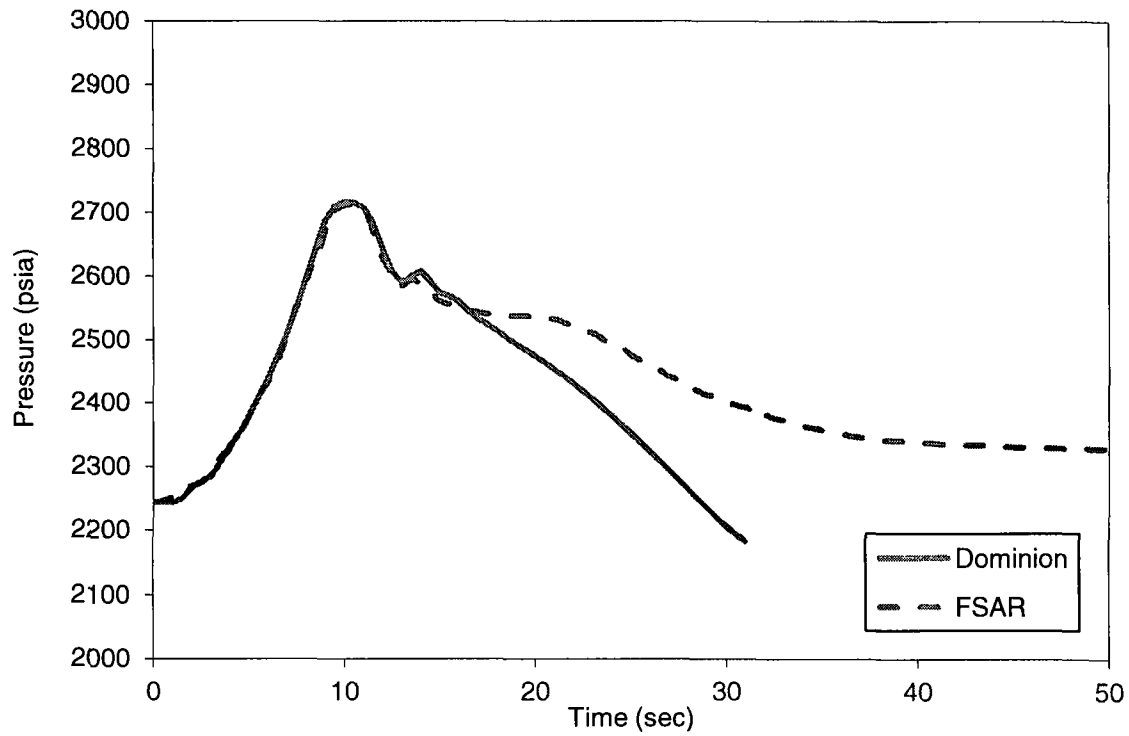


Figure 4.1-3 LOL – Nuclear Power

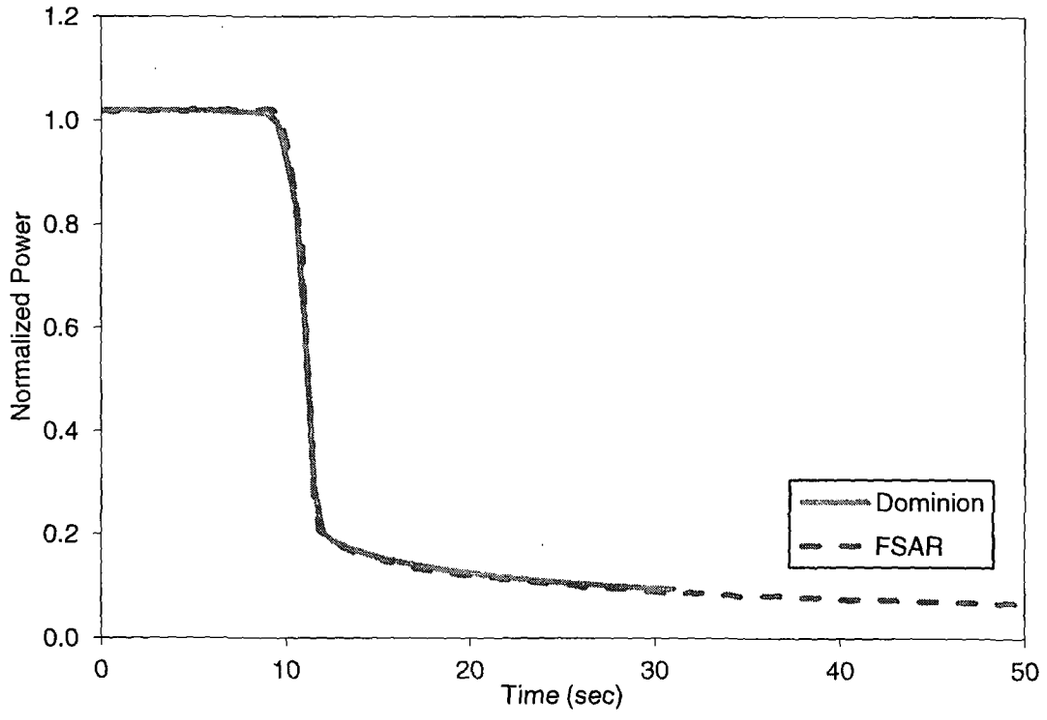


Figure 4.1-4 LOL– Vessel Inlet Temperature

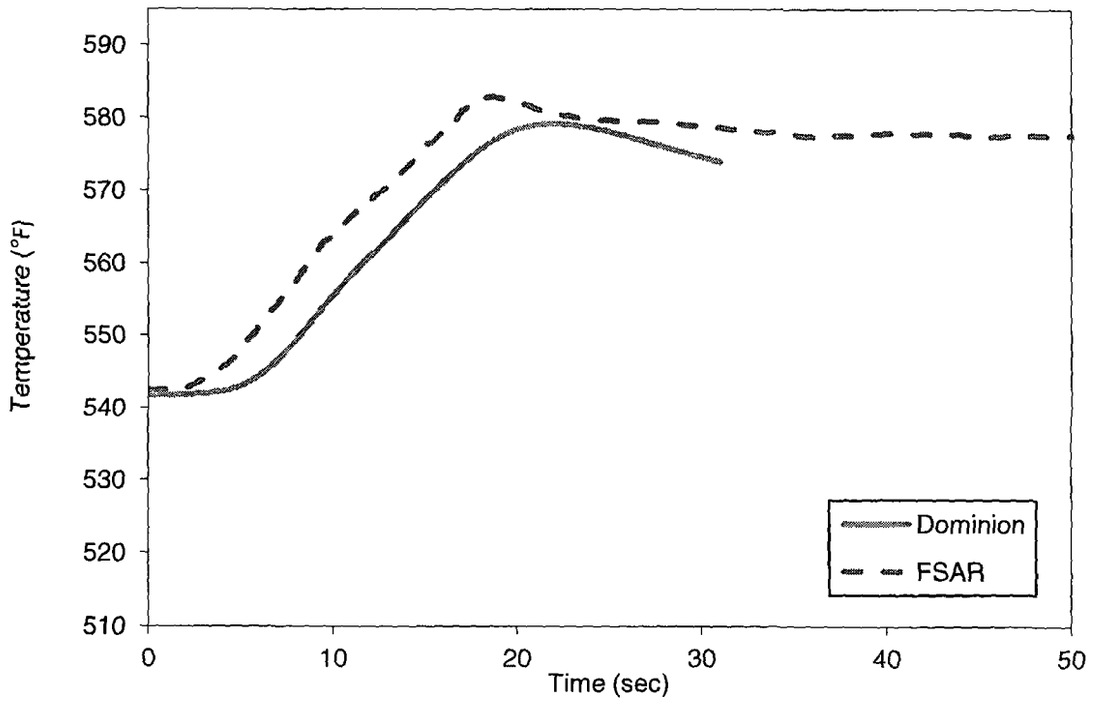
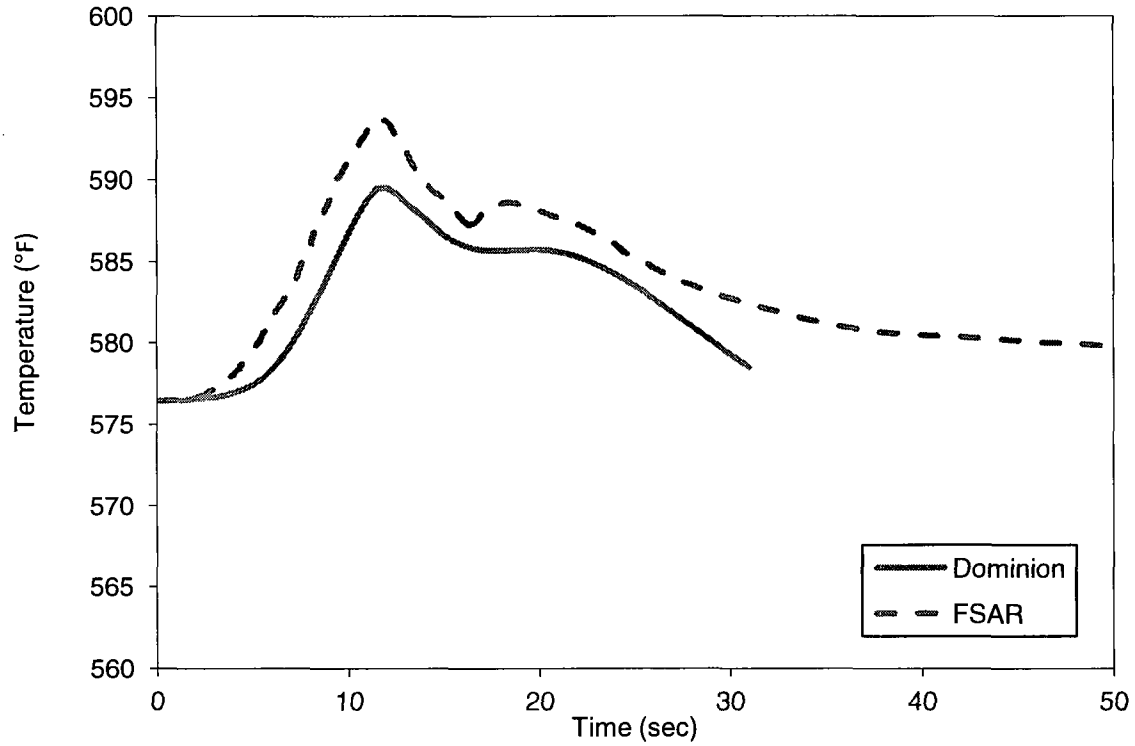


Figure 4.1-5 LOL – Vessel Average Temperature



Summary - LOL

The Dominion MPS3 analysis provides results that are similar to the FSAR analysis for the LOL event. The RCS peak pressures are essentially the same although the pressures diverge somewhat later in the event after pressure relief begins due to differences in MSSV modeling. There are small differences in the RCS temperature response due to differences in the SG models, however, this has no effect on the RCS peak pressure. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

4.2 Locked Rotor

The Locked Rotor / Shaft Break (LR) event is defined as an instantaneous seizure of a Reactor Coolant Pump (RCP) rotor, rapidly reducing flow in the affected reactor coolant loop leading to a reactor trip on a low-flow signal from the Reactor Protection System. The event creates a rapid expansion of the reactor coolant and reduced heat transfer in the steam generators, causing an insurge to the pressurizer and pressure increase throughout the reactor coolant system (RCS).

The LR transient scenario presented here was developed to analyze primary RCS overpressurization. It is initiated by setting one RCP speed to zero as the system is operating at full power. The reactor coolant low loop flow reactor trip is credited, with a setpoint of 85% of the initial flow. The input summary is provided in Table 4.2-1. Most of the input parameters are the same as those used in the FSAR Chapter 15 analyses.

Table 4.2-1 LR Input Summary

Parameter	Value	Notes
Initial Conditions		
NSSS Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	594.5	Nominal + 5°F
Pressurizer Pressure (psia)	2300	Includes +50 psia uncertainty
Pressurizer Level (%)	64	Nominal
SG Level (%)	50	Nominal
Assumptions/Configuration		
Reactor trip	-	Only Low RCS Loop Flow is credited
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
AFW flow	-	Not credited
SG tube plugging (%)	10	Max value
Reactivity Parameters		
Doppler Reactivity Feedback	Most Negative	Dominion model adjusted to use FSAR Doppler Power Coefficient
Moderator Feedback	Most Positive	

Results – LR RCS Overpressure Case

Pressure in the RCS increases during a LR event due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open. The magnitude of the Dominion model pressure response both in the reactor vessel lower plenum, Figure 4.2-1, and at the RCP exit, Figure 4.2-2, is greater than the FSAR model response, while following the same trends as the FSAR data. At the limiting point in the

transient response, the Dominion model conservatively predicts a pressure approximately 58 psi greater than the FSAR model in the reactor vessel lower plenum. The difference between the Dominion model and FSAR model's peak responses is the same at the RCP exit as in the lower plenum.

The Dominion faulted loop flow response (Figure 4.2-3) and unfaulted loop flow response (Figure 4.2-4) are in good agreement with the FSAR model response up to or just beyond the point of rod insertion . Following reactor trip there is some divergence in the unfaulted loop flow trends, which are consistent with the core heat flux predictions and assumed minor differences in the loop friction losses between the Dominion and FSAR models. With respect to the faulted loop flow response, the maximum reverse flow seen in the FSAR model is slightly greater than seen in the Dominion model, which is also attributed to small differences in the loop friction losses between the Dominion and FSAR models.

For the total core inlet flow response (Figure 4.2-5), the Dominion model predicts a lower flow than the FSAR model for approximately the first 4 seconds of the transient. After 4 seconds the FSAR and Dominion model core flow responses cross and the Dominion model predicts a slightly higher core flow rate. Ultimately the limiting point in the transient occurs prior to 4 seconds such that RETRAN-3D produces a more limiting response than the FSAR model for the Locked Rotor/Shaft Break event.

The nuclear power response, Figure 4.2-6, predicted by the Dominion model agrees well with the FSAR data, with the Dominion model response slightly over predicting power during rod insertion following the reactor trip on low RCS flow. Similarly, the Dominion model core heat flux response, Figure 4.2-7, also slightly over predicts the FSAR model's response in the same time frame during control rod insertion. Additionally, the Dominion model heat flux response shows a slightly larger decrease at the initiation of the event over the decrease seen in the FSAR data. Both the initial under prediction of the heat flux response, followed by an over prediction during the rod insertion is indicative of the fuel rod heat transfer being modeled differently in the vendor methods than in the Dominion model. However, the over prediction of both nuclear power and heat flux will lead to conservative results at the limiting point in the transient for both RCS overpressurization and DNB during rod insertion. Overall the nuclear power and heat flux predictions are very similar.

A summary of the LR transient analysis comparison is provided in Table 4.2-2.

Table 4.2-2 LR RCS Overpressure Results

Parameter	Dominion	FSAR
Sequence of Events:		
Reactor Trip on Low RCS Flow (sec)	1.1	0.1
Peak RCS Pressure (sec)	3.8	4.1
Peak RCS Pressure (psia)	2680.75	2616.6

Summary - LR RCS Overpressure Case

The Dominion Millstone analysis provides responses that are similar to the FSAR analysis for the LR event, with the Dominion model predicting higher peak RCS pressures. Differences are attributed to loop friction losses and fuel rod modeling differences. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

