VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 11, 2003

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

Serial No.	03-323
NL&OS/ETS	R0
Docket Nos.	50-338/339
License Nos.	NPF-4/7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGES
USE OF REALISTIC LBLOCA AND SMALL BREAK LOCA METHODOLOGY
FOR THE ANALYSIS OF FRAMATOME ADVANCED MARK-BW FUEL

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests an amendment to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. This amendment request supplements the amendment request made in our March 28, 2002 letter (Serial No. 02-167) for use of Framatome ANP Advanced Mark-BW fuel. The proposed changes revise the analytical methods in TS 5.6.5.b, Core Operating Limits Report that were included in the March 28, 2002 amendment request. These changes are necessary to permit use of the realistic large break loss of coolant accident (RLBLOCA) analytical methods for the analysis of the Advanced Mark-BW fuel. In addition, the proposed changes include a reference to the Framatome small break loss of coolant accident (SBLOCA) methods to support the future Framatome SBLOCA analyses. These proposed changes to TS Section 5.6.5.b supercede the changes to TS Section 5.6.5.b requested in our March 28, 2002 letter.

The Framatome RLBLOCA methodology requires the use of certain inputs generated using applicable physics methods. The existing Technical Specifications already contain an NRC-approved methodology (VEP-NE-1-A) that supply such inputs. However, Framatome topical report, EMF-96-029, is being included to provide another NRC-approved methodology applicable to Westinghouse three-loop reactors with 157 fuel assemblies with 17x17 fuel rod arrays. Both methodologies are capable of providing the necessary inputs for the Framatome RLBLOCA methodology. The attachments to this letter provide documentation of the proposed Technical Specifications changes and the assessment of these changes associated with the use of the RLBLOCA and SBLOCA analysis methods. Also, attached for information only, are the related Bases changes. The Bases changes will be revised in accordance with the Technical Specifications Bases Control Program following NRC approval of the proposed license amendment.

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As discussed in Attachment 1, the use of RLBLOCA and SBLOCA methods will comply with the requirements and limitations provided in the Safety Evaluation Reports (SER) issued by the USNRC for the applicable Framatome topical reports. The proposed Technical Specifications changes for use of the RLBLOCA and SBLOCA analytical methods are provided in Attachment 2.

It has been determined that the use of RLBLOCA methods (including the use of Framatome physics methods) and SBLOCA methods for the analysis of Advanced Mark-BW fuel does not constitute a significant hazard as defined in 10 CFR 50.92, as discussed in Attachment 3. In addition, the proposed changes have been determined to qualify for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9), as discussed in Attachment 4. The proposed changes and supporting evaluations have been reviewed by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

If you have any questions or require additional information on this, please contact Mr. Thomas Shaub at (804) 237-2763.

Very truly yours,

L. N. Hartz
Vice President – Nuclear Engineering

Commitments made in this letter: None

Attachments

cc: U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW
Suite 23 T85
Atlanta, Georgia 30303

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

Mr. S. R. Monarque
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Mail Stop 8H12
Rockville, MD 20852

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center
4201 Dominion Blvd.
Suite 300
Glen Allen, Virginia 23060

Commissioner
Bureau of Radiological Health
1500 East Main Street
Suite 240
Richmond, VA 23218

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 11th day of June, 2003.

My Commission Expires: March 31, 2004.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Framatome Fuel Transition Program

Discussion of Proposed Technical Specifications Changes

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

DISCUSSION OF CHANGES

INTRODUCTION

In a March 28, 2002 letter (Reference 1) Virginia Electric and Power Company (Dominion), operator of the North Anna Power Station (NAPS), requested changes to the technical specifications and exemptions to 10CFR50 to permit use of Framatome ANP, Inc[1] Advanced Mark-BW fuel. Dominion has decided, based on recent interactions with the NRC staff and as documented in Reference 2, to use the Framatome Realistic Large Break Loss of Coolant Accident (RLBLOCA) methodology for the large break LOCA analyses. In addition, recent requests for information from the NRC (Reference 3) require that a small break LOCA (SBLOCA) analysis be performed and submitted. These proposed technical specifications changes supplement the technical specifications changes proposed in Reference 1 and are necessary to employ the Framatome RLBLOCA and SBLOCA methodologies. The remainder of the Reference 1 proposed technical specifications changes are unaffected by this supplement. Framatome will deliver fuel assemblies of the Advanced Mark-BW design to Dominion, beginning with Cycle 17 for North Anna Unit 2 and Cycle 18 for Unit 1. In addition, the associated technical specifications bases changes necessary to employ the Framatome SBLOCA and RLBLOCA methodologies are included for information.