Figure 4.2-1 LR – Reactor Vessel Lower Plenum Pressure

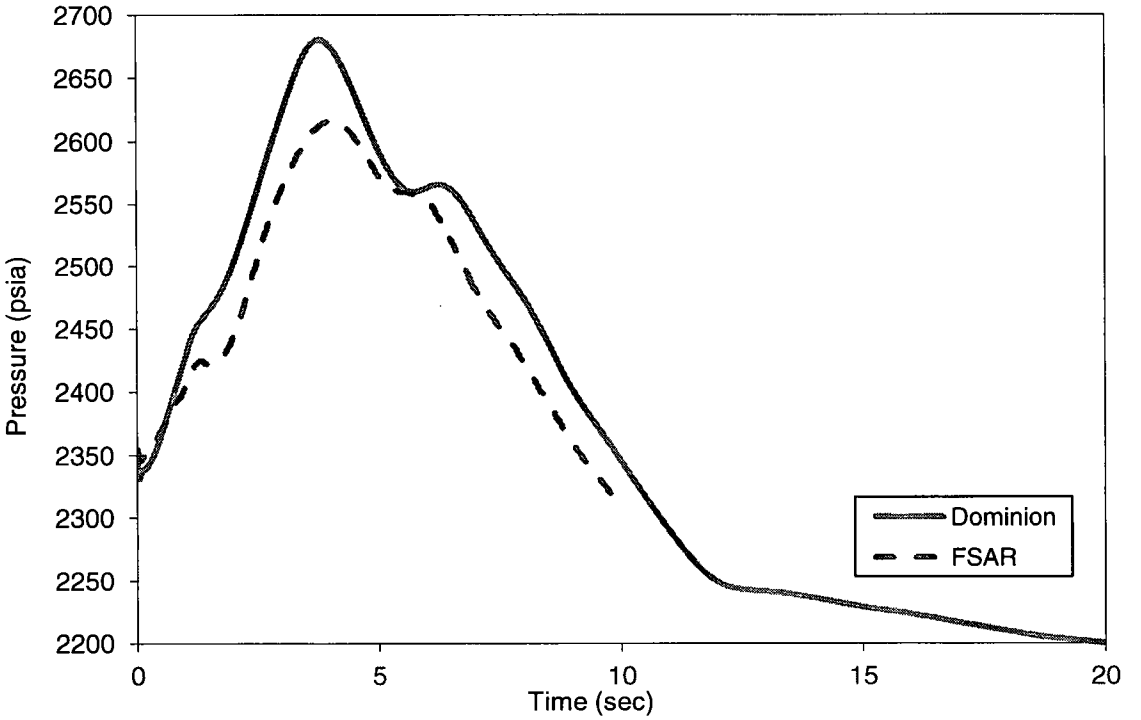


Figure 4.2-2 LR – RCP Outlet Plenum Pressure

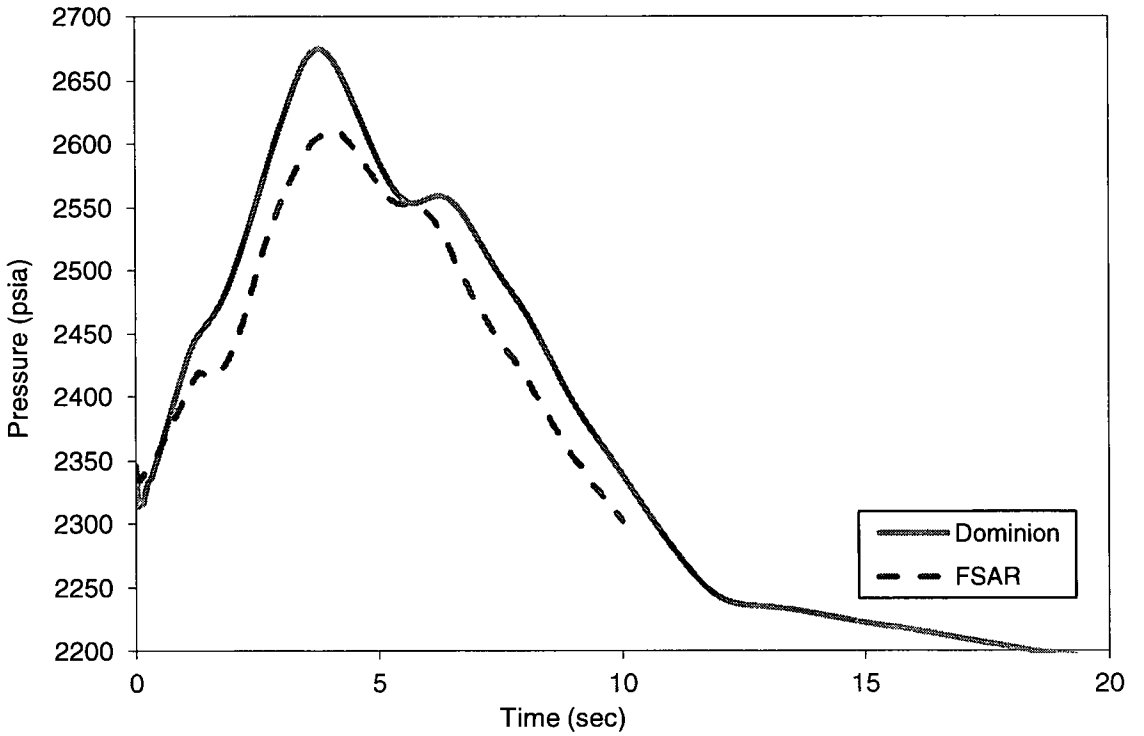


Figure 4.2-3 LR – Faulted Loop Normalized Flow

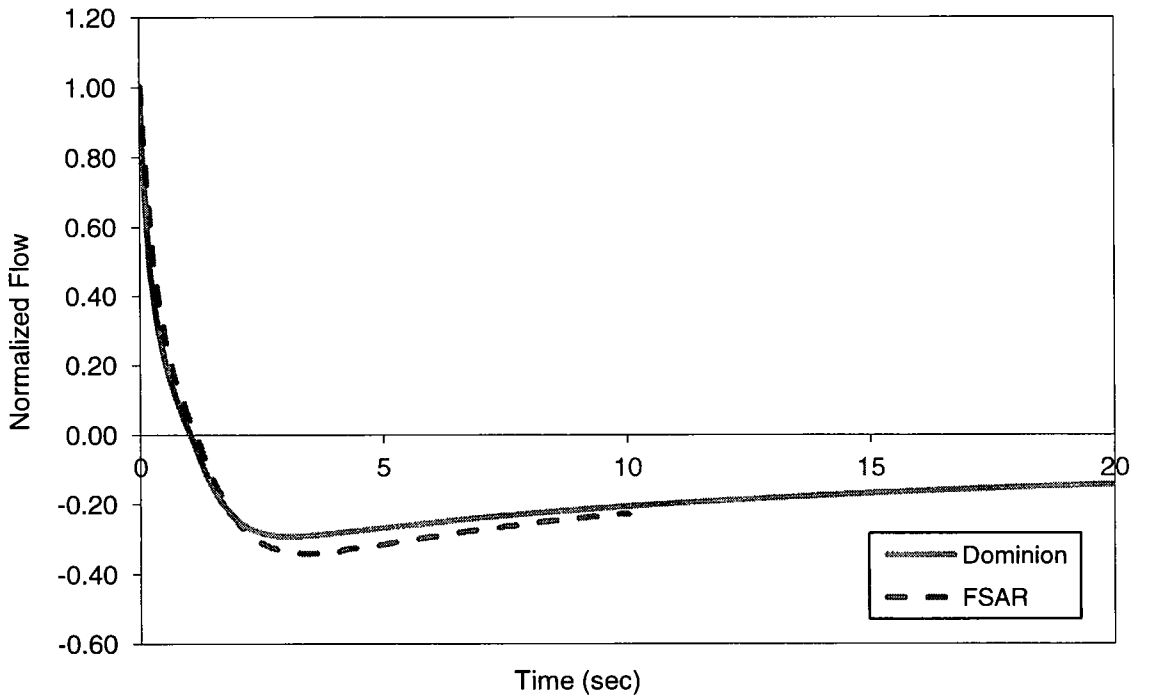


Figure 4.2-4 LR – Unfaulted Loop Normalized Flow

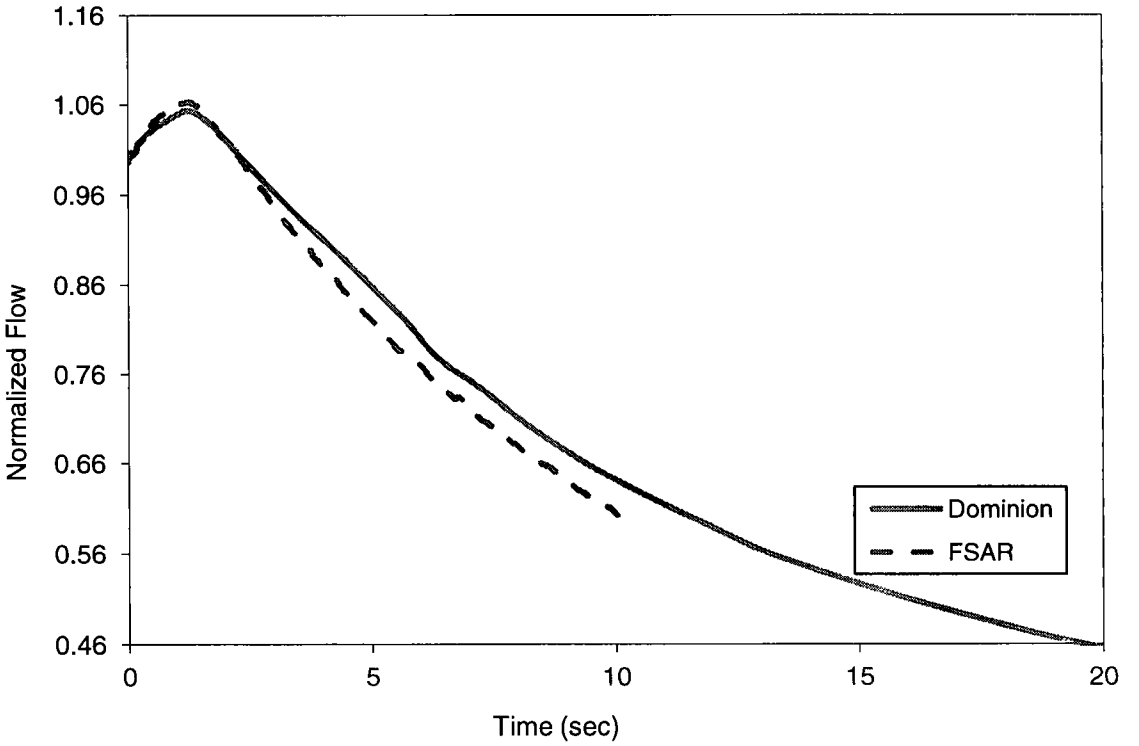


Figure 4.2-5 LR – Core Inlet Normalized Flow

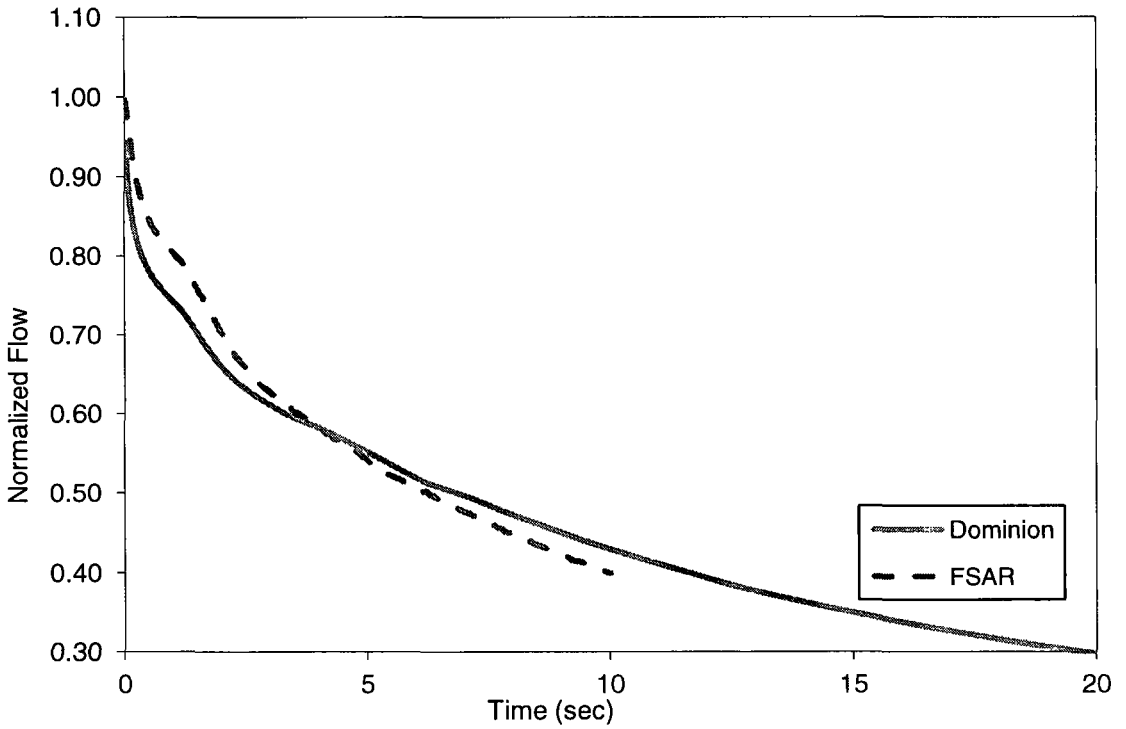


Figure 4.2-6 LR – Nuclear Power

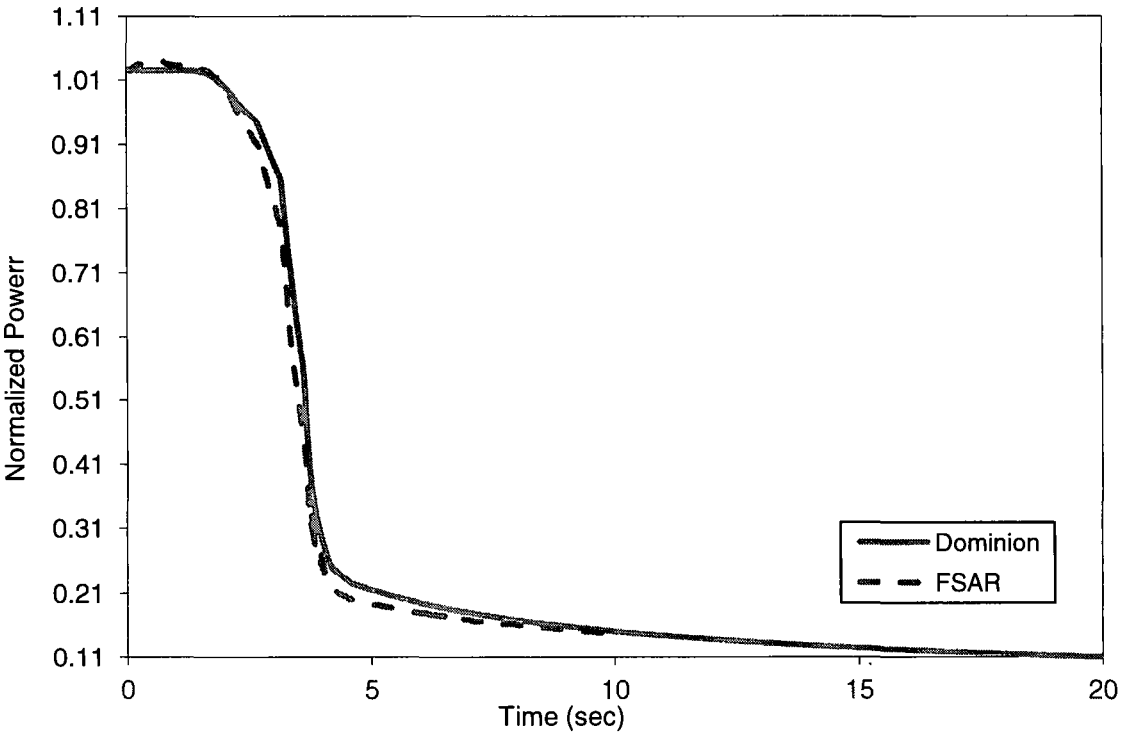
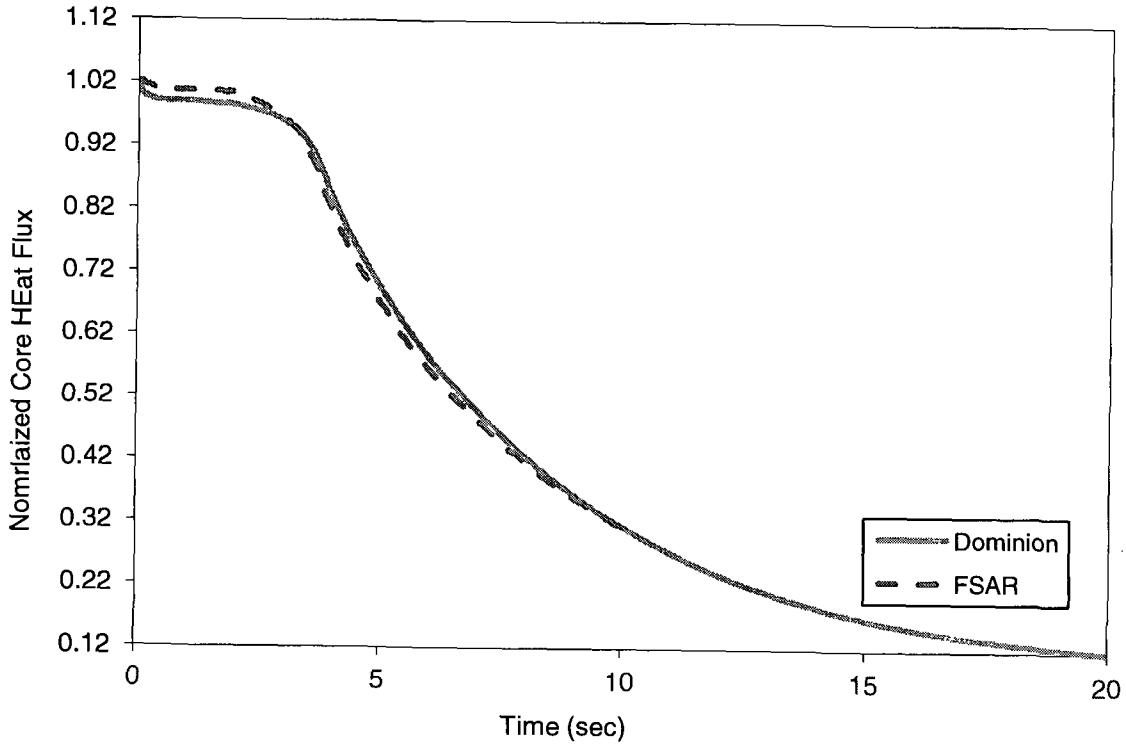


Figure 4.2-7 LR – Core Heat Flux



LR Peak Cladding Temperature

The Locked Rotor event is also analyzed to demonstrate that a coolable core geometry is maintained. A hot spot evaluation is performed to calculate the peak cladding temperature and oxidation level. The Dominion Hot Spot model is described in Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient." (Reference 2) The Dominion Hot Spot model was used to evaluate the MPS3 PCT and oxidation level for the LR event.

The Dominion hot spot model is used to predict the thermal-hydraulic response of the fuel for a hypothetical core hot spot during a transient. The hot spot model describes a one-foot segment of a single fuel rod assumed to be at the location of the peak core power location during a transient. The hot spot model uses boundary conditions from the LR system transient analysis to define inlet flow and core average power conditions. The hot spot model uses MPS3-specific values for fuel dimensions, fuel material properties, fluid volume, and junction flow areas.

The hot spot model is run to 0.1 seconds and a restart file is saved. Upon restart, the fuel/cladding gap conductance (thermal conductivity) is modified to simulate gap closure by setting the gap heat transfer coefficient to 10,000 Btu/ft^2-hr-°F for a gap conductance of 2.708 Btu/ft-hr-°F. The hot spot model input summary is provided in Table 4.2-3. Most of the input parameters are the same as those used in the FSAR Chapter 15 analyses. Where differences from the FSAR inputs exist, they are indicated in the Notes column.

Table 4.2-3 Hot Spot Model Input Summary

Parameter	Value	Notes
Computer Code Used	RETRAN-3D	FSAR uses VIPRE
Initial Conditions		
Ratio of Initial to Nominal Power	1.02	
RCS Flow (gpm)	363,200	
Hot Spot Peaking Factor	2.60	
Assumptions/Configuration		
Pre-DNB Film Heat Transfer Coefficient	Thom	
Time of DNB (sec)	0.1	
Post DNB Film Boiling Heat Transfer Coefficient	Bishop-Sandberg-Tong	
Fuel Pin Model		
Post DNB Gap Heat Transfer Coefficient (Btu/hr-ft2-°F)	10,000	
Gap Thermal Expansion Model activated?	Yes	
Zircaloy-Water Reaction activated?	Yes	

LR Peak Cladding Temperature Results

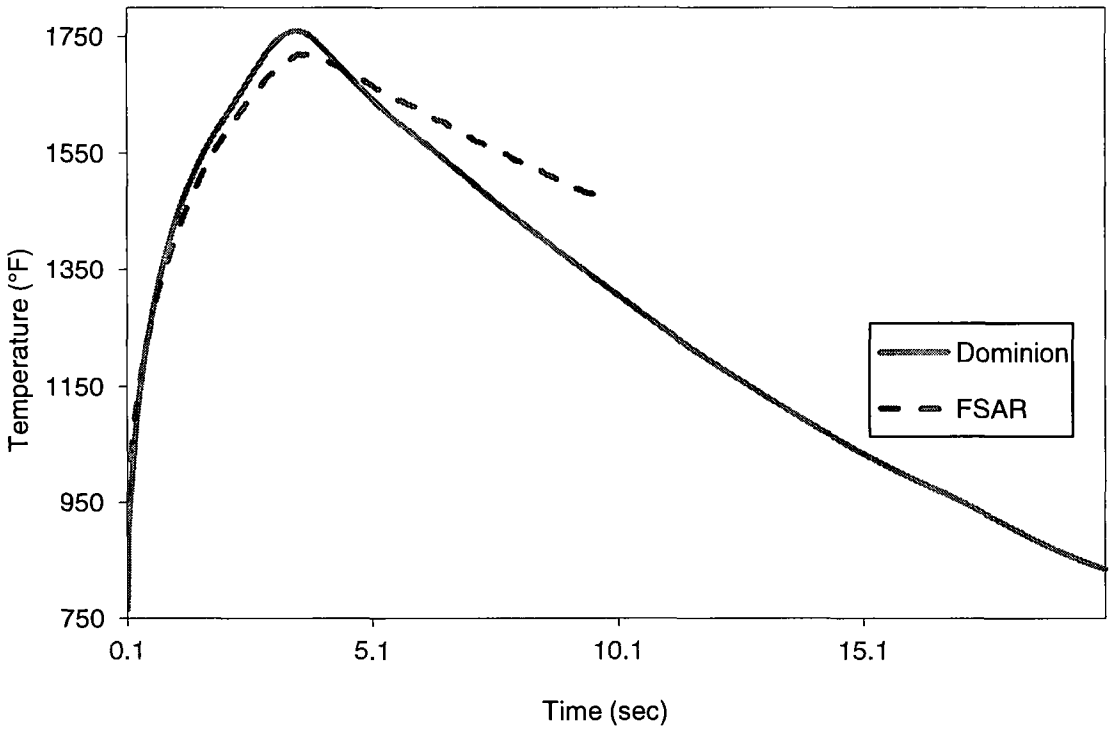
The peak cladding temperature obtained from Dominion's MPS3 hot spot model for the locked rotor event is 1760.0 °F. The maximum zircaloy-water reaction depth is 3.60281E-06 feet, which corresponds to approximately 0.19% by weight based on the nominal cladding thickness of 1.875E-03 feet. A summary of the LR Peak Cladding Temperature Hot Spot analysis comparison is provided in Table 4.2-4. The cladding inner surface temperature is shown in Figure 4.2-8.

Table 4.2-4 LR Hot Spot Results

Parameter	Dominion	FSAR
Peak Cladding Temperature	1760.0 °F	1718.3 °F
Maximum Zr-water reaction (w/o)	0.19	0.22

The Dominion peak cladding temperature and maximum oxidation values are comparable to the FSAR values. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

Figure 4.2-8 LR Hot Spot – Cladding Inner Surface Temperature



4.3 Loss of Normal Feedwater

The Loss of Normal Feedwater (LONF) event causes a reduction in heat removal from the primary side to the secondary system. Following a reactor trip, heat transfer to the steam generators continues to degrade resulting in an increase in RCS fluid temperature and a corresponding insurge of fluid into the pressurizer. There is the possibility of RCS pressure exceeding allowable values or the pressurizer becoming filled and discharging water through the relief valves. The event is mitigated when Auxiliary Feedwater (AFW) flow is initiated and adequate primary to secondary side heat removal is restored. This analysis shows that the AFW system is able to remove core decay heat, pump heat and stored energy such that there is no loss of water from the RCS and pressure limits are not exceeded. The LONF input summary is provided in Table 4.3-1.

Table 4.3-1 LONF Input Summary

Parameter	Value	Notes
Initial Conditions		
NSSS Power (MW)	3723	Includes 2% uncertainty
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	583	HFP nominal + 6 F
RCS Pressure (psia)	2300	Nominal + 50 psi
Pressurizer Level (%)	71.6	Nominal + 5%
SG Mass (lbm)	~89000	Dominion model adjusted to be consistent with FSAR analysis
Assumptions/Configuration		
Low-Low Level Reactor Trip Setpoint	0%	Percent of narrow range span
Pressurizer: sprays, heaters, PORVs	-	Assumed operable
AFW Temperature (F)	120	Max value
AFW Pump configuration	-	One motor-driven pump per SG
Auxiliary feedwater flow rate (gpm)	-	Variable as function of SG press.
Local Conditions Heat Transfer model	active	SG secondary side FSAR= multi-node SG
Decay Heat Multiplier	1.2133	FSAR decay heat constants are applied for this case
Reactivity Parameters		
Doppler Reactivity Feedback	Most negative	Dominion model adjusted to use FSAR Doppler Power Coefficient
Moderator Feedback	Most Positive	

Results - LONF

The results for the LONF comparison analysis are presented in Table 4.3-2 and Figures 4.3-1 through 4.3-7. The loss of feedwater flow to the steam generators (SG) results in a reduction in SG level until a reactor trip occurs on Low-Low SG level. Normalized power is shown on Figure 4.3-1 and normalized core heat flux in Figure 4.3-2. The nuclear power response and heat flux response predicted by the Dominion model are in excellent agreement with the FSAR data, indicating that the scram on low-low steam generator level occurred at essentially the same time shown for the FSAR data. The results continue to demonstrate good agreement through the end of the event.

Figure 4.3-3 shows the steam generator pressure response. The Dominion steam generator pressure is initialized at a slightly different pressure than the FSAR model because the Dominion model initial condition is adjusted to minimize the steam generator area adjustment. Between 10 and 34 seconds the FSAR pressure increases more rapidly to a pressure ~43 psi greater than the Dominion model prediction when the steam line is isolated. This difference is attributed to differing heat transfer degradation in the MNSG model used in the FSAR analysis versus the SNSG model used in the RETRAN-3D model. Steam line isolation occurs at nearly the same time, causing pressure to increase rapidly. The peak pressure is limited by the main steam safety valves (MSSVs), resulting in an almost identical peak pressure in both the Dominion and FSAR responses. However, the Dominion model pressure decreases following the peak value, where the FSAR model response remains at a constant value near the peak value, due to differences in MSSV modeling. Figure 3.1-4 shows the steam generator liquid mass. The steam generator liquid mass depletes faster in the Dominion cases than in the FSAR cases. This is consistent with the increased relief flow as shown in the steam generator pressure response.

The response in the pressurizer is shown in Figures 4.3-5 and 4.3-6. Between the FSAR and Dominion model, the pressure responses are in good agreement until around 45 – 50 seconds where the Dominion pressure is lower than the FSAR, reflecting less heat transfer degradation during this period. This is followed by a second pressure peak that is higher for Dominion than the FSAR. Based on the sharpness of the Dominion peak compared with the FSAR data, this difference is most likely driven by differences in the pressurizer spray models and primary to secondary heat transfer.

For the pressurizer water volume, shown in Figure 4.3-6, the Dominion model results follow the same trends as the FSAR data, but drops lower in the period from 63 to 900 seconds, then demonstrates a strong insurge during the second heat-up period in the transient while peaking at a somewhat lower value than the FSAR. The difference seen in the pressurizer

volume results is primarily due to the previously discussed MSSV modeling differences and the resultant increased steam release from the Dominion model compared to the FSAR model as well as possible differences in the pressurizer spray models.

Table 4.3-2 LONF Results

Parameter	Dominion	FSAR
Peak PZP Liquid Volume (ft3)	1588.96	1730.85

Figure 4.3-1 LONF - Nuclear Power

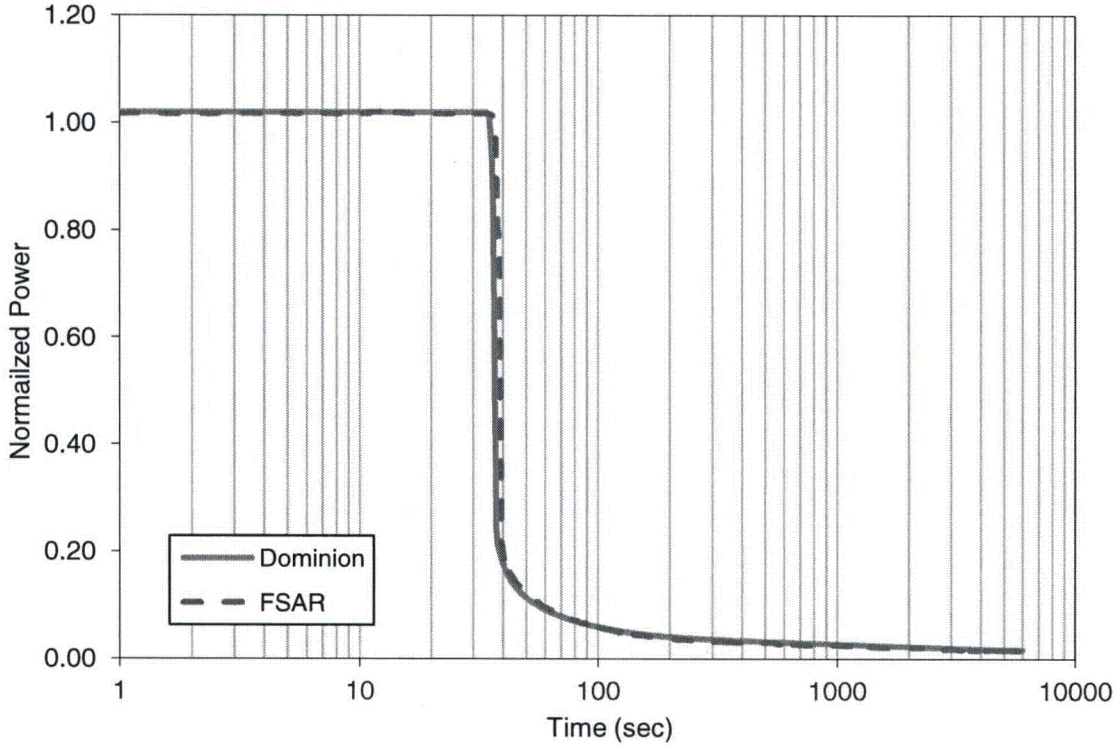


Figure 4.3-2 LONF - Normalized Core Heat Flux

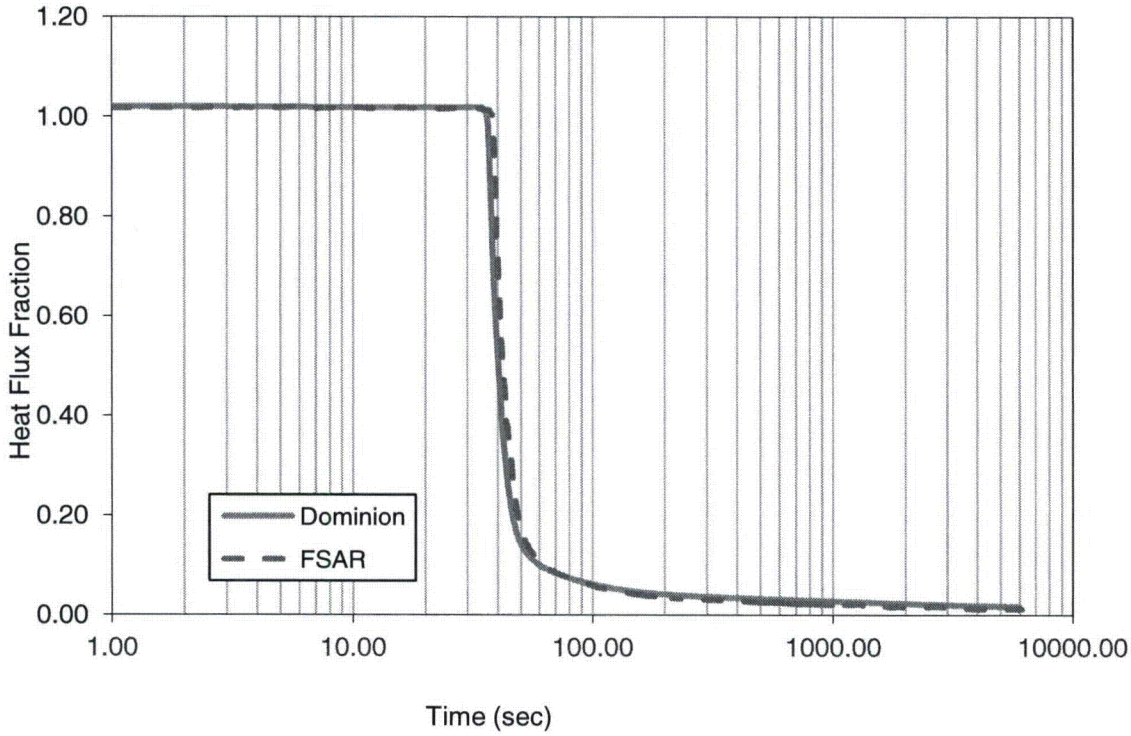


Figure 4.3-3 LONF – Steam Generator Pressure

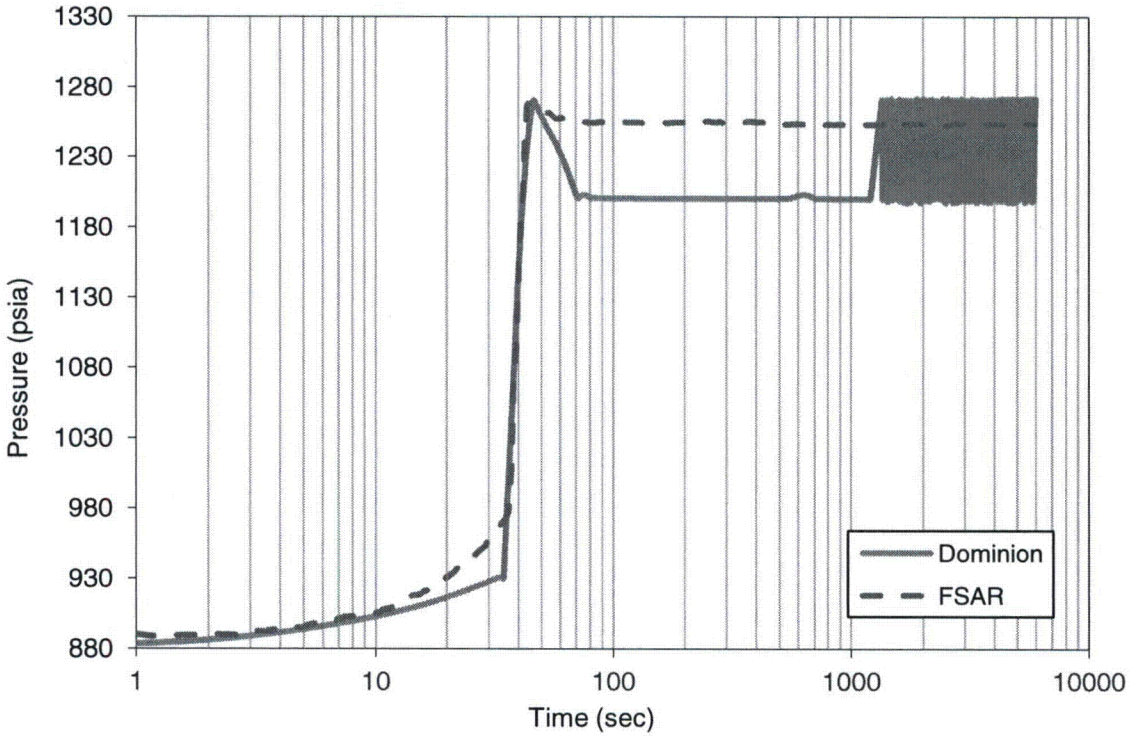


Figure 4.3-4 LONF – Steam Generator Liquid Mass

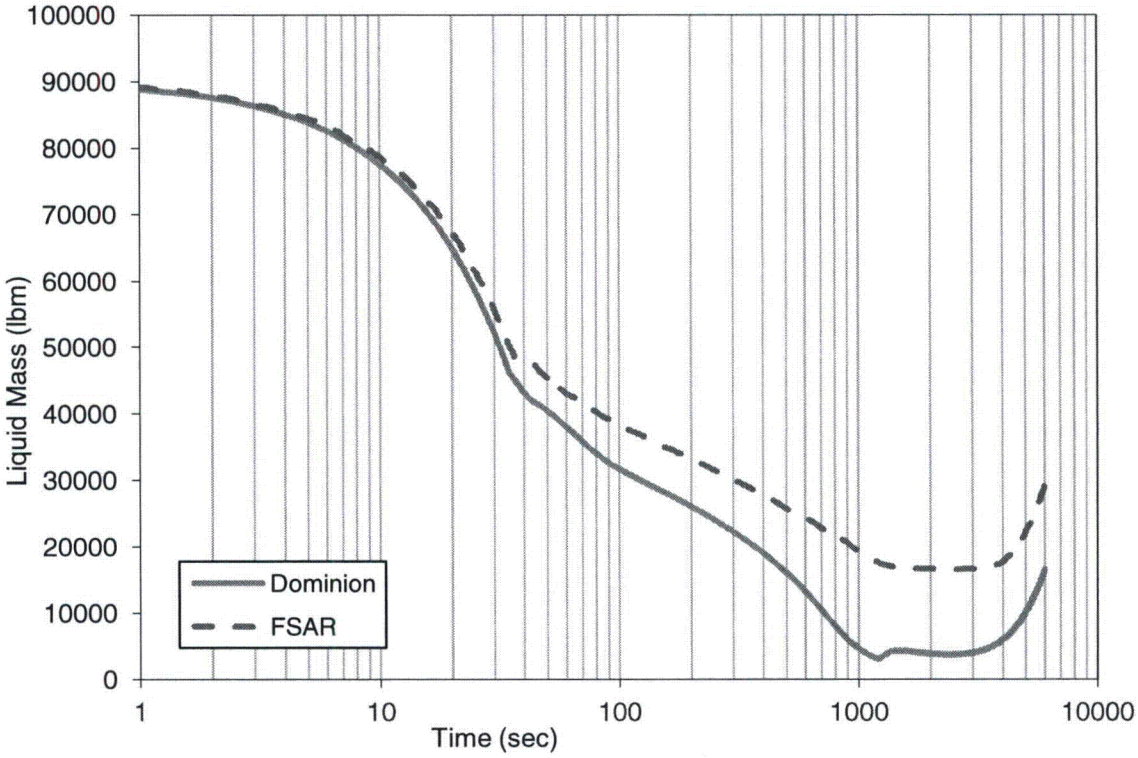


Figure 4.3-5 LONF – Pressurizer Pressure

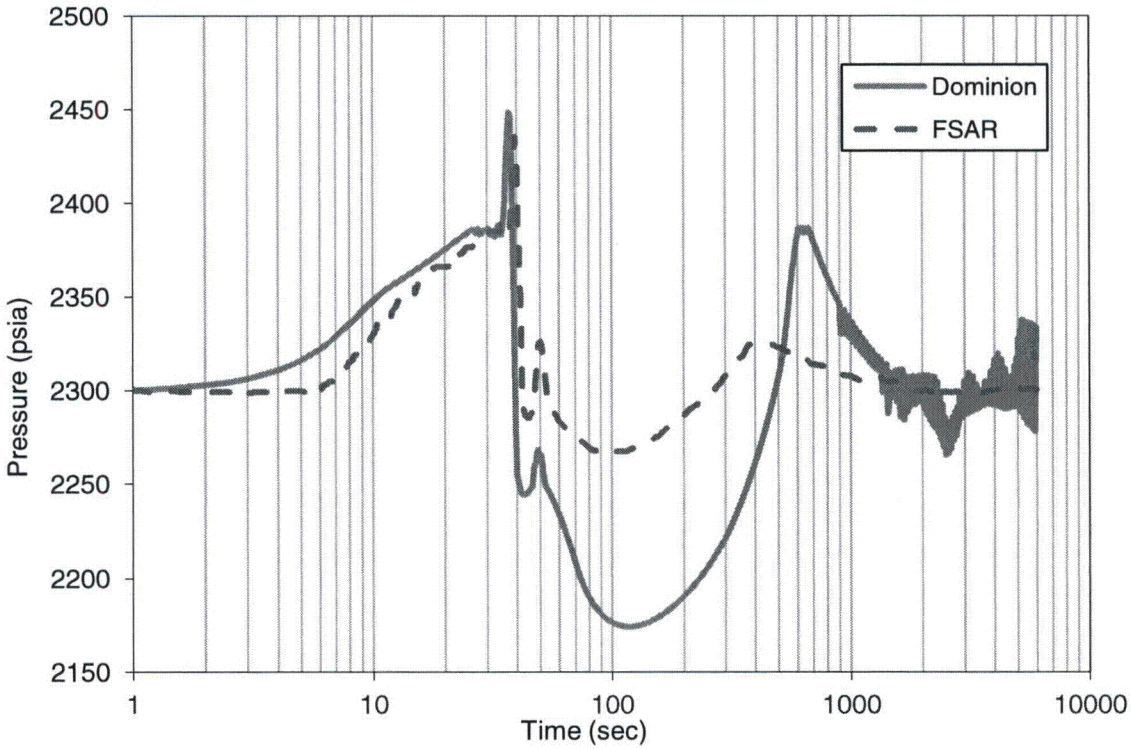


Figure 4.3-6 LONF – Pressurizer Water Volume

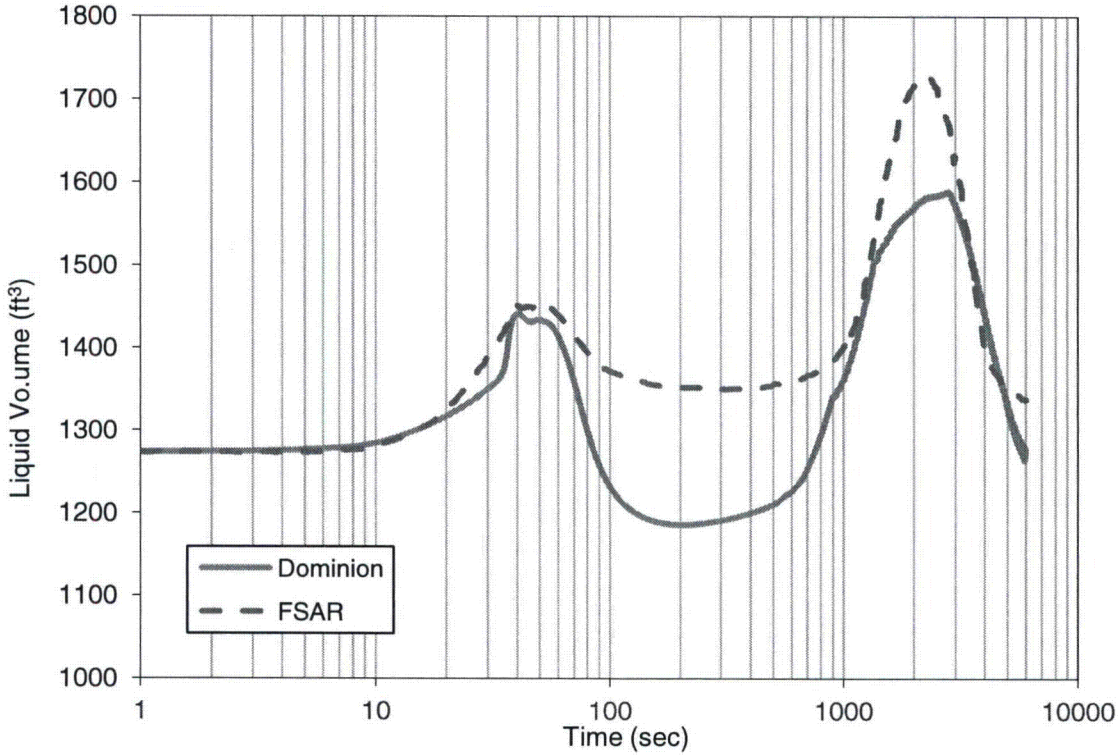
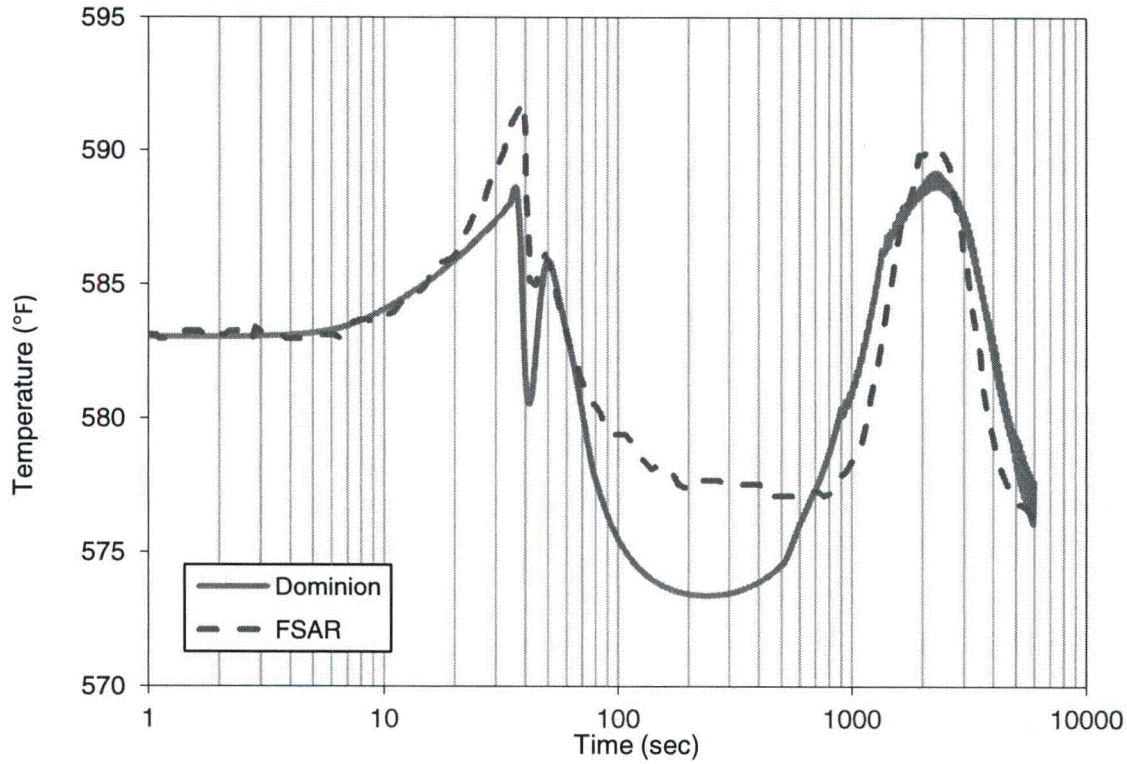


Figure 4.3-7 LONF – Loop Average Temperature



Summary - LONF

The Dominion analysis provides results that are similar to the FSAR analysis for the LONF event. The major differences result from the main steam safety relief valve modeling, which results in higher steam releases and a subsequent increase in heat transfer following the reactor trip. In addition, the steam generator nodalization and related heat transfer along with other modeling differences such as pressurizer spray also affect the transient response. These effects are cumulative resulting in a somewhat smaller long-term pressurizer insurge and higher pressurizer pressure peak compared to the FSAR results. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

4.4 Main Steam Line Break

The Main Steam Line Break (MSLB) event is a rupture in the main steam piping resulting in a rapid depressurization of the SG secondary and corresponding cooldown of the primary. The temperature reduction results in an insertion of positive reactivity with the potential for core power increase and DNBR violation.

The MSLB transient scenario presented here is modeled as an instantaneous, double-ended break at the nozzle of one steam generator from hot shutdown conditions with offsite power available. The input summary is provided in Table 4.4-1.

Table 4.4-1 MSLB Input Summary

Parameter	Value	Notes
Initial Conditions		
Core power (MW)	~1%	HZP
Pump power (MW)	0.0	
RCS Flow (gpm)	363,200	Thermal Design Flow
Vessel T_{AVG} (F)	557	HZP nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	28	HZP nominal
SG Level (%)	50	Nominal
Assumptions/Configuration		
Heat transfer option	Forced HT Map (note 1)	FSAR uses Forced + Free Convection HT Map
Main feedwater flow (% HFP value)	100	initiated at time 0 sec
Auxiliary feedwater flow rate (gpm)	Max	initiated at time 0 sec;
SG tube plugging (%)	0	Minimum value
Reactivity Parameters		
RWST Boron	Credited	FSAR does not credit boron from the SI system
Accumulator Boron	Not Credited	
Doppler Reactivity Feedback	Doppler Only Power defect, DTC model disabled	FSAR - Doppler power defect plus DTC included in moderator density feedback
Moderator Feedback	Moderator density feedback	Moderator density feedback

1 - Dominion method maximizes heat transfer coefficients for the faulted SG secondary side.

Results – MSLB with Offsite Power Available

The faulted loop steam flow and steam generator pressure responses shown in Figure 4.4-1 and Figure 4.4-3 match the FSAR data reasonably well with the steam flow and pressure in the Dominion model remaining somewhat higher than the FSAR data. This is partly caused by the slightly larger break junction area and the higher initial steam pressure for the Dominion model. In addition, the Dominion model uses conservatively high heat transfer coefficients in the faulted steam generator, which allow the faulted steam generator to pull heat faster from the primary side.

The Intact loop steam flow (Figure 4.2-2) shows a different response due to differences in the MSIV closure. In the Dominion model, the MSIVs close linearly over 10 seconds, while the FSAR model uses a delay of 10 seconds to conservatively increase RCS overcooling. The initial steam flow is higher for the Dominion case, decreasing below the FSAR value as the MSIVs close. The steam generator mass and pressure responses, shown in Figure 4.4-8 and Figure 4.4-4, reveals the differences in MSIV modeling with the Dominion model releasing somewhat less liquid inventory prior to valve closure.

For both the faulted and intact loops the main feedwater and auxiliary feedwater responses (Figure 4.4-5) give an excellent match to the FSAR data. The steam generator inventory (Figure 4.4-7) for the faulted loop depletes faster in the Dominion model than in the FSAR case due to the higher steaming rate from the faulted steam generator and the quicker and more conservative return to power.

The nuclear power and core heat flux responses (Figure 4.4-9 and Figure 4.4-10) calculated by the Dominion model peak higher and more quickly than the FSAR data. This response is contributed to by the greater cooling effects of the faulted steam generator on the RCS due to its higher steam production. The quicker return to power is also a result of differences in the nodalization and mixing at the core inlet and outlet between the Dominion model and the FSAR model. The return to power also drops off approximately 50 seconds sooner in the Dominion model. This is also caused by the higher steam rate in the Dominion model which causes the faulted steam generator to dry out sooner. The power response for both models is not affected by the delivery of boron to the RCS. This is because the FSAR model does not credit boron and in the Dominion model boron does not reach the RCS from the SI system until after the termination of the transient. Overall, the Dominion model results in a more conservative response for core heat flux and power.

The pressurizer pressure response (Figure 4.4-12) agrees very well with the pressure predicted by the FSAR model for the first 50 seconds of the transient, after which the FSAR data falls approximately 100 psi lower than the pressure calculated by the Dominion model. This difference is a result of using only a single upper head leakage path in the Dominion model. The upper head leakage is taken from the three intact loops and does not credit any flow from the lower temperature, faulted loop. This causes the upper head temperature to remain slightly higher than would actually be the case, which allows a vapor bubble in the upper head to form sooner and become larger. This in turn prevents the RCS pressure from falling lower.

The pressurizer drains at approximately the same rate for the Dominion model and FSAR models (Figure 4.4-13). However, for the Dominion model the pressurizer begins to refill approximately 100 seconds sooner. The quicker refilling is a result of the higher and quicker return to power which causes the RCS temperature to rise sooner in the Dominion model. This causes the RCS fluid inventory to expand which results in the pressurizer refilling sooner in the Dominion model than is seen from the FSAR model.