The proposed technical specification changes and bases changes have been reviewed, and have been determined to qualify for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed technical specification changes.

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

The proposed changes are administrative in nature, involving addition of references that support the Core Operating Limits Report (COLR). The specific proposed changes are provided below.

TS 5.6.5.b, CORE OPERATING LIMITS REPORT (COLR)

This section is revised to include modifications to existing references and additional references that reflect the proposed changes above. Most of the additional references describe the analytical methods used in determining core limits that are applicable to the Advanced Mark-BW fuel product. The following changes supersede the changes proposed for TS 5.6.5.b in Reference 1:

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

1 Framatome ANP, Inc. will be identified simply as "Framatome" for the remainder of this document.

13. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."

14. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."

15. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."

16. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

17. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."

18. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).

In addition, there are several places in the bases that require a revision to accommodate the RLBLOCA methodology. The use of RLBLOCA methodology is an allowable alternative to 10CFR50 Appendix K deterministic LOCA. As the TS bases were originally written to accommodate the Appendix K framework, they must now be revised to reflect the probabilistic nature of the RLBLOCA methodology. 10CFR50.46 paragraph (a)(1)(i) allows for the use RLBLOCA methods in place of Appendix K provided that the RLBLOCA methodology employed demonstrates that the LOCA analysis criteria will be met with a high degree of probability. As shown in the attached markup, the statements requiring that the peak cladding temperature during an LBLOCA must remain below 2200°F have been revised to read:

"During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks."

These associated bases changes are submitted for information only.

SAFETY SIGNIFICANCE

Proposed TS 5.6.5.b references 1, 13, 14, and 15 comprise a group of documents that was previously included and justified in Reference 1. The applicability of the methodologies to reload design is demonstrated through evaluations previously provided to the NRC in Reference 1.

Proposed TS 5.6.5.b reference 16 documents the Framatome Realistic Large Break LOCA (RLBLOCA) methodology. Use of this methodology will address NRC concerns regarding the LOCA analyses performed for the fuel transition. The Topical Report EMF-2103 (P), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," is approved by the NRC (Reference 4). Framatome has demonstrated this methodology to be applicable to 3 and 4-loop Westinghouse plants and, as such, the methodology may be applied in establishing reload design parameters limits for North Anna Power Station Units 1 and 2.

Proposed TS 5.6.5.b reference 17 documents a methodology for reactor core physics analysis and power distribution behavior. The Framatome RLBLOCA methodology inherently requires the use of certain inputs that are generated with such applicable physics methods. The topical report, EMF-96-029 (P)(A), "Reactor Analysis System for PWRs," has been added to provide another NRC-approved methodology applicable to Westinghouse three-loop reactors with 157 fuel assemblies with 17x17 fuel rod arrays. Per the qualification requirements of EMF-96-029 (P)(A), the applicability of the North Anna Units 1 and 2 SAV95 physics models has been demonstrated through benchmark analyses that comply with the appropriate SER requirements and limitations. This topical has been previously used in the analysis of a similar three-loop Westinghouse design reactor (H. B. Robinson). These assessments demonstrate that the methodology may be applied in support of RLBLOCA analyses for North Anna Power Station Units 1 and 2.

Proposed TS 5.6.5.b Reference 18 documents the Framatome Small Break LOCA (SBLOCA) methodology. Use of this methodology will address NRC concerns regarding the SBLOCA (reference 3) as it relates to the fuel transition. The Topical Report BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," is currently approved by the NRC. Framatome has demonstrated this methodology to be applicable to 3-loop Westinghouse plants using 17x17 fuel, dry containment design, and conventional ECCS; and as such, the methodology may be applied in establishing reload design parameter limits for North Anna Power Station Units 1 and 2.

CONCLUSIONS

The new references for TS 5.6.5.b described above have been demonstrated to be applicable for the establishment and support of reload design parameters for North Anna Power Station Units 1 and 2.

REFERENCES

1) Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes and Exemption Request, Use of Framatome ANP Advanced Mark-BW Fuel," Serial Number 02-167, March 28, 2002.

2) Letter from D. A. Christian to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Revised LOCA Analysis Schedule in Support of Proposed Technical Specifications Changes and Exemption Request to Use