Table 4.4-2 MSLB with Offsite Power Results

Event	Time (sec) From Start of Transient	
	Dominion	FSAR
Steam Line Ruptures	0	0
Manual Reactor Trip	0	0
Increase MFW to 100% of Nominal HFP Value	0	0
Initiate Maximum AFW To Faulted Steam Generator	0	0
Main Feedwater Isolation	7.5	8.2
MSIVs Closed	12.5	13.5
Pressurizer Empty	15.5	20.5
Criticality Attained	34	28
Safety Injection Flow Initiation	40.5	72.8
Faulted Steam Generator Dries Out	297	~350

Figure 4.4-1 MSLB – Faulted Loop Steam Flow

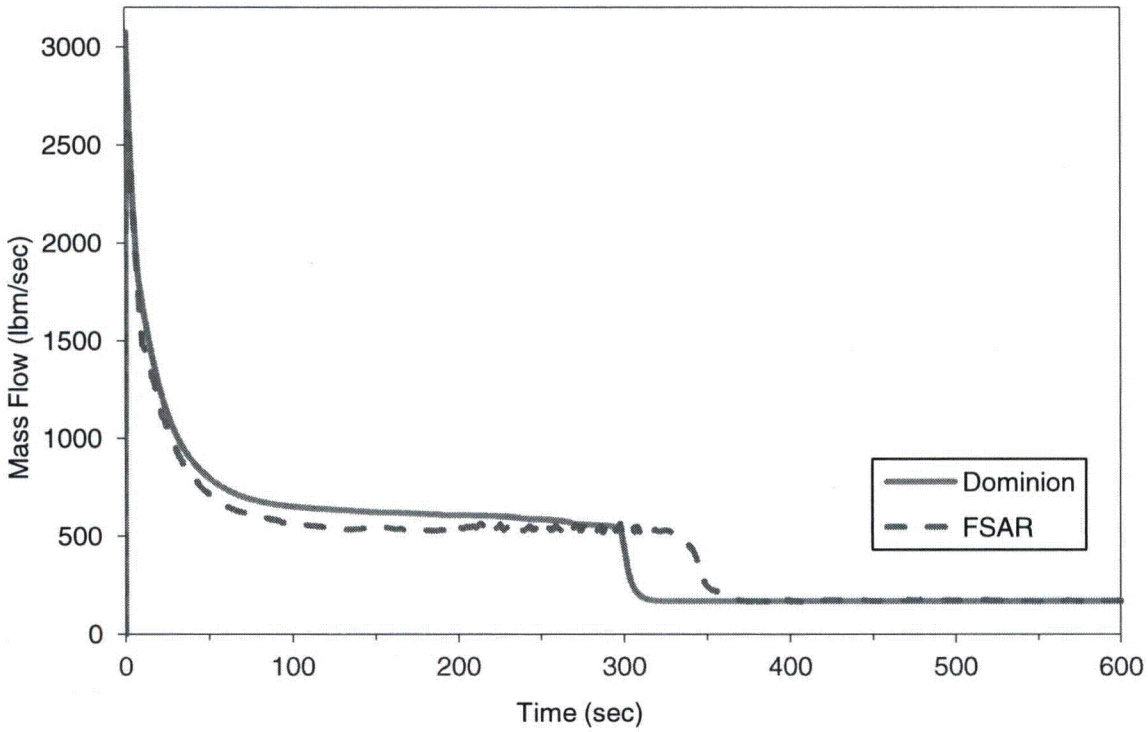


Figure 4.4-2 MSLB – Intact Loop Steam Flow

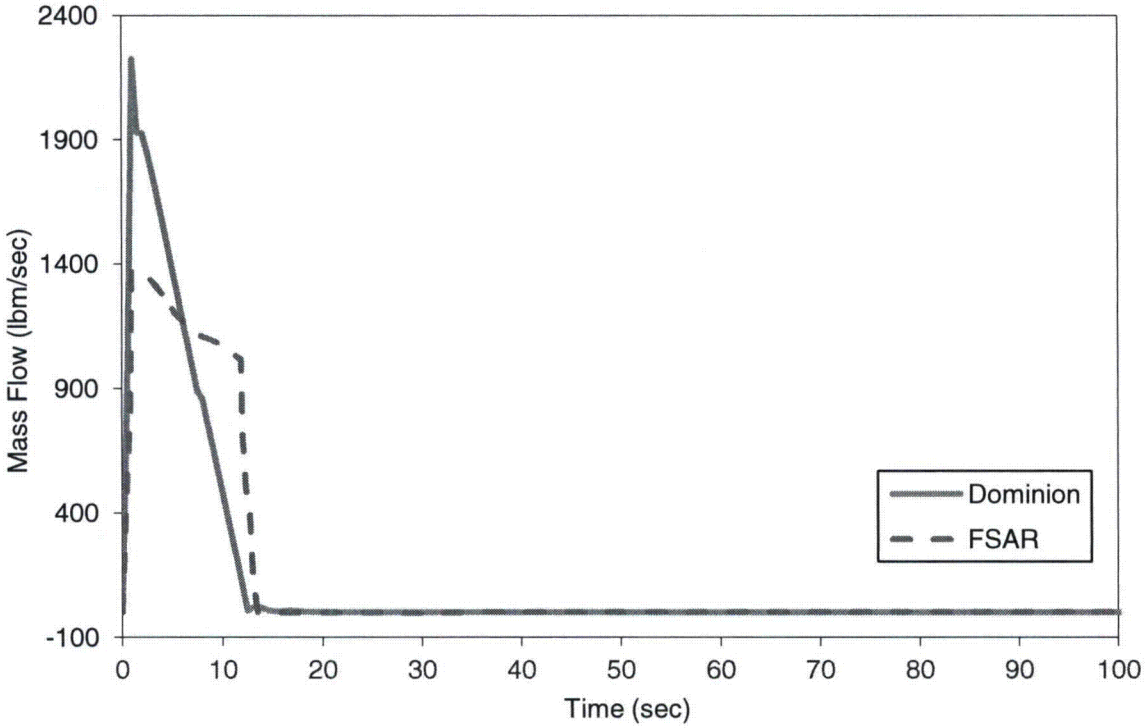


Figure 4.4-3 MSLB – Faulted Loop Steam Generator Pressure

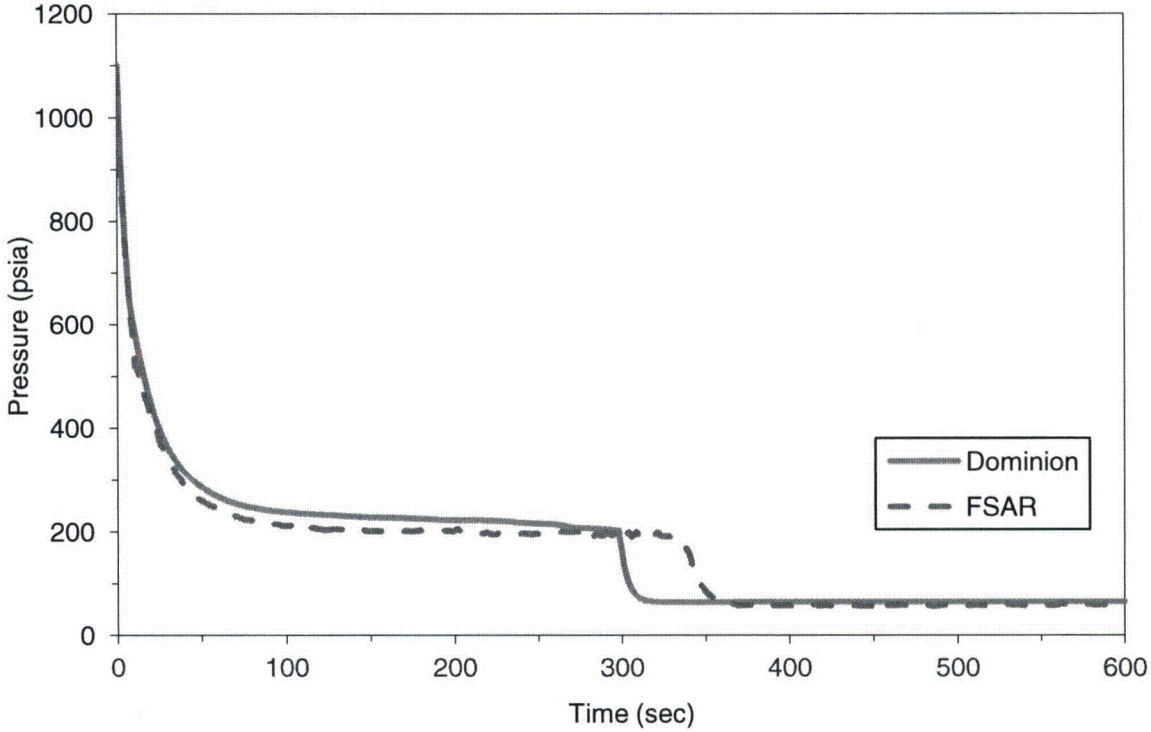


Figure 4.4-4 MSLB –Intact Loop Steam Generator Pressure

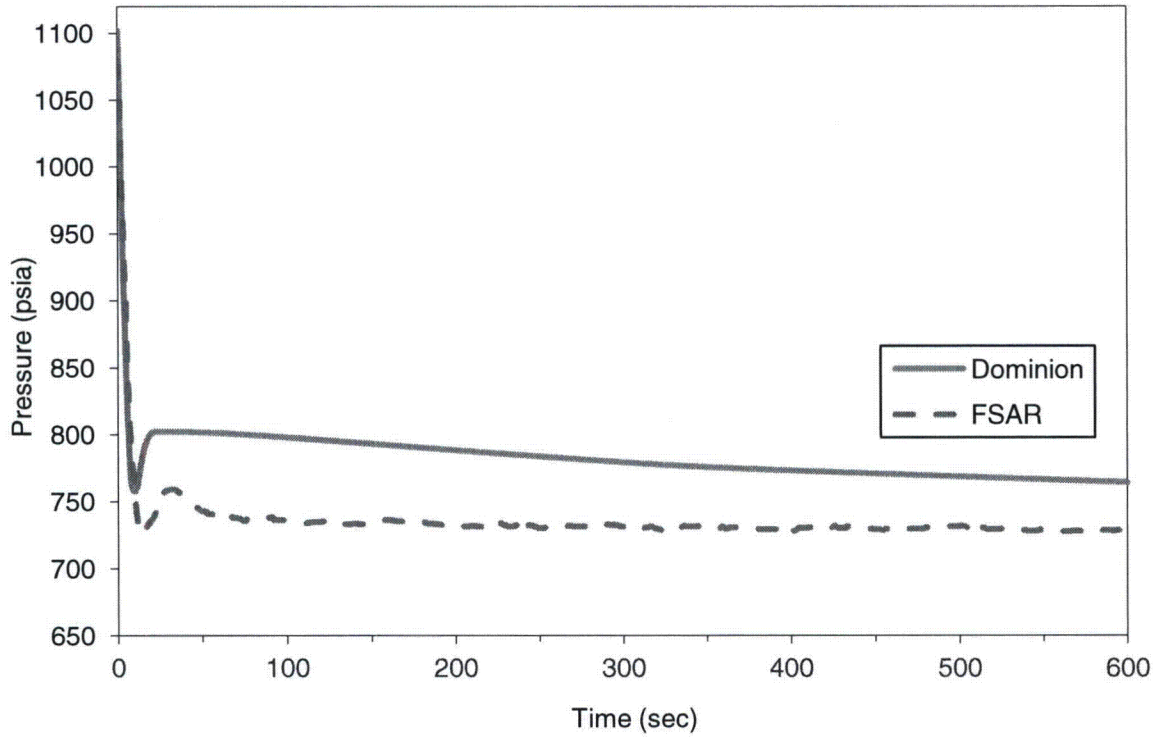


Figure 4.4-5 MSLB – Faulted Loop Total Feedwater Flow

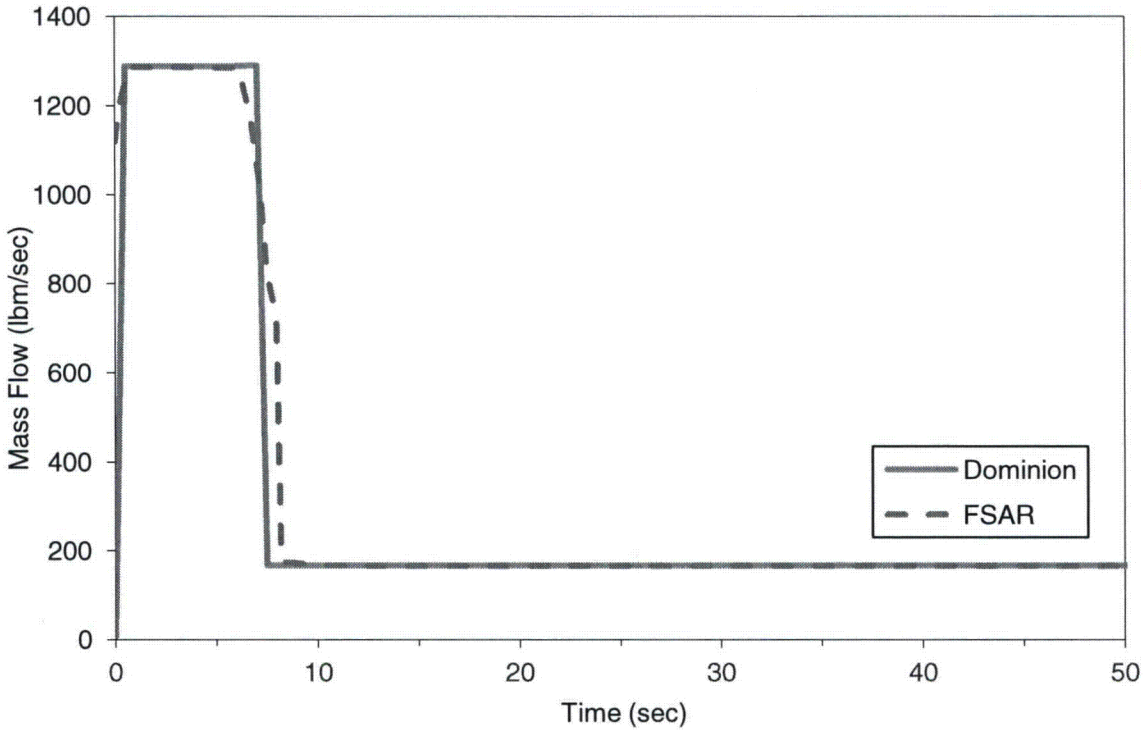


Figure 4.4-6 MSLB – Intact Loop Total Feedwater Flow

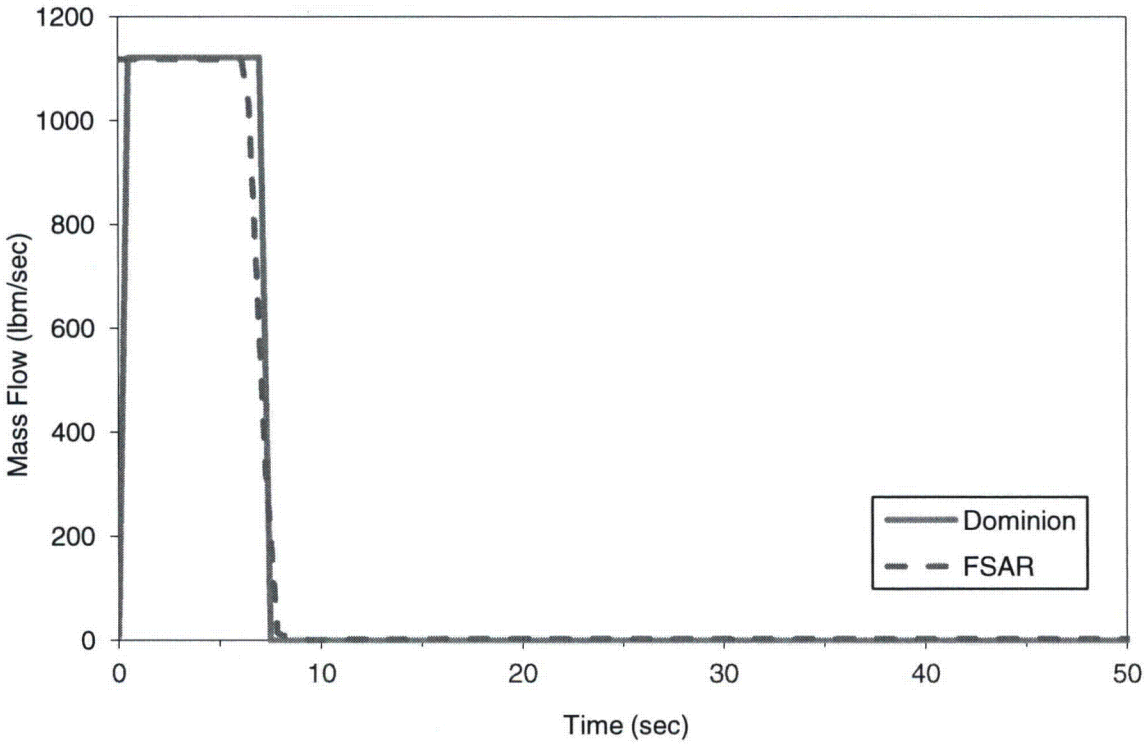


Figure 4.4-7 MSLB – Faulted Loop SG Liquid Mass

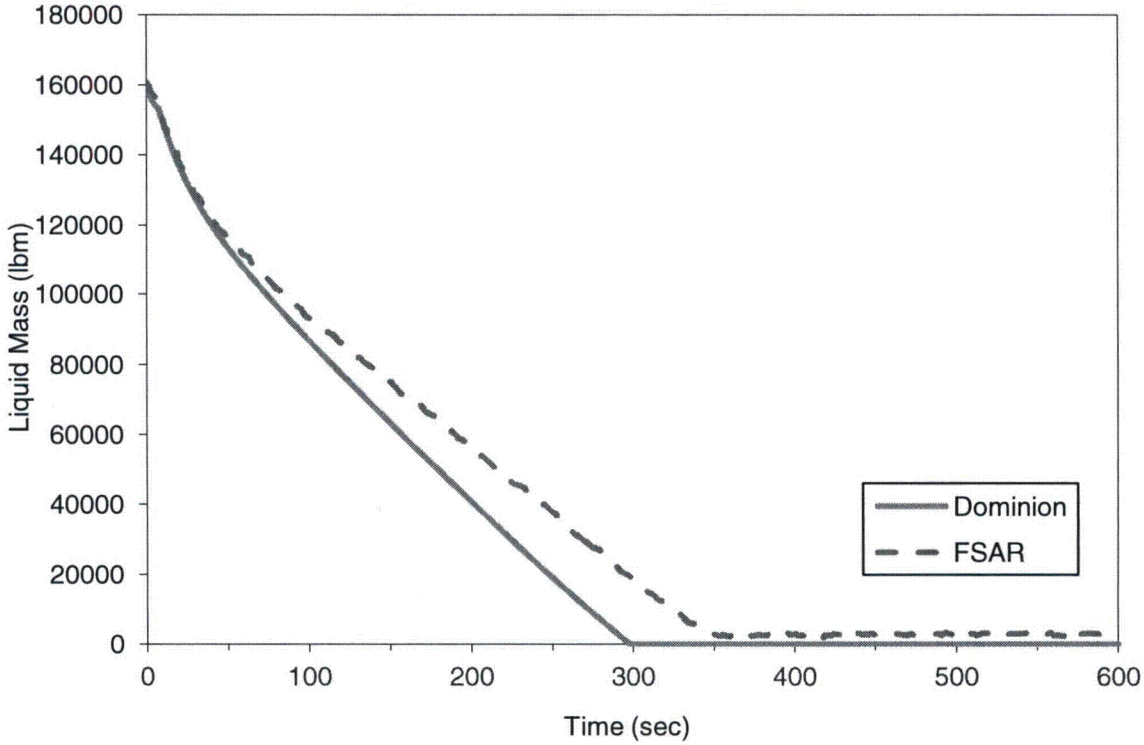


Figure 4.4-8 MSLB – Intact Loop SG Liquid Mass

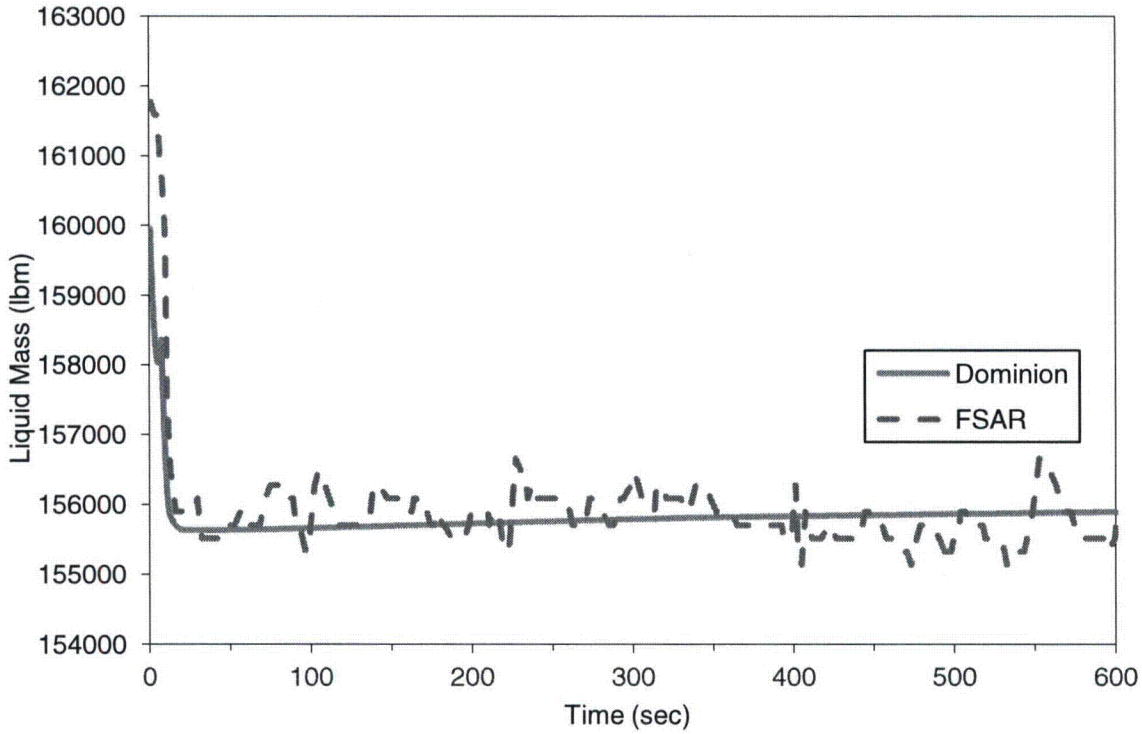


Figure 4.4-9 MSLB – Normalized Core Power

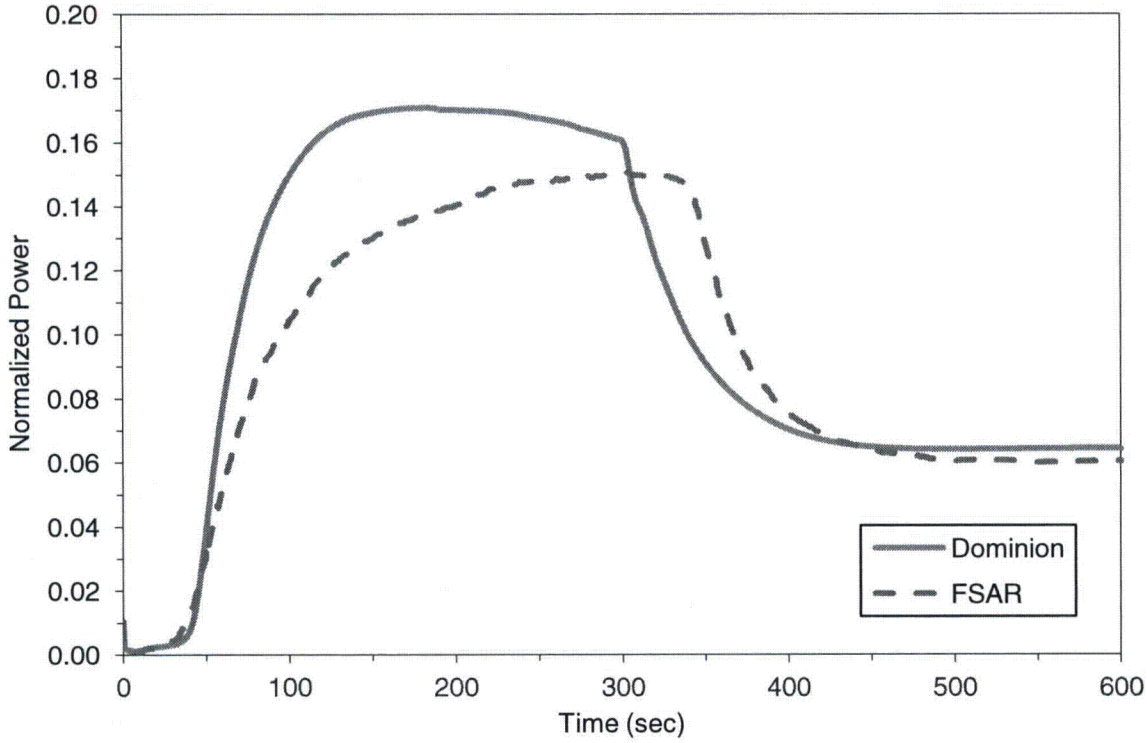


Figure 4.4-10 MSLB – Normalized Core Heat Flux

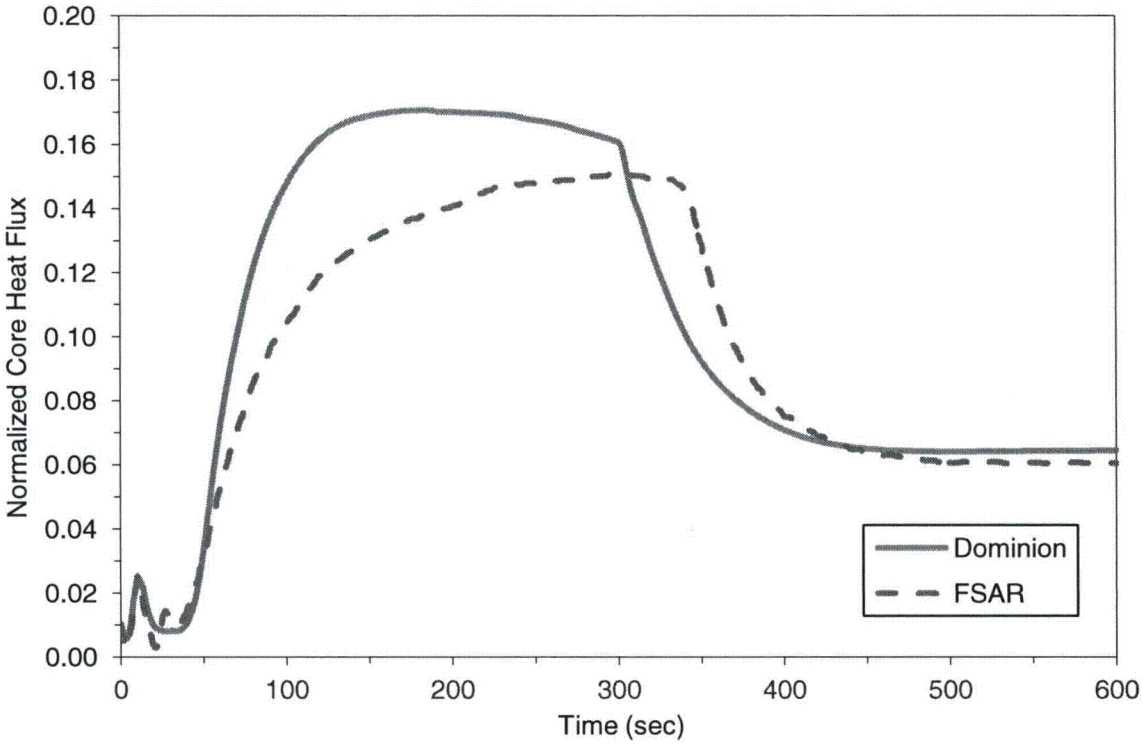


Figure 4.4-11 MSLB – Reactivity Feedback

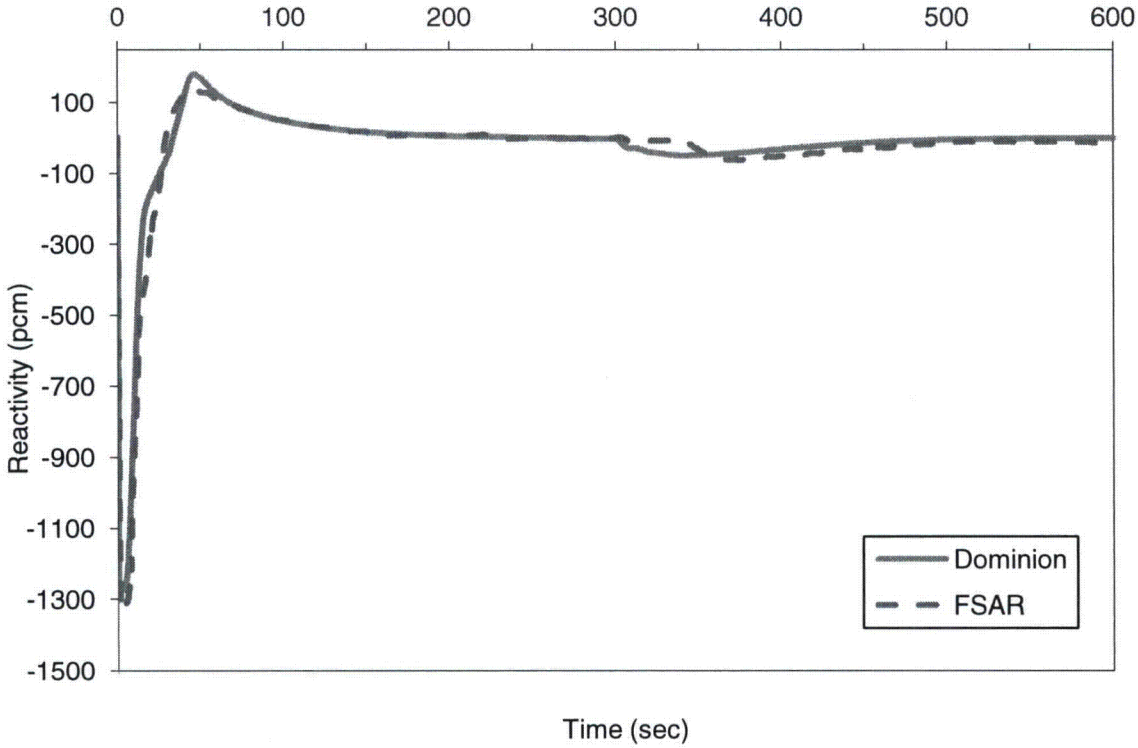


Figure 4.4-12 MSLB – Pressurizer Pressure

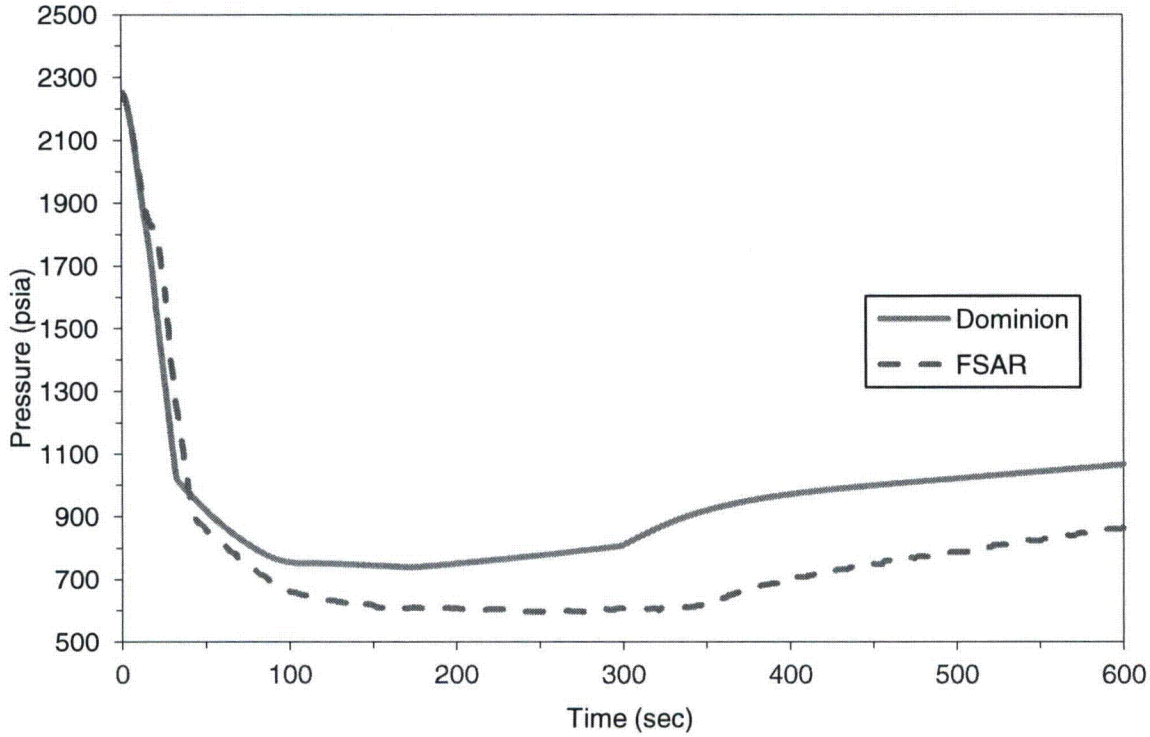


Figure 4.4-13 MSLB – Pressurizer Liquid Volume

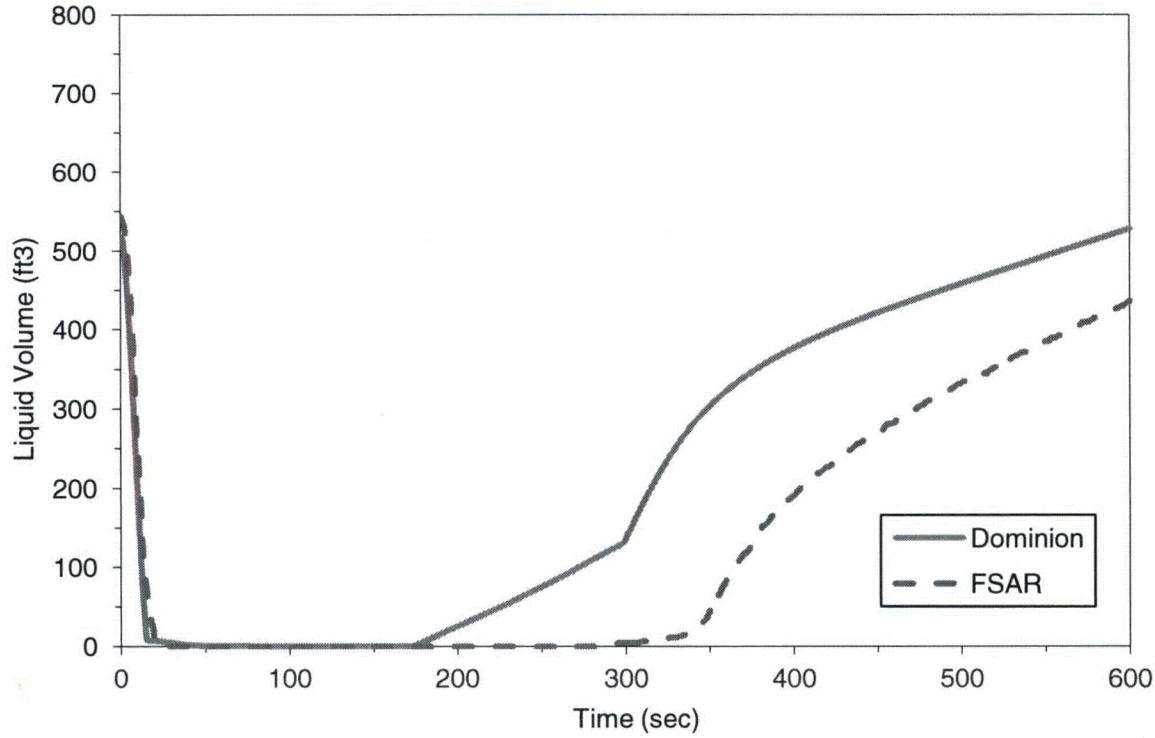


Figure 4.4-14 MSLB – Faulted Loop Vessel Inlet Temperature

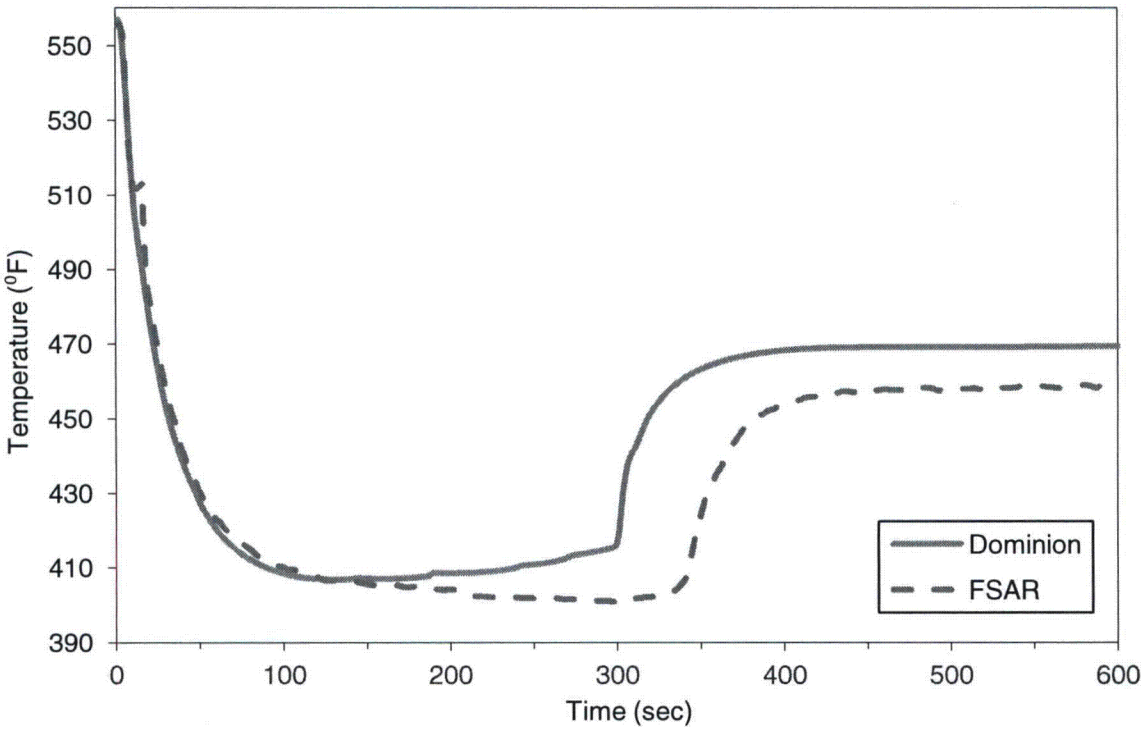
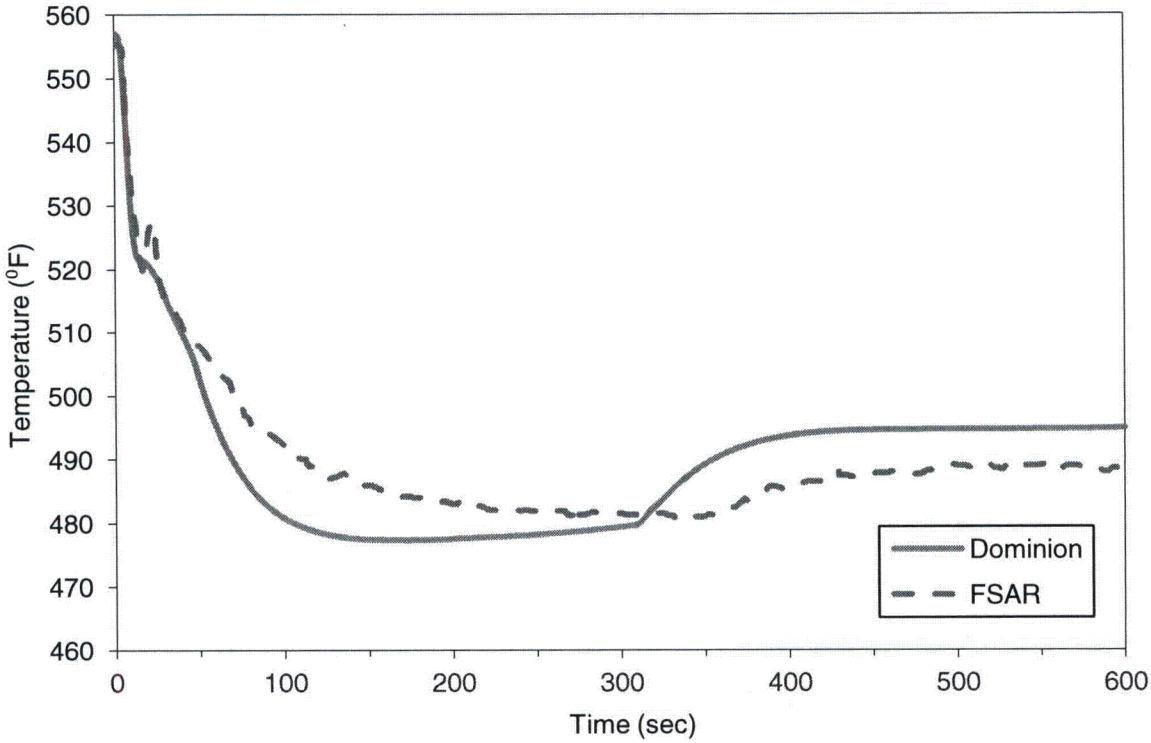


Figure 4.4-15 MSLB – Intact Loop Vessel Inlet Temperature



Summary - MSLB

This section presents a comparison of a RETRAN-3D Main Steam Line Break transient calculation with the Millstone model using the Dominion RETRAN transient analysis methods (Reference 1) compared to the FSAR results. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR. The key observations from these comparisons are that:

1) The peak power and heat flux reached with the Dominion methods is higher than the FSAR result.

2) Core and steam generator nodalization affects asymmetric transients such as a MSLB.

4.5 Control Rod Bank Withdrawal at Power

The Control Rod Bank Withdrawal at Power (RWAP) event is defined as the inadvertent addition of core reactivity caused by the withdrawal of rod control cluster assembly (RCCA) banks when the core is above no load conditions. The RCCA bank withdrawal results in positive reactivity insertion, a subsequent increase in core nuclear power, and a corresponding rise in the core heat flux. The RWAP event described here is terminated by the Reactor Protection System on a high neutron flux trip or the Overtemperature ΔT trip (OTΔT), consistent with the FSAR analyses.

The RWAP event is simulated by modeling a constant rate of reactivity insertion starting at time zero and continuing until a reactor trip occurs. The Dominion analysis involves two different reactivity insertion rates, 1 pcm/sec and 100 pcm/sec that match the reactivity insertion rates presented plots in the FSAR. Most of the input parameters are the same as those used in the FSAR Chapter 15 analyses. Where differences from the FSAR inputs exist, they are indicated in the Notes column.

Table 4.5-1 RWAP Input Summary

Parameter	Value	Notes
Initial Conditions		
NSSS Power (MW)	3650	Nominal
RCS Flow (gpm)	379,200	Minimum Measured Flow
Vessel T_{AVG} (F)	589.5	Nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	64	Nominal
SG Level (%)	50	Nominal
Initial Fuel Temperature	Minimum	Uses current FSAR analysis conductivity adjustments
Assumptions/Configuration		
Reactor trip	-	High neutron flux or OTΔT
Automatic rod control	-	Not credited
Pressurizer level control	-	Not credited
Pressurizer heaters	-	Not credited
Pressurizer sprays, PORVs	-	Active
SG tube plugging (%)	10	Max value
Reactivity Parameters		
Doppler Reactivity Feedback	Least Negative	
Moderator Feedback	Most Positive	

Results – RWAP 1 pcm/sec Case

Figure 4.5-1 shows the core power response. The core power rate of increase for the Dominion model is greater than the FSAR data. This leads to the Dominion modeling tripping on high neutron flux at about 73 seconds. The FSAR case rises in power at a slower rate, which trips on an OTΔT signal at about 93 seconds. The difference in reactor trip mechanisms between the Dominion and FSAR cases is reasonable considering the breakpoint for switching between OTΔT and high flux as shown in FSAR Figure 15.4-10. The pressure response also affects the OTΔT setpoint such that the lower FSAR pressure (see below) will act to reduce the setpoint.

The pressurizer pressure response is shown in Figure 4.5-2. For the Dominion model, the pressure rises faster than the FSAR result. At about 42 seconds, the Dominion model reaches the pressurizer relief valve setpoint and begins to cycle. The FSAR more slowly increases in pressure and reaches the relief valve set point around 10 seconds prior to the reactor trip. The difference in pressure response can be attributed to the difference in core power response as each cases pressure response initially mimics the energy generated by the core as seen in Figure 4.5-1 and the higher spray flow assumed in the FSAR analysis, which acts to suppress pressure. The same can be seen in the vessel average temperature response where the FSAR case lags the Dominion response, yet reaches a temperature approximately 5 degrees higher than the Dominion case due to the FSAR case tripping later in the transient.

Table 4.5-2 RWAP 1 pcm/sec Time Sequence of Events

Event	Time (seconds)	
	Dominion	**FSAR**
Reactivity Insertion at 3 pcm/sec	0.0	0.0
Reactor Trip Signal Initiated	73.64*	93.63**

* *Trip on high neutron flux*

** *Trip on OTΔT*

Results – RWAP 100 pcm/sec Case

Figure 4.5-4 shows the core power response for the current FSAR analysis and the Dominion model. The Dominion model trips on a high neutron flux at about 1.68 seconds, compared to about 1.79 seconds for the current FSAR analysis. The 100 pcm/sec transient is a fast transient and the time period before the reactor trip is so brief that any differences in fuel pin heat transfer modeling assumptions have little impact on Doppler reactivity

feedback. Overall, the Dominion model peaks at a higher, thus more conservative power level.

The pressurizer pressure response is shown in Figure 4.2-5. The Dominion model matches very well with the FSAR analysis. The main difference being that the Dominion model peaks at a higher pressure than the FSAR analysis. This correlates with the power response shown in Figure 4.2-4 where the Dominion model peaks at a higher overall nuclear power. Figure 4.2-6 shows the vessel average temperature. For the 100 pcm/sec case the Dominion model matches very closely with the FSAR analysis

Table 4.5-3 RWAP 100 pcm/sec Time Sequence of Events

Event	Time (seconds)	
	Dominion	**FSAR**
Reactivity Insertion at 1 pcm/sec	0.0	0.0
Reactor Trip Signal Initiated	1.68*	1.79*

* *Trip on high neutron flux*

Figure 4.5-1 RWAP – 1 pcm/sec Nuclear Power

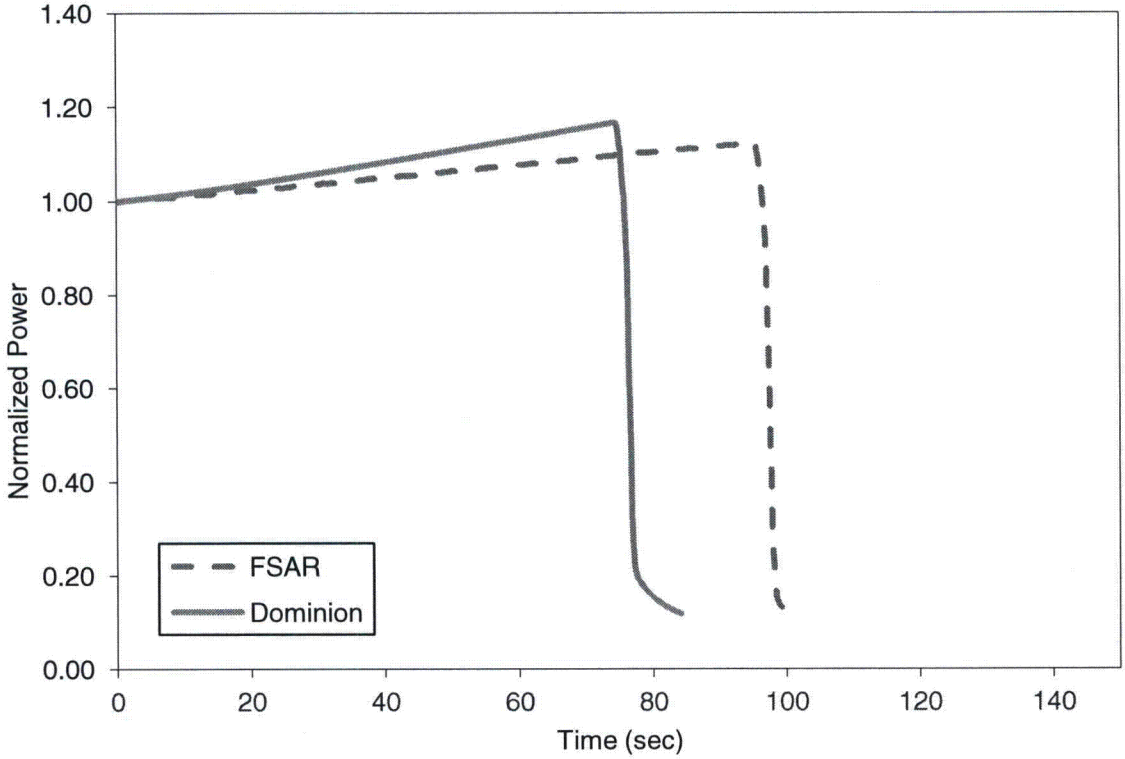


Figure 4.5-2 RWAP – 1 pcm/sec Pressurizer Pressure

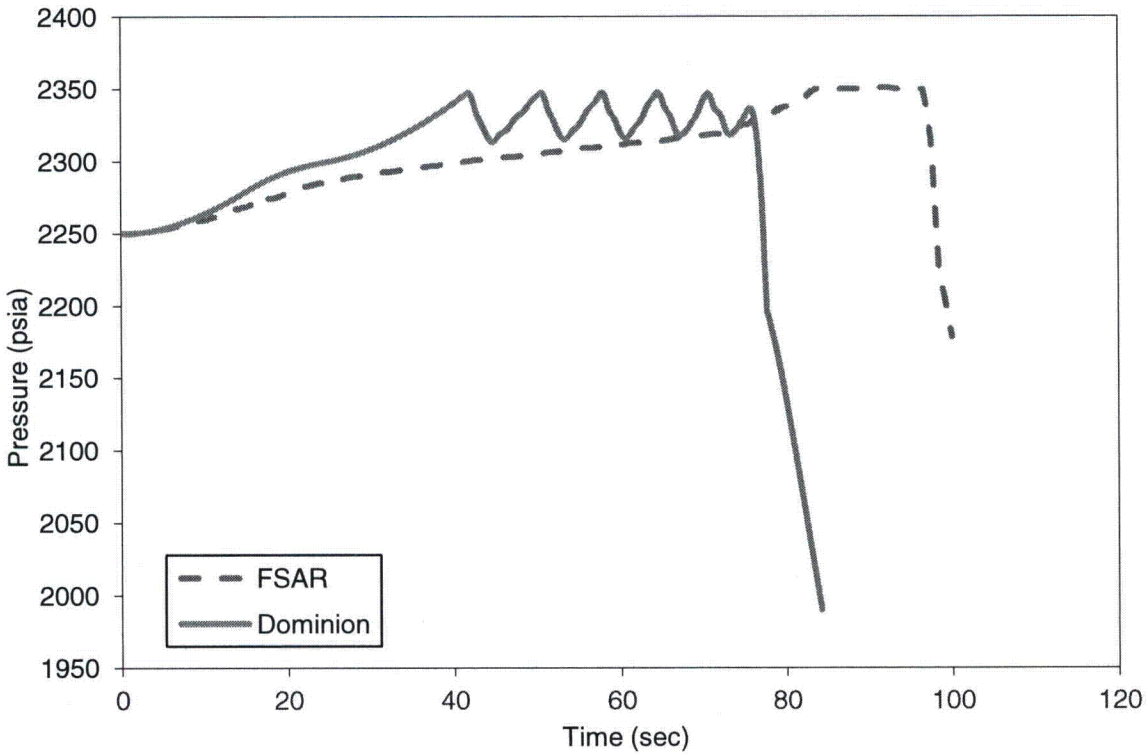


Figure 4.5-3 RWAP – 1 pcm/sec Vessel Average Temperature

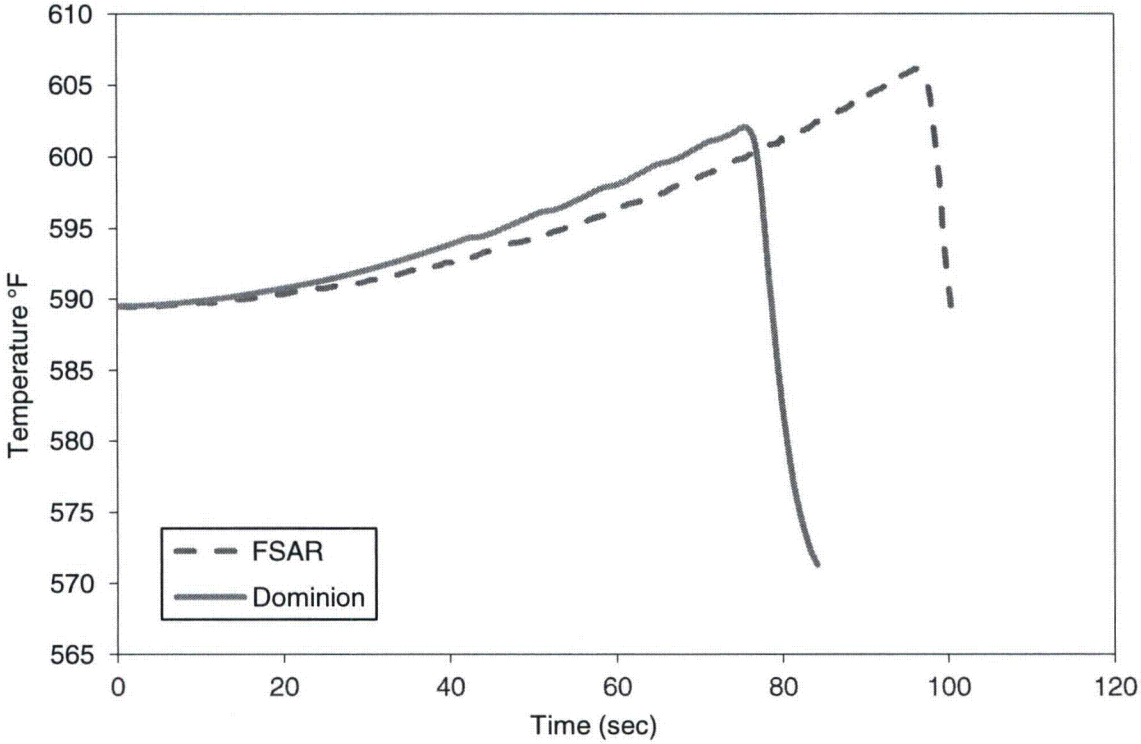


Figure 4.5-4 RWAP – 100 pcm/sec Nuclear Power

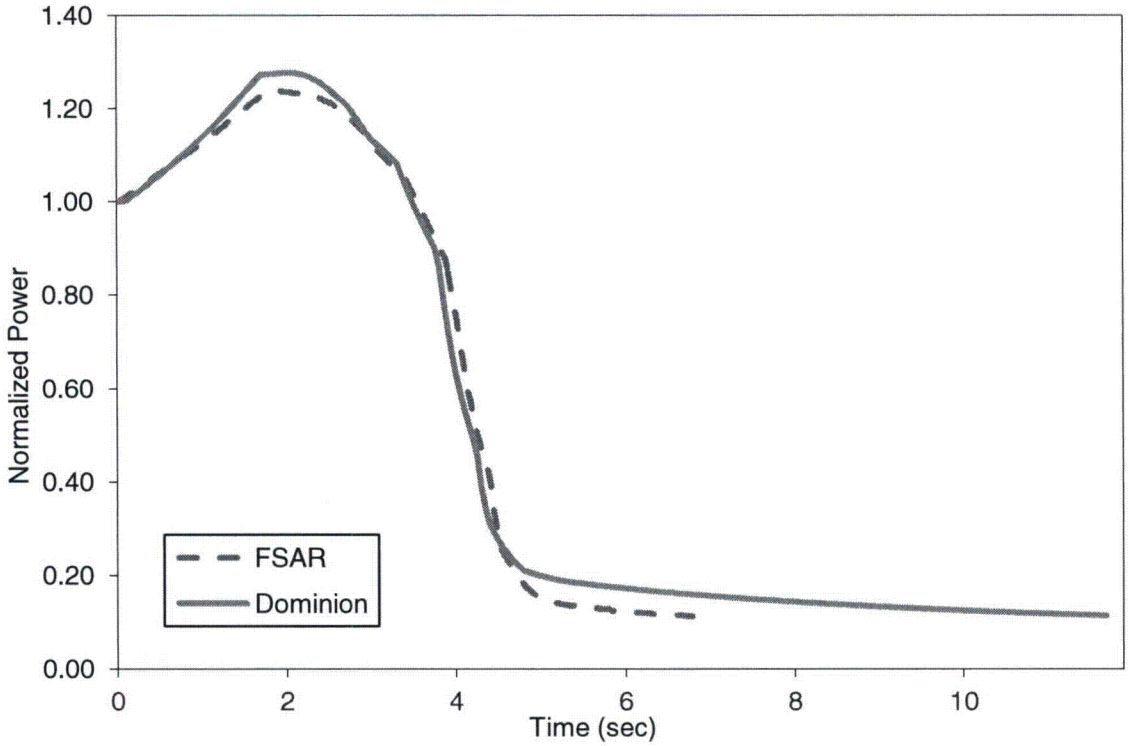


Figure 4.5-5 RWAP – 100 pcm/sec Pressurizer Pressure

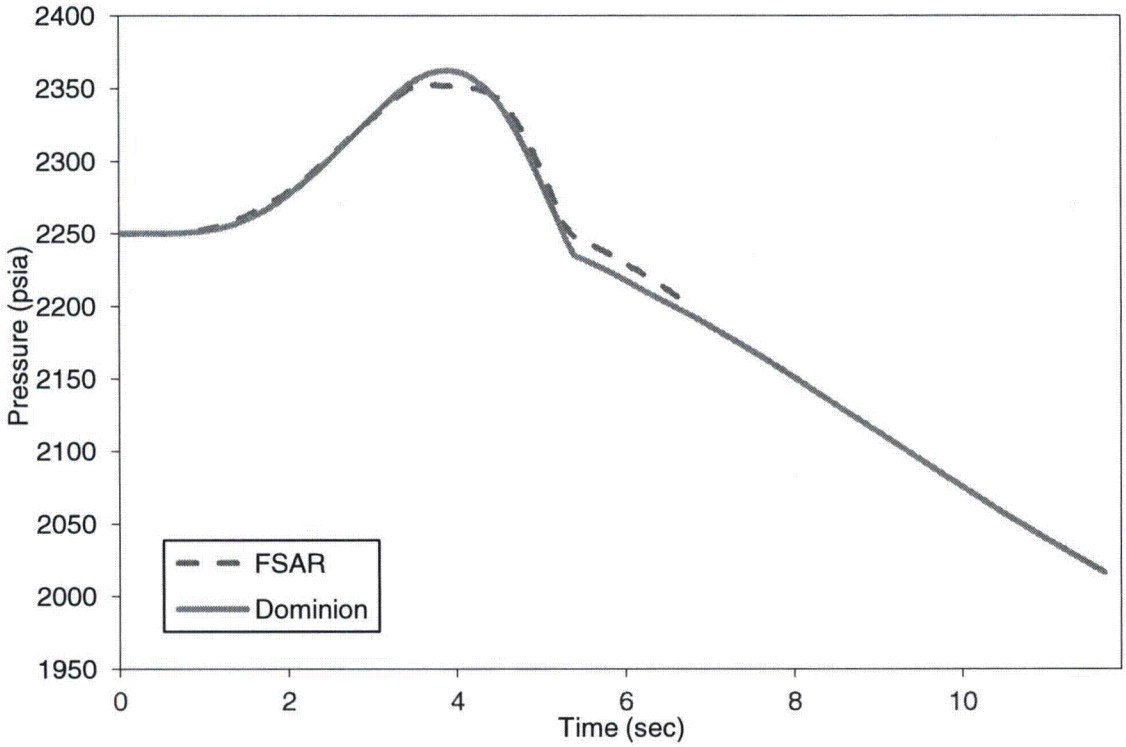
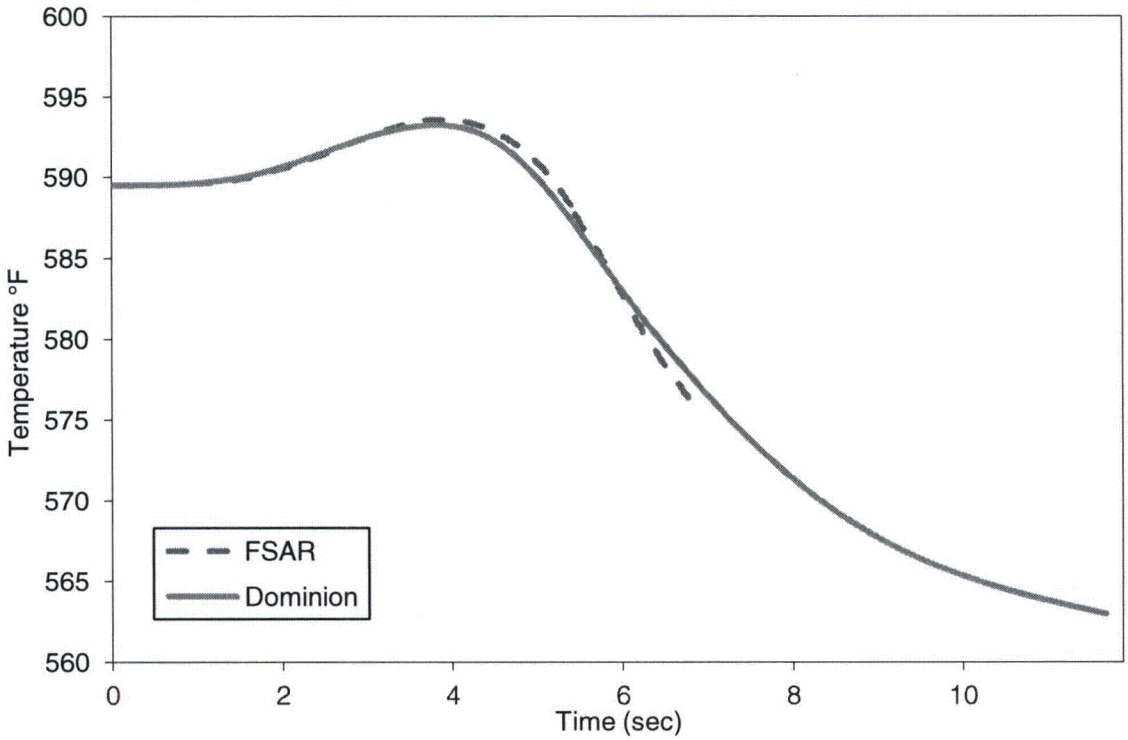


Figure 4.5-6 RWAP – 100 pcm/sec Vessel Average Temperature



Summary - RWAP

The Dominion Millstone model provides results that are similar to the FSAR analysis for the RWAP event. At higher insertion rates, the results match very well. At lower insertion rates, the power increases at a greater rate in the Dominion model than the FSAR model. However, the temperature increases to a higher peak in the FSAR analysis. The Dominion MPS3 analysis is presented for benchmark comparison, and does not replace the existing AOR.

5.0 Conclusions

This attachment presents benchmarking transient analyses performed with the MPS3 RETRAN model developed in accordance with VEP-FRD-41-P-A. These analysis results are compared with current Millstone FSAR results. The following conclusions are drawn based on these analyses.

1) It is demonstrated that the Dominion RETRAN-3D model and analysis methods can predict the response of transient events with results that compare well to FSAR results.

2) Where there are differences between the Dominion results and the FSAR results, they are understood based on differences in noding, inputs, or other modeling assumptions.

3) The Dominion Millstone RETRAN-3D model is consistent with current Dominion methods (Reference 1). These methods have been applied extensively for Surry and North Anna licensing, engineering and plant support analyses.

4) The RETRAN comparison analyses satisfy the applicability assessment criteria and provide further validation of the conclusion that Dominion's RETRAN analysis methods are applicable to Millstone and can be applied to Millstone licensing analysis for reload core design and safety analysis.

6.0 References

1) Topical Report, VEP-FRD-41-P-A, Rev. 0.2, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," March 2015.

2) Topical Report, VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient," December 1984.

ATTACHMENT 6

DEVELOPMENT OF STATISTICAL DESIGN LIMITS

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

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1.0 Introduction

This attachment provides the plant-specific application of the Statistical DNBR Methodology for Millstone Power Station Unit 3 (MPS3) cores containing the Westinghouse 17x17 Robust Fuel Assembly – 2 (RFA-2) fuel product. The Westinghouse RFA-2 fuel product contains modified mid-grids and modified intermediate flow mixer grids (IFMs). Specifically, this attachment supports the application of U.S. Nuclear Regulatory Commission (USNRC) approved Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology" (Reference 1) to MPS3, where DNBR stands for Departure from Nucleate Boiling Ratio. It provides the technical basis and documentation required by the USNRC to evaluate the plant specific application of the VEP-NE-2-A methodology to MPS3. This application employs the VIPRE-D thermal-hydraulic computer code (DOM-NAF-2-P-A, Reference 2) with the Westinghouse WRB-2M, ABB-NV and WLOP Critical Heat Flux (CHF) correlations (VIPRE-D/WRB-2M/ABB-NV/WLOP code/correlation combination) for the thermal-hydraulic analysis of the Westinghouse 17x17 RFA-2 fuel product at MPS3. In particular, Dominion requests the review and approval of the Statistical Design Limits (SDLs) documented herein as per 10 CFR 50.59(c)(2)(vii) they constitute a Design Basis Limit for a Fission Product Barrier (DBLFPB).

Dominion is seeking approval for the inclusion of Topical Report VEP-NE-2-A (Reference 1) and Fleet Report DOM-NAF-2-P-A, Appendix C and D, (Reference 2) to the Technical Specification (T.S.) 6.9.1.6.b list of USNRC-approved methodologies used to determine core operating limits (i.e., the reference list of the MPS3 Core Operating Limits Report (COLR)). This would allow Dominion the use of the VIPRE-D/WRB-2M/ABB-NV/WLOP code/correlation combinations to perform licensing calculations for the Westinghouse 17x17 RFA-2 fuel in MPS3 cores, using the deterministic design limits (DDLs) qualified in Appendices C and D of Fleet Report DOM-NAF-2-P-A, and the SDLs identified herein.

With these approvals, Dominion will be licensed to perform in-house Departure from Nucleate Boiling (DNB) analyses for the intended uses described in Fleet Report DOM-NAF-2-P-A to support MPS3 with the Westinghouse 17x17 RFA-2 fuel product.

2. Background

2.1 Fleet Report DOM-NAF-2-P-A

The computer code VIPRE (Versatile Internals and Components Program for Reactors - EPRI) was developed for the Electric Power Research Institute (EPRI) by Battelle Pacific Northwest Laboratories to perform detailed thermal-hydraulic analyses to predict CHF and DNBR of reactor cores. VIPRE-01 was approved by the U.S. Nuclear Regulatory Commission (USNRC) in References 3 and 4 for referencing in licensing applications. VIPRE-D is the Dominion version of the VIPRE computer code based upon VIPRE-01. VIPRE-D was developed to fit the specific needs of Dominion's nuclear plants and fuel products by adding vendor specific CHF correlations and customizing its input and output. However, Dominion has not made any modifications to the NRC-approved constitutive models and algorithms contained in VIPRE-01.

Dominion's Fleet Report DOM-NAF-2-P-A (including Appendix C, which describes the verification and qualification of the WRB-2M CHF correlation and Appendix D, which describes the verification and qualification of the ABB-NV and WLOP CHF correlations) (Reference 2) has been reviewed and approved by the USNRC in References 5 and 6, respectively. Fleet Report DOM-NAF-2-P-A provides the necessary documentation to describe Dominion's use of the VIPRE-D code, including modeling and qualification for Pressurized Water Reactors (PWR) thermal-hydraulic design and demonstrated that the VIPRE-D methodology is appropriate for PWR licensing applications. Appendix C qualifies the WRB-2M CHF correlation with the VIPRE-D code and lists the deterministic code/correlation DNBR limits. The WRB-2M CHF correlation is applicable for the DNBR evaluation of the Westinghouse 17x17 RFA-2 fuel product [refer to Section 2.3 for further discussion]. Appendix D qualifies the ABB-NV and WLOP CHF correlations with the VIPRE-D code and lists the deterministic code/correlation DNBR limits. The ABB-NV and WLOP CHF correlations are applicable for the DNBR evaluation of the Westinghouse 17x17 RFA-2 fuel product.

In addition, Section 2.1 of Fleet Report DOM-NAF-2-P-A listed the information to be provided to the USNRC by Dominion for the review and approval of any plant specific application of the VIPRE-D code:

1) Technical Specifications change request (TSCR) to add DOM-NAF-2-P-A and relevant Appendixes to the plant's COLR list.
2) Statistical Design Limit(s) for the relevant code/correlation(s) (Section 4.1)
3) Any technical specification changes related to thermal over-temperature ΔT ($OT\Delta T$), over-power ΔT ($OP\Delta T$), axial power distribution ($F\Delta I$), enthalpy rise factor ($F\Delta H$) or other reactor protection function, as well as revised Reactor Core Safety Limits (Section 4.5).
4) List of FSAR transients for which the code/correlations will be applied (Section 3.9).

This attachment provides the technical basis (items 1 through 4 above) for the USNRC review and approval of the implementation of Fleet Report DOM-NAF-2-P-A including Appendices C and D to analyze the 17x17 RFA-2 fuel at MPS3 with the VIPRE-D/WRB-2M/ABB-NV/WLOP code/correlation combinations, as well as the SDL obtained by this implementation (DOM-NAF-2-P-A Condition 2). This attachment also documents that the existing Reactor Core Safety Limits and protection functions ($OT\Delta T$, $OP\Delta T$, $F\Delta I$, etc.) do not require revision as a consequence of this implementation (DOM-NAF-2-

P-A Condition 3). The list of FSAR transients for which the code/correlation pair will be applied is also included herein (DOM-NAF-2-P-A Condition 4).

2.2 Topical Report VEP-NE-2-A

In 1985, Virginia Power (Dominion) submitted to the USNRC Topical Report VEP-NE-2-A (Reference 1) describing a proposed methodology for the statistical treatment of key uncertainties in core thermal-hydraulic DNBR analysis. The methodology provided DNBR margin through the use of statistical rather than deterministic uncertainty treatment. The methodology was reviewed and approved by the USNRC in May 1987, and the Safety Evaluation Report (SER) provided by the USNRC listed the following conditions for its use (Reference 7):

1) The selection and justification of the Nominal Statepoints used to perform the plant specific implementation must be included in the submittal (Sections 3.6 and 3.8).
2) Justification of the distribution, mean and standard deviation for all the statistically treated parameters must be included in the submittal (Section 3.2).
3) Justification of the value of model uncertainty must be included in the plant specific submittal (Section 3.4).
4) For the relevant CHF correlations, justification of the 95/95 DNBR limit and the normality of the M/P distribution, its mean and standard deviation must be included in the submission, unless there is an approved Topical Report documenting these (such as Reference 2).

This attachment provides the technical basis (items 1 through 4 above) for the USNRC review and approval of the implementation of the Dominion Statistical DNBR Evaluation Methodology for 17x17 RFA-2 fuel at MPS3 with the VIPRE-D/WRB-2M/ABB-NV code/correlation combinations (VEP-NE-2-A Condition 1). This attachment documents that the selection and justification of the Nominal Statepoints for MPS3 (VEP-NE-2-A Condition 1), the justification of the distribution, mean and standard deviation for all the statistically treated parameters (VEP-NE-2-A Condition 2), and the justification of the value of model uncertainty for MPS3 (VEP-NE-2-A Condition 3).

3.0 Implementation of the Statistical DNBR Evaluation Methodology

3.1 Methodology Review

In Appendix C to Fleet Report DOM-NAF-2-P-A (Reference 2), Dominion calculated a DDL for the VIPRE-D/WRB-2M code/correlation pair and in Appendix D to Fleet Report DOM-NAF-2-P-A (Reference 2), Dominion calculated a DDL for the VIPRE-D/ABB-NV code/correlation pair. The Statistical DNBR Evaluation Methodology (Reference 1) is employed herein to develop SDLs for the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs for Westinghouse RFA-2 fuel at MPS3. The VIPRE-D/WLOP code/correlation will not be used for statistical analyses for Westinghouse RFA-2 fuel at MPS3. This new limit combines the correlation uncertainty with the DNBR sensitivities to uncertainties in key DNBR analysis input parameters. Even though the new DNBR limit (the SDL) is larger than the deterministic code/correlation design limit, its use is advantageous as the Statistical DNBR Evaluation Methodology permits the use of nominal values for operating initial conditions instead of requiring the application of evaluated uncertainties to the initial conditions for statepoint and transient analysis.

The SDL is developed by means of a Monte Carlo analysis. The variation of actual operating conditions about nominal statepoints due to parameter measurement and other key DNB uncertainties is modeled through the use of a random number generator. Two thousand random statepoints are generated for each nominal statepoint. The random statepoints are then supplied to the thermal-hydraulics code VIPRE-D, which calculates the minimum DNBR (MDNBR) for each statepoint. Each MDNBR is randomized by a code/correlation uncertainty factor as described in Reference 1 using the upper 95% confidence limit on the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pair measured-to-predicted (M/P) CHF ratio standard deviation (Reference 2). The standard deviation of the resultant randomized DNBR distribution is increased by a small sample correction factor to obtain a 95% upper confidence limit, and is then combined Root-Sum-Square with code and model uncertainties to obtain a total DNBR standard deviation (s_{total}). In accordance with Reference 1, the SDL is then calculated as:

$$SDL = 1 + 1.645 * s_{total} \qquad \text{[Equation 3.1]}$$

in which the 1.645 multiplier is the z-value for the one-sided 95% probability of a normal distribution. This SDL provides peak fuel rod DNB protection at greater than 95/95.

As an additional criterion, the SDL is tested to determine the full core DNB probability when the peak pin reaches the SDL. This process is performed by summing the DNB probability of each rod in the core, using a bounding fuel rod census curve and the DNB sensitivity to rod power. If necessary, the SDL is increased to reduce the full core DNB probability to 0.1% or less.

3.2 Uncertainty Analysis

This section is included herein to satisfy Condition 2 in the SER (Reference 7) of VEP-NE-2-A (Reference 1).

Consistent with VEP-NE-2-A, inlet temperature, pressurizer pressure, core thermal power, reactor vessel flow rate, core bypass flow, the nuclear enthalpy rise factor and the engineering enthalpy rise factor were selected as the statistically treated parameters in the implementation analysis. The magnitudes and functional forms of the uncertainties for the statistically treated parameters were derived in a rigorous analysis of plant hardware and measurement/calibration procedures, and have been summarized in Table 3.2-1.

The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and inlet temperature were quantified by Westinghouse for the MPS3 Stretch Power Uprate submittal (Reference 8). Total uncertainties were quantified at the 2σ level, corresponding to two-sided 95% probability. The standard deviations, σ, were obtained by dividing the total uncertainty by 1.96, which is the z-value for the two-sided 95% probability of a normal distribution.

The magnitude and distribution of uncertainty for pressurizer pressure (system pressure) per the pressurizer pressure control system were quantified. The pressurizer pressure uncertainty is a normal, two-sided, 95% probability distribution. The calculations of the MPS3 SDLs use a standard deviation (σ) of 30.0 psia.

The magnitude and distribution of uncertainty on the average temperature (Tavg) per the Tavg rod control system were quantified. The average temperature uncertainty is a normal, two-sided, 95% probability distribution. The calculations of the MPS3 SDLs use a standard deviation (σ) of 2.5°F.

The core power uncertainty is defined as a normal, two-sided, 95% probability distribution. The calculations of the MPS3 SDLs use a standard deviation (σ) of 1.0%.

The reactor coolant system (RCS) flow uncertainty is defined as a normal, two-sided, 95% probability distribution. The calculations of the MPS3 SDLs use a standard deviation (σ) of 1.5%.

The measured $F_{\Delta H}^{N}$ uncertainty is defined as a normal, two-sided, 95% probability distribution. The calculations of the MPS3 SDLs use a standard deviation (σ) of 2.0%.

The magnitude and distribution of uncertainty on the engineering hot channel factor, $F_{\Delta H}^{E}$, is quantified as a uniform probability distribution with a magnitude of $\pm$ 3.0%. The Statistical DNBR Evaluation Methodology (Reference 1) treats the $F_{\Delta H}^{E}$ uncertainty as a uniform probability distribution.

The total core bypass flow consists of separate flow paths through the thimble tubes, direct leakage to the outlet nozzle, baffle joint leakage flow, upper head spray flow and core-baffle gap flow. These five components were each quantified based on the current MPS3 core configuration, their uncertainties conservatively modeled and the flows and uncertainties totaled. The Monte Carlo analysis ultimately used a best estimate bypass flow of 7.6% with an uncertainty of 1.0%. The analysis assumed that the probability was uniformly distributed.

Table 3.2-1: MPS3 Parameter Uncertainties

PARAMETER	NOMINAL VALUE	STANDARD DEVIATION	UNCERTAINTY	DISTRIBUTION
Pressure [psia]	2250	30 psi	±58.8 psi at 2σ	Normal
Temperature [°F]	557.06	2.5°F	±4.9°F at 2σ	Normal
Power [MWt]	3,712	1.0%	±1.96% at 2σ	Normal
Flow [gpm]	379,200	1.5%	±2.94% at 2σ	Normal
$F_{\Delta H}^{N}$	1.635	2.0%	±4.0% at 2σ	Normal
$F_{\Delta H}^{E}$	1.0	N/A	±3.0%	Uniform
Bypass [%]	7.6	N/A	±1.0%	Uniform

3.3 CHF Correlations

The WRB-2M/ABB-NV/WLOP CHF correlations are used for the calculation of DNBRs in the Westinghouse 17x17 RFA-2 fuel product. Only the WRB-2M and ABB-NV CHF correlations are applicable to the operating conditions for which the Statistical DNBR Evaluation Methodology applies. Table 3.3-1 presents the DDL correlation data for VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs. The ABB-NV correlation is only used below the first mixing grid and the WLOP correlation is used when the operating conditions are outside of the range of validity of the WRB-2M and ABB-NV CHF correlations, such as the main steam-line break evaluation, where there is reduced temperature and pressure. The WLOP CHF correlation is used deterministically.

Table 3.3-1: CHF Code/Correlation Data (Reference 2)

	WRB-2M	ABB-NV[1]	
		Parametric	Non-Parametric
Average M/P	1.0006	0.9742	1.0185
S(M/P)	0.0640	0.0518	0.0654

The NRC raised a concern on the use of the calculated CHF correlation statistics for the determination of statistical DNB limits in NRC Information Notice IN-2014-01 (Reference 9). The use of the calculated

[1] For the calculation of the SDL for the VIPRE-D/ABB-NV code/correlation pair at MPS3 both the parametric and non-parametric sets presented in Appendix D to DOM-NAF-2-P-A were examined to ensure that a conservative SDL has been developed for the use of the VIPRE-D/ABB-NV code/correlation pair at MPS3.

correlation standard deviation is consistent with the methodology of VEP-NE-2-A (Reference 1). As described in VEP-NE-2-A, a 95% upper confidence limit is applied to the calculated correlation statistics. The 95% upper confidence limit, K(95), is a sample size correction factor that gives a one-sided 95% upper confidence limit on the estimated standard deviation of a given population. It may be calculated by Equation 3.2.

$$K(95) = \sqrt{\frac{2\cdot(n-1)}{\left(\sqrt{2n-3}-1.645\right)^2}} \qquad \text{[Equation 3.2]}$$

The use of the sample size correction factor given in Equation 3.2 is consistent with past applications of the methodology of VEP-NE-2-A (Reference 10, approved in References 11 and 12). Additional discussion on IN-2014-01 may be found in Section 3.6.1.

3.4 Model Uncertainty Term

This section is included herein to satisfy Condition 3 in the SER (Reference 7) of the Statistical DNBR Evaluation Methodology Topical Report (Reference 1).

The VIPRE-D 21-channel production model for MPS3 with the 17x17 RFA-2 fuel product was used in the development of the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pair SDLs for MPS3. Since the production model that Dominion intends to use for all MPS3 evaluations was used to develop the SDLs, there is no additional uncertainty associated with the use of this model. In summary, it is concluded that no correction for model uncertainty is necessary, and the model uncertainty term is set to zero for the calculation of the total DNBR standard deviation.

3.5 Code Uncertainty

The code uncertainty accounts for any differences between Dominion's VIPRE-D and other thermal-hydraulic codes, with which the WRB-2M and ABB-NV CHF data were correlated, and any effect due to the modeling of a full core with a correlation based upon bundle test data. These uncertainties are clearly independent of the correlation, the model, and parameter induced uncertainties. The code uncertainty was quantified at 5%; consistent with the factors specified for other thermal/hydraulic codes in Reference 1. The basis for this uncertainty is described in detail by USNRC staff in Reference 7. In Reference 7, the USNRC Staff refers to the 5% uncertainty as being a 2σ value. The 5% code uncertainty serves as a conservative factor that may be shown to be wholly or partially unnecessary at a later time. A one-sided 95% confidence level on the code uncertainty is then 3.04% (= (5.0%) / 1.645). The use of the 1.645 divisor (the one-sided 95% tolerance interval multiplier) is conservative since the USNRC Staff considers the 5% uncertainty to be a 2σ value.

3.6 Monte Carlo Calculations

In order to perform the Monte Carlo analysis, nine Nominal Statepoints covering the full range of normal operation and anticipated transient conditions were selected for both the WRB-2M and ABB-NV CHF correlations. These statepoints must span the range of conditions over which the statistical methodology will be applied. The Nominal Statepoints were selected to cover the DNB limiting range of the Reactor Core Safety Limits (RCSL) and within the validation range of applicability of the associated correlations. In order to apply the methodology to low flow events, low flow statepoints are also included. The selected Nominal Statepoints are listed in Tables 3.6-1 and 3.6-2.

Table 3.6-1: Nominal Statepoints for Westinghouse 17x17 RFA-2 Fuel at MPS3 with VIPRE-D/WRB-2M

STATEPOINT	PRESSURIZER PRESSURE [psia]	INLET TEMPERATURE [°F]	POWER [%]	FLOW [%]	FΔHN	MDNBR
A	2395.0	584.31	121	100	1.635	1.230
B	2395.0	599.09	112	100	1.635	1.230
C	2250.0	575.89	121	100	1.635	1.230
D	2250.0	584.03	116	100	1.635	1.231
E	2000.0	562.91	121	100	1.635	1.231
F	2282.0	573.81	95	67.25	1.660	1.231
G	2395.0	610.00	95	85.22	1.660	1.232
H	1840.0	544.00	121	91.49	1.635	1.231
I	2250.0	557.06	100	64.46	1.635	1.232

Table 3.6-2: Nominal Statepoints for Westinghouse 17x17 RFA-2 Fuel at MPS3 with VIPRE-D/ABB-NV

STATEPOINT	PRESSURIZER PRESSURE [psia]	INLET TEMPERATURE [°F]	POWER [%]	FLOW [%]	FΔHN	MDNBR
A	2385.0	476.80	121	100	1.635	1.180
B	2385.0	564.49	80	100	1.733	1.180
C	2250.0	472.35	121	100	1.635	1.180
D	2250.0	558.21	80	100	1.733	1.180
E	2000.0	460.91	121	100	1.635	1.180
F	2000.0	545.11	80	100	1.733	1.180
G	1840.0	451.18	121	100	1.635	1.180
H	1840.0	535.49	80	100	1.733	1.180
I	2250.0	442.50	100	64.00	1.635	1.180

The Monte Carlo analysis itself consisted of 2000 calculations performed around each of the nine Nominal Statepoints for each CHF correlation. As described in Section 3.1, the DNBR standard deviation at each Nominal Statepoint was augmented by the code/correlation uncertainty, the small sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation.

The Total s_{Total}, is obtained using the Root-Sum-Square method according to Equation 3.3:

$$s_{TOTAL} = \sqrt{s_{DNBR}^2 \cdot \left(1.0 + \sqrt{\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}^2 + \frac{1}{N}}\right)^2 + F_C^2 + F_M^2} \qquad \text{[Equation 3.3]}$$

where:

s_{DNBR} is the standard deviation for the Randomized DNBR distribution.

The factor $\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}$ is the uncertainty in the standard deviation of the 2,000 Monte Carlo simulations, and provides a 95% upper confidence limit on the standard deviation.

$^1/_N$ is the uncertainty in the mean of the correlation. N is the number of degrees of freedom in the original correlation database (given in DOM-NAF-2-P-A, Reference 2).

F_C is the code uncertainty, that has been defined as 5% (2σ value), i.e., $^{5.0\%}/_{1.645} = 3.04\%$ (1σ value). See Section 2.5 in Reference 1.

F_M is the model uncertainty, which is 0.0 since the Monte Carlo simulation is run with the production model.

Note that this equation differs slightly from the equation listed in Reference 1. It has an additional factor applied to the Randomized DNBR s_{DNBR}, the $^1/_N$ factor to correct for the uncertainty in the mean of the correlation. This factor has been used in previous implementations of the Statistical DNBR Evaluation Methodology, such as Reference 13 as supplemented by Reference 14, which was subsequently approved in Reference 15.

The limiting peak fuel rod SDL was calculated to be 1.225 for the VIPRE-D/WRB-2M code/correlation pair and 1.177 for the VIPRE-D/ABB-NV code/correlation pair. The Monte Carlo Statepoint analysis is summarized in Tables 3.6-3 and 3.6-4.

Table 3.6-3: Peak Pin SDL Results for MPS3 17x17 RFA-2 Fuel with VIPRE-D/WRB-2M

STATEPOINT	Randomized DNB s_{DNBR}	Total DNB s_{TOTAL}	Pin Peak $SDL_{95/95}$
A	0.1231	0.1352	1.222
B	0.1214	0.1335	1.220
C	0.1246	0.1368	1.225
D	0.1216	0.1337	1.220
E	0.1227	0.1349	1.222
F	0.1230	0.1352	1.222
G	0.1171	0.1290	1.212
H	0.1184	0.1304	1.214
I	0.1244	0.1366	1.225

Table 3.6-4: Peak Pin SDL Results for MPS3 17x17 RFA-2 Fuel with VIPRE-D/ABB-NV[2]

STATEPOINT	Parametric			Non-Parametric		
	Randomized DNB s_{DNBR}	Total DNB s_{TOTAL}	Pin Peak $SDL_{95/95}$	Randomized DNB s_{DNBR}	Total DNB s_{TOTAL}	Pin Peak $SDL_{95/95}$
A	0.0751	0.0849	1.140	0.0928	0.1039	1.171
B	0.0784	0.0882	1.145	0.0957	0.1069	1.176
C	0.0753	0.0851	1.140	0.0929	0.1040	1.171
D	0.0769	0.0867	1.143	0.0938	0.1049	1.173
E	0.0758	0.0855	1.141	0.0931	0.1042	1.171
F	0.0775	0.0873	1.144	0.0945	0.1057	1.174
G	0.0779	0.0876	1.144	0.0953	0.1065	1.175
H	0.0793	0.0890	1.146	0.0962	0.1074	1.177
I	0.0745	0.0843	1.139	0.0921	0.1032	1.170

3.6.1 Discussion Related to NRC IN-2014-01

As discussed in Section 3.3, the NRC has issued IN-2014-01 (Reference 9). The development of the SDLs herein used the calculated standard deviations of the correlations approved in DOM-NAF-2-P-A (Reference 2). This is consistent with the methodology of VEP-NE-2-A (Reference 1). The acceptability of the use of the calculated standard deviations is based on the use of a 95% upper confidence factor that is essentially equivalent to the Owen's tables for ensuring a 95% probability at a 95% confidence limit.

Starting with Equation 3.1 with the value of N set equal to n and substituting Equation 3.3 in for s_{total}, Equation 3.4 is obtained.

$$SDL = 1 + 1.645 * \sqrt{s_{DNBR}^2 \cdot \left(1.0 + \sqrt{\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}^2 + \frac{1}{n}}\right)^2 + F_C^2 + F_M^2} \qquad \text{[Equation 3.4]}$$

Neglecting the terms for the code, F_c, and model, F_M, from Equation 4 and pulling the term s_{DNBR} out from under the radical, Equation 3.5 is obtained.

$$SDL = 1 + s_{DNBR} * 1.645 * \sqrt{\left(1.0 + \sqrt{\left\{\sqrt{\frac{n-1}{\chi^2}} - 1.0\right\}^2 + \frac{1}{n}}\right)^2} \qquad \text{[Equation 3.5]}$$

[2] For the calculation of the SDL for the VIPRE-D/ABB-NV code/correlation pair at MPS3 both the parametric and non-parametric sets presented in Appendix D to DOM-NAF-2-P-A were examined to ensure that a conservative SDL has been developed for the use of the VIPRE-D/ABB-NV code/correlation pair at MPS3.

As demonstrated in Section 2.4 of VEP-NE-2-A (Reference 1), Equation 3.2 may be substituted into Equation 3.5 as shown in Equation 3.6.

$$SDL = 1 + s_{DNBR} * 1.645 * \sqrt{\left(1.0 + \sqrt{\left\{\sqrt{\frac{2n-2}{(\sqrt{2n-3}-1.645)^2}} - 1.0\right\}^2 + \frac{1}{n}}\right)^2} \qquad \text{[Equation 3.6]}$$

From inspection of Equation 3.6, the Owen's factor is essentially equivalent to the term

$$1.645 * \sqrt{\left(1.0 + \sqrt{\left\{\sqrt{\frac{2n-2}{(\sqrt{2n-3}-1.645)^2}} - 1.0\right\}^2 + \frac{1}{n}}\right)^2}\ .$$

The use of the correction factor to s_{DNBR} ensures that the SDL developed in accordance with the methodology of VEP-NE-2-A and using the calculated correlation statistics provides a 95% probability at a 95% confidence level (95/95) that the peak rod does not experience DNB. Thus, the use of the calculated correlation statistics as an input to the calculation of the SDL compared to the approved correlation statistics is acceptable and reasonable following the methodology of VEP-NE-2-A.

3.7 Full Core DNB Probability Summation

After the development of the peak pin 95/95 DNBR limit, the data statistics are used to determine the number of rods expected in DNB. The DNB sensitivity to rod power is estimated as $\partial(\text{DNBR})/\ \partial(1/F_{\Delta H})$. The specific values of $\partial(\text{DNBR})/\ \partial(1/F_{\Delta H})$, denoted β, are listed in Tables 3.7-1 and 3.7-2.

To ensure that the calculations are conservative, a one-sided tolerance limit of β is used:

$$\beta^* = \beta - t(\alpha, \nu) \cdot se(\beta)$$

in which:

β^* is the one-sided tolerance limit on β

$t(\alpha,\nu)$ is the T-statistic with significance level α and ν degrees of freedom. For 2,000 observations at a 0.05 level of significance t(0.05,2000) = 1.645.

$se(\beta)$ is the standard error of β.

The variable $1/F_{\Delta H}$ is the most statistically significant independent variable in the linear regression model, yielding R^2 values larger than 99%. The value of the statistic parameter F of $1/F_{\Delta H}$ was the largest for all statepoints, which indicates that the variable $1/F_{\Delta H}$ accounts for the largest amount of the variation in the DNBR.

Table 3.7-1: $\partial(DNBR)/\partial(1/F_{\Delta H})$ Estimation for WRB-2M

STATEPOINT	β	se(β)	β^*	R^2
A	4.834531	0.003741	4.828377	99.9%
B	4.746314	0.004507	4.738900	99.9%
C	4.981443	0.004826	4.973504	99.9%
D	4.932317	0.005322	4.923563	99.9%
E	4.986427	0.005193	4.977885	99.9%
F	5.294904	0.004914	5.286821	99.9%
G	4.894448	0.004375	4.887251	99.9%
H	4.953918	0.005269	4.945249	99.9%
I	5.296037	0.005149	5.287567	99.9%

Table 3.7-2: $\partial(DNBR)/\partial(1/F_{\Delta H})$ Estimation for ABB-NV

STATEPOINT	β	se(β)	β^*	R^2
A	2.227729	0.001180	2.225788	99.97%
B	2.499780	0.001637	2.497087	99.96%
C	2.241232	0.001251	2.239175	99.96%
D	2.516105	0.001525	2.513596	99.97%
E	2.356426	0.001494	2.353968	99.96%
F	2.561183	0.001645	2.558478	99.96%
G	2.372180	0.001454	2.369788	99.96%
H	2.585910	0.001682	2.583143	99.96%
I	2.282289	0.001282	2.280179	99.96%

A representative fuel rod census curve used for the determination of the SDL is listed in Table 3.7-3. The full core DNB probability summation is evaluated on a reload basis to verify the applicability of the fuel rod census ($F_{\Delta H}^N$ versus % of core with $F_{\Delta H}^N$ greater than or equal to a given $F_{\Delta H}$ limit) used in the implementation analysis. The limiting full-core DNB probability summation resulted in an SDL of 1.195 for WRB-2M and 1.183 for ABB-NV. The DNB probability summations for VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs are summarized in Tables 3.7-4 and 3.7-5, respectively.

Table 3.7-3: Representative Fuel Rod Census for a Maximum Peaking Factor $F_{\Delta H}$ = 1.635

MAXIMUM % OF FUEL RODS IN CORE WITH $F_{\Delta H} \geq$ to:	$F_{\Delta H}$ LIMIT
0.0	1.6350
0.1	1.6330
0.2	1.6300
0.3	1.6270
0.4	1.6250
0.5	1.6220
0.6	1.6180
0.7	1.6150
0.8	1.6120
0.9	1.6080
1.0	1.6040
1.5	1.5930
2.0	1.5810
2.5	1.5710
3.0	1.5620
4.0	1.5500
5.0	1.5380
6.0	1.5270
7.0	1.5160
8.0	1.5050
9.0	1.4960
10.0	1.4900
20.0	1.4580
30.0	1.4110
40.0	1.3490
PEAK	1.6350

Table 3.7-4: Full Core DNB Probability Summation for 17x17 RFA-2 Fuel with VIPRE-D/WRB-2M

STATEPOINT	s_{TOTAL}	% of Rods in DNB	Full Core $SDL_{99.9}$
A	0.1352	0.099599	1.194
B	0.1335	0.099639	1.192
C	0.1368	0.099300	1.195
D	0.1337	0.098525	1.190
E	0.1349	0.099453	1.191
F	0.1352	0.099496	1.187
G	0.1290	0.099055	1.181
H	0.1304	0.098831	1.183
I	0.1366	0.099260	1.190

Table 3.7-5: Full Core DNB Probability Summation for 17x17 RFA-2 Fuel with VIPRE-D/ABB-NV

STATEPOINT	s_{TOTAL}	% of Rods in DNB	Full Core $SDL_{99.9}$
A	0.1039	0.098456	1.181
B	0.1069	0.097684	1.181
C	0.1040	0.098327	1.181
D	0.1049	0.097770	1.176
E	0.1042	0.098860	1.178
F	0.1057	0.099208	1.176
G	0.1065	0.098826	1.183
H	0.1074	0.099684	1.179
I	0.1032	0.098381	1.178

3.8 Verification of Nominal Statepoints

Condition 1 of the USNRC's SER for VEP-NE-2-A (Reference 7) requires that the Nominal Statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may potentially be applied.

It is therefore necessary to demonstrate that s_{total} as calculated herein is maximized for any conceivable set of conditions at which the core may approach the SDL. To do so, a regression analysis is performed using the unrandomized DNBR standard deviations at each Nominal Statepoint as the dependent variable (i.e., the raw MDNBR results obtained from the Monte Carlo simulation). The Nominal

Statepoint pressures, inlet temperatures, powers and flow rates are used as the independent variable. If no clear trend appears in the plot it can be concluded that the standard deviation has been maximized. If a clear trend is displayed, the regression function is determined. This regression equation is evaluated to determine the values of the independent variable for which the standard deviation would be maximized, and it is verified that the Nominal Statepoints selected bound those conditions. In addition, the residuals of the regression are plotted again against all the independent variables, and it is verified that no trends are discernible.

Tables 3.8-1 and 3.8-2 show the R^2 coefficients obtained for the verification of the nominal statepoints for WRB-2M and ABB-NV, respectively. The largest linear curve fit R^2 coefficient is 7.70% for WRB-2M and 34.95% for ABB-NV, thus validating that there is no dependence.

An evaluation of all the data, linear fits, and R^2 coefficients indicates that there are no discernible trends in the database. Therefore, it was concluded that s_{TOTAL} had been maximized for any conceivable set of conditions at which the core may approach the SDL and that the selected Nominal Statepoints provide a bounding standard deviation for any set of conditions to which the methodology may potentially be applied. Figure 3.8-1 and Figure 3.8-2 display a sample regression plots for WRB-2M and ABB-NV, respectively, and clearly shows the trends discussed above.

Table 3.8-1: R^2 Coefficients for the Verification of the Nominal Statepoints for MPS3 17x17 RFA-2 Fuel with VIPRE-D/WRB-2M

	R^2 - Linear Regression
Pressure	2.75%
Temperature	7.70%
Flow Rate	1.27%
Power	3.04%

Table 3.8-2: R^2 Coefficients for the Verification of the Nominal Statepoints for MPS3 17x17 RFA-2 Fuel with VIPRE-D/ABB-NV

	R^2 - Linear Regression
Pressure	23.62%
Temperature	34.95%
Flow Rate	24.30%
Power	27.76%

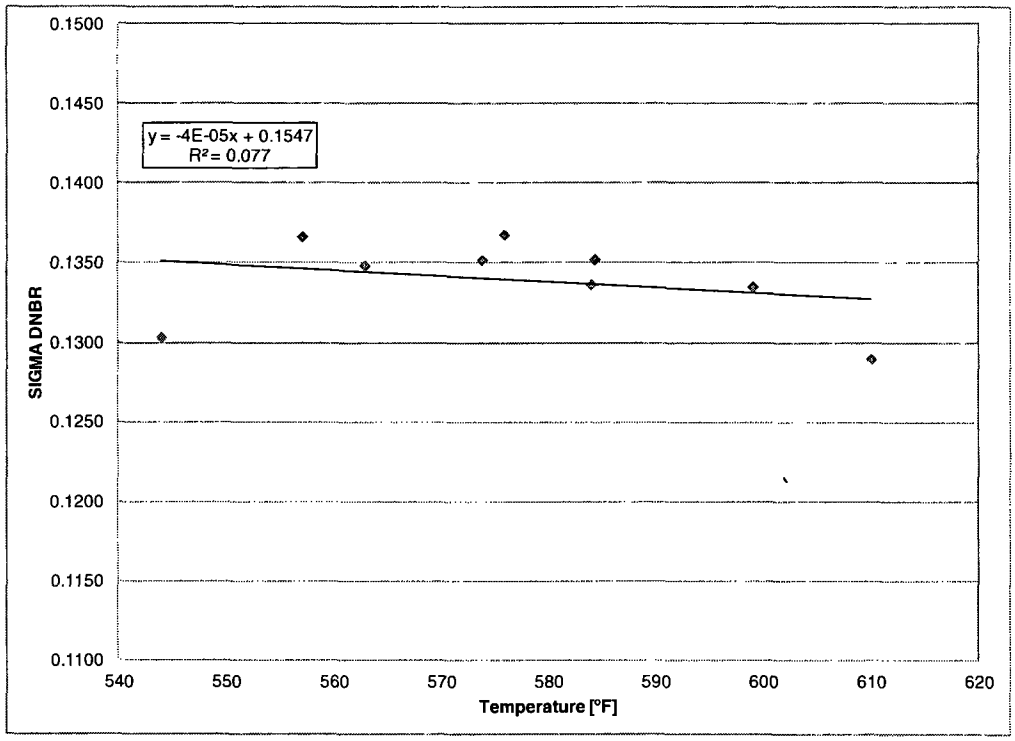

Figure 3.8-1: Variation of the Unrandomized Standard Deviation with Temperature for the WRB-2M CHF Correlation

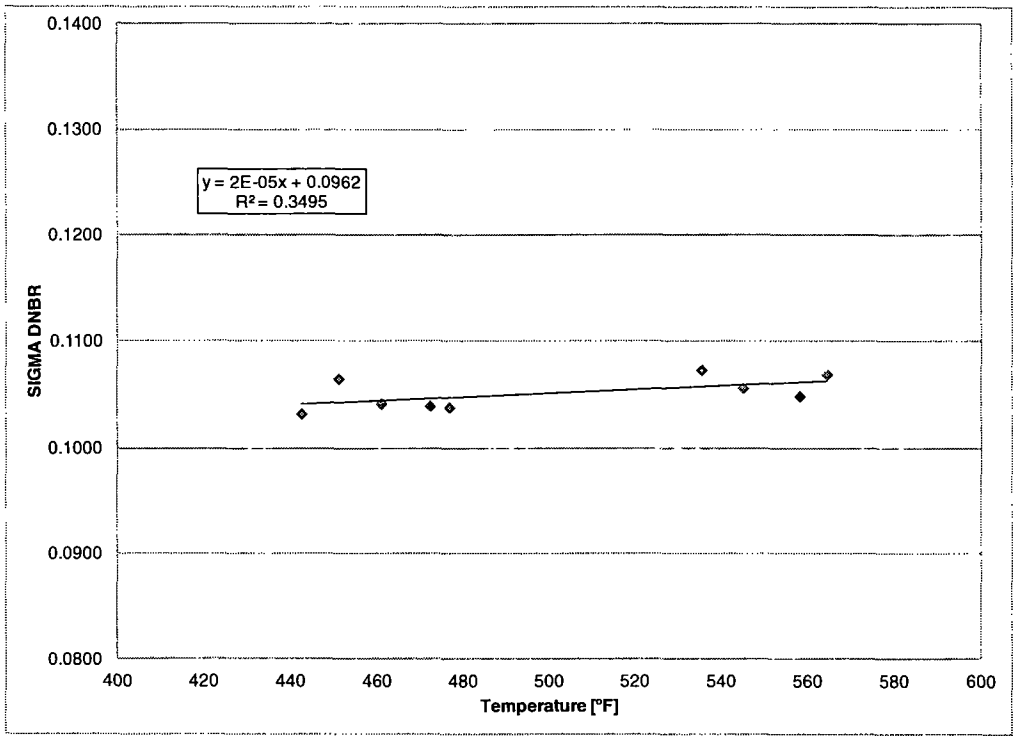

Figure 3.8-2: Variation of the Unrandomized Standard Deviation with Temperature for the ABB-NV CHF Correlation

3.9 Scope of Applicability

This section is included herein to satisfy Condition 4 in the SER (Reference 7) of VEP-NE-2-A (Reference 1).

The Statistical DNBR Evaluation Methodology may be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical (RWSC) which is initiated from zero power), and to the complete Loss of Flow, the Locked Rotor Accident, the Single Rod Cluster Control Assembly Withdrawal at Power, and feedwater system pipe break. The accidents to which the methodology is applicable are listed in Table 3.9-1. This table corresponds to Table 2.1-1 in Reference 2. This methodology will not be applied to accidents that are initiated from zero power where the parameter uncertainties are higher.

The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and hot channel uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

Table 3.9-1: FSAR Transients Analyzed with VIPRE-D/WRB-2M/ABB-NV/WLOP for MPS3

ACCIDENT	MPS3 FSAR SECTION	APPLICATION
Feedwater system malfunctions that result in a decrease in feedwater temperature (Excessive heat removal due to feedwater system malfunctions)	15.1.1	STAT-DNB
Feedwater system malfunctions that result in an increase in feedwater flow (Excessive heat removal due to feedwater system malfunctions)	15.1.2	STAT-DNB
Excessive increase in secondary steam flow (Excessive load increase)	15.1.3	STAT-DNB
Inadvertent opening of a steam generator relief or safety valve (Accidental depressurization of the main steam system)	15.1.4	DET-DNB
Steam system piping failure (Rupture of a main steam pipe)	15.1.5	DET-DNB
Loss of external electrical load (Loss of external electrical load and/or turbine trip)	15.2.2	STAT-DNB
Turbine trip (Loss of external electrical load and/or turbine trip)	15.2.3	STAT-DNB
Inadvertent closure of a main steam isolation valve (Loss of external electrical load and/or turbine trip)	15.2.4	STAT-DNB

ACCIDENT	MPS3 FSAR SECTION	APPLICATION
Loss of condenser vacuum and other events resulting in turbine trip (Loss of external electrical load and/or turbine trip)	15.2.5	STAT-DNB
Loss of nonemergency AC power to the station auxiliaries (Loss of external electrical load and/or turbine trip)	15.2.6	STAT-DNB
Loss of normal feedwater flow (Loss of normal feedwater flow)	15.2.7	STAT-DNB
Feedwater system pipe break (Major rupture of a main feed water pipe)	15.2.8	STAT-DNB
Partial loss of forced reactor coolant flow (Loss of forced reactor coolant flow)	15.3.1	STAT-DNB
Complete loss of forced reactor coolant flow(Loss of forced reactor coolant flow)	15.3.2	STAT-DNB
Reactor coolant pump shaft seizure (Locked reactor coolant pump rotor or shaft break)	15.3.3	STAT-DNB
Reactor coolant pump shaft break (Locked reactor coolant pump rotor or shaft break)	15.3.4	STAT-DNB
Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Rod cluster control assembly bank withdrawal from subcritical)	15.4.1	DET-DNB
Uncontrolled rod cluster control assembly bank withdrawal at power (Rod cluster control assembly bank withdrawal at power)	15.4.2	STAT-DNB
Rod cluster control assembly misalignment (Single rod cluster control assembly withdrawal at full power)	15.4.3	STAT-DNB
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Uncontrolled boron dilution)	15.4.6	STAT-DNB
Inadvertent operation of the emergency core cooling system during power operation (Inadvertent operation of the emergency core cooling system during power operation)	15.5.1	STAT-DNB
Inadvertent opening of a pressurizer safety or relief valve (Accidental depressurization of the reactor coolant system)	15.6.1	STAT-DNB

3.10 Summary of Analysis

The steps of the SDL derivation analysis may be summarized as follows:

In accordance with the Statistical DNBR Evaluation Methodology, 2,000 random statepoints are generated about each nominal statepoint and VIPRE-D is then executed to obtain MDNBRs. The standard deviation for the distribution of 2,000 MDNBRs is referred to as the unrandomized standard deviation. At the limiting Nominal Statepoint, the standard deviation of the randomized DNBR distributions, which is the unrandomized standard deviation corrected for CHF correlation uncertainty. This value was then combined Root Sum Square with code and model uncertainty standard deviations to obtain a total DNBR standard deviation, listed in Tables 3.6-3 and 3.6-4. The use of total DNBR standard deviation in Equation 3.1 yields a peak pin DNBR limit of 1.225 for VIPRE-D/WRB-2M and 1.177 for VIPRE-D/ABB-NV with at least 95% probability at a 95% confidence level. The total DNBR standard deviation was then used to obtain 99.9% DNB protection in the full core of 1.195 for VIPRE-D/WRB-2M and 1.183 for VIPRE-D/ABB-NV. Therefore the VIPRE-D/WRB-2M code/correlation pair SDL for MPS3 17x17 RFA-2 fuel is set to 1.23 and the VIPRE-D/ABB-NV code/correlation pair SDL for MPS3 17x17 RFA-2 fuel is set to 1.19.

4.0 Application of VIPRE-D/WRB-2M/ABB-NV/WLOP to MPS3

VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs together with the Statistical DNBR Evaluation Methodology will be applied to all Condition I and II DNB events (except Rod Withdrawal from Subcritical, RWSC), and to the Complete Loss of Flow event and the Locked Rotor Accident. The Statistical DNBR Evaluation Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and $F_{\Delta H}^{E}$ (engineering component) uncertainties. These uncertainties are convoluted statistically into the DNBR limit.

The WRB-2M, ABB-NV and WLOP CHF correlations are used for the calculation of DNBRs in the Westinghouse 17x17 RFA-2 fuel product. The ABB-NV correlation is only used below the first mixing grid and WLOP correlation is only used when the operating conditions are outside of the range of validity of the WRB-2M and ABB-NV CHF correlations, such as the main steam-line break evaluation, where there are reduced temperature and pressure. The WLOP CHF correlation is used deterministically.

In addition, there are a few events that will be evaluated with deterministic models because they do not meet the applicability requirements of the Statistical DNBR Evaluation Methodology (see the events in Table 3.9-1 labeled 'DET-DNB'). These events are initiated from bounding operating conditions considering the nominal value and the appropriate uncertainty value, and require the application of the bypass flow, $F_{\Delta H}^{N}$ (measurement component) and $F_{\Delta H}^{E}$ uncertainties. The events modeled deterministically are limited by the DDL stated in DOM-NAF-2-P-A (Reference 2).

4.1 VIPRE-D/WRB-2M/ABB-NV SDLs for MPS3

The SDLs for MPS3 cores containing Westinghouse 17x17 RFA-2 fuel being analyzed with the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs were derived in Section 3 of this attachment. The SDL for VIPRE-D/WRB-2M code/correlation pair is determined to be 1.23. The SDL for VIPRE-D/ABB-NV code/correlation pair is determined to be 1.19. The SDL limit provides a peak fuel rod DNB protection with at least 95% probability at a 95% confidence level and a 99.9% DNB protection for the full core. This SDL is plant specific as it already includes the MPS3 specific uncertainties for the key parameters accounted for in the application of the Statistical DNBR Evaluation Methodology. Therefore, these limits are applicable to the analysis of statistical DNB events of Westinghouse 17x17 RFA-2 fuel in MPS3 cores with the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pairs.

4.2 Safety Analysis Limits (SAL)

In the performance of in-house DNB thermal-hydraulic evaluations, design limits and safety analysis limits are used to define the available retained DNBR margin for each application. The difference between the safety analysis (self-imposed) limit and the design limit is the available retained DNBR margin.

For deterministic DNB analyses, the design DNBR limit is set equal to the applicable code/correlation limit and it is termed the DDL. For statistical DNB analyses, the design DNBR limit is set equal to the applicable SDL. These design limits are two of the DBLFPB described in Reference 16. The DDLs and SDLs are fixed and any changes to their value require USNRC review and approval. However, the safety analysis limits for deterministic and statistical DNB analyses (SAL_{DET} and SAL_{STAT}, respectively) may be changed without prior USNRC review and approval, provided the changes meet the criteria established in Reference 16.

A deterministic and statistical SAL equal to 1.50 has been selected for 17x17 RFA-2 fuel at MPS3 with the VIPRE-D/WRB-2M, VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs. This SAL is applicable for all deterministic analyses for a maximum peaking factor $F_{\Delta H}^{N}$ equal to 1.65 and for all statistical analyses for a maximum peaking factor $F_{\Delta H}^{N}$ equal to 1.587.

Table 4.2-1: DNBR Limits for WRB-2M, ABB-NV and WLOP

VIPRE-D/WRB-2M	
DDL	1.14
SDL	1.23
SAL	1.50
VIPRE-D/ABB-NV	
DDL	1.14
SDL	1.19
SAL	1.50
VIPRE-D/WLOP	
DDL	1.22
SAL	1.50

4.3 Retained DNBR Margin

The difference between the safety analysis (self-imposed) limit and the design limit is the available retained DNBR margin:

$$Retained\ DNBR\ Margin\ [\%] = 100 * \left(\frac{SAL - DDL}{SAL}\right)$$

The resulting available retained DNBR margins are listed in Tables 4.3-1 and 4.3-2.

Table 4.3-1: DNBR Limits and Retained DNBR Margin for Deterministic DNB Applications

DETERMINISTIC DNB APPLICATIONS			
DNB CORRELATION	DDL	SAL_{DET}	RETAINED DNBR MARGIN [%]
WRB-2M	1.14	1.50	24.0
ABB-NV	1.14	1.50	24.0
WLOP	1.22	1.50	18.6

Table 4.3-2: DNBR Limits and Retained DNBR Margin for Statistical DNB Applications

STATISTICAL DNB APPLICATIONS			
DNB CORRELATION	SDL	SAL_{STAT}	RETAINED DNBR MARGIN [%]
WRB-2M	1.23	1.50	18.0
ABB-NV	1.19	1.50	20.6

This method of defining retained DNBR margin allows all of the DNBR margin to be found in a single, clearly defined location. The retained DNBR margin may be used to offset generic DNBR penalties, such as a rod bow penalty.

The reload thermal-hydraulics evaluation prepared as part of the reload safety analysis process (Reference 17) presents tables and descriptions of retained DNBR margin and applicable penalties. Retained DNBR margin is tracked separately for each CHF correlation and for statistical and deterministic analyses.

4.4 Verification of Existing Reactor Core Safety Limits, Protection Setpoints and MPS3 FSAR Chapter 15 Events

This section is included herein to satisfy Condition 3 of the plant specific application list in Section 2.1 of DOM-NAF-2-P-A (Reference 2).

To demonstrate that the DNB performance of the Westinghouse 17x17 RFA-2 fuel is acceptable, Dominion performed calculations for full-core configurations of Westinghouse 17x17 RFA-2 fuel. The calculations were performed using the VIPRE-D/WRB-2M, VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs and selected statepoints including: the reactor core safety limits (RCSL), various

limiting axial flux shapes, rod withdrawal from subcritical (RWSC), rod withdrawal at power (RWAP), loss of flow (LOFA), locked rotor events (LOCROT), hot zero power steam line break (MSLB), dropped rod limit line (DRLL), and static rod misalignment (SRM). These various statepoints provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power $F_{\Delta H}$), pressure and temperature (RCSL); (b) limiting axial flux shapes at several axial offsets (AO); and (c) low flow (LOFA and LOCROT). The statepoints for the RWSC and MSLB were evaluated with deterministic DNB methods. The remaining statepoints were evaluated using statistical DNB methods. The evaluation criterion for these analyses is that the minimum DNBR must be equal to or greater than the applicable safety analysis limit (SAL) listed in Table 4.2-1.

The minimum DNBR values are demonstrated to be equal to or greater than the applicable safety analysis limit for all of the analyses that are performed to address statepoints of the Reactor Core Safety Limits, the OTΔT, OPΔT and FΔI trip setpoints, as well as all the evaluated Chapter 15 events (including the LOFA and LOCROT) with an $F_{\Delta H}^{N}$ of 1.587 for statistical events at a Rated Thermal Power of 3650 MWt.

5.0 Conclusions

This attachment supports the application of USNRC approved VEP-NE-2-A to MPS3. It provides the technical basis and documentation required to evaluate the plant specific application of the VEP-NE-2-A method to MPS3. This application employs the VIPRE-D code with the Westinghouse WRB-2M, ABB-NV and WLOP Critical Heat Flux (CHF) correlations (VIPRE-D/WRB-2M, VIPRE-D/ABB-NV and VIPRE-D/WLOP) for the thermal-hydraulic analysis of Westinghouse 17x17 RFA-2 fuel assemblies at MPS3. In particular, Dominion requests the review and approval of the Statistical Design Limits (SDLs) of 1.23 for VIPRE-D/WRB-2M and 1.19 for VIPRE-D/ABB-NV as documented herein as per 10 CFR 50.59(c)(2)(vii) since they constitute Design Basis Limits for Fission Products Barrier (DBLFPB).

Dominion is also seeking the approval for the inclusion of Topical Report VEP-NE-2-A and Fleet Report DOM-NAF-2-P-A, Appendices C and D, to the MPS3 Technical Specification 6.9.1.6.b list of USNRC approved methodologies used to determine core operating limits (i.e., the reference list of the Millstone Unit 3 COLR). This would allow Dominion the use of the VIPRE-D/WRB-2M, VIPRE-D/ABB-NV and VIPRE-D/WLOP code/correlation pairs to perform licensing calculations for the Westinghouse 17x17 RFA-2 fuel in MPS3 cores, using the DDLs qualified in Appendices C and D of Fleet Report DOM-NAF-2-P-A, and the SDLs documented herein.

6.0 References

1. Topical Report, VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

2. Fleet Report, DOM-NAF-2-P-A, Rev. 0.3, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," September 2014.

3. Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4," May 1, 1986.

4. Letter from A. C. Thadani (NRC) to Y. Y. Yung (VIPRE-01 Maintenance Group), "Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, 'VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores,' (TAC No. M79498)," October 30, 1993.

5. Letter from D. Wright (NRC) to D. A. Christian (Dominion), "Kewaunee Power Station, Millstone Power Station, Units 2 and 3, North Anna Power Station, Unit Nos. 1 and 2, and Surry Power Station, Unit Nos. 1 and 2 – Appendix C to the Dominion Fleet Report DOM-NAF-2, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code" (TAC Nos. MD8703, MD8704, MD8705, MD8707, MD8708, MD8709)," Dominion Serial No. 09-290, April 22, 2009 (ADAMS Accession No. ML091030639).

6. Letter from V. Sreenivas (USNRC) to D. A. Heacock, "North Anna and Surry Power Stations Units 1 and 2, Issuance of Amendments Regarding Addition of an Analytical Methodology to the North Anna and Surry Core Operating Limits reports and an Increase to the Surry Minimum Temperature for Criticality (TAC Nos. MF2364, MF2365, MF2366, and MF2367)," Dominion Serial No. 14-410, August 12, 2014 (ADAMS Accession No. ML ML14169A359).

7. Letter from L. B. Engle (NRC) to W. L. Stewart (Virginia Power), "Statistical DNBR Evaluation Methodology, VEP-NE-2, Surry Power Station, Units No. 1 & No. 2 (Surry-1&2) and North Anna Power Station, Units No. 1 & No. 2 (NA-1&2)," Dominion Serial No. 87-335, May 28, 1987.

8. Letter from G. T. Bischof (Dominion) to USNRC, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, License Amendment Request, Stretch Power Uprate," Serial No. 07-0450, July 13, 2007.

9. NRC Information Notice, IN-2014-01, "Fuel Safety Limit Calculation Inputs were Inconsistent with NRC-Approved Correlation Limit Values," February 21, 2014.

10. Letter from J. A. Price (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed License Amendment Request (LAR), Addition of Analytical Methodology to COLR," Serial No. 10-404, July 19, 2010 (ADAMS Accession No. ML102020165).

11. Letter from V. Sreenivas (USNRC) to D. A. Heacock (Dominion), "North Anna Power Station, Units 1 and 2, Issuance of Amendments Regarding Addition of an Analytical Methodology to Core Operating Limits Reports for the Critical Heat Flux Correlation (TAC Nos. ME4262 and ME4263)," Dominion Serial No. 12-129, February 29, 2012 (ADAMS Accession No. ML12054A162).

12. Letter from V. Sreenivas (USNRC) to D. A. Heacock (Dominion), "North Anna Power Station, Units 1 and 2 – Correction Letter, RE: Issuance of Amendments Regarding Addition of an Analytical Methodology to Core Operating Limits Reports for the Critical Heat Flux Correlation (TAC Nos. ME4262 and ME4263)," Dominion Serial No. 12-184, March 13, 2012 (ADAMS Accession No. ML12066A208).

13. Letter from E. S. Grecheck (Dominion) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specification Changes, Addition of Analytical Methodology to COLR," Serial No. 05-419, July 5, 2005 (ADAMS Accession No. ML051890034).

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15. Letter from S. Monarque (NRC) to D. A. Christian (Dominion), "North Anna Power Station, Unit Nos. 1 and 2, Issuance of Amendments on Changes to Analytical Methodology and Core Operating Limits Report (TAC Nos. MC7526 and MC7527)," Dominion Serial No. 06-643, July 21, 2006 (ADAMS Accession No. ML062020005).

16. Technical Report, NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," Nuclear Energy Institute, November 2000.

17. Topical Report, VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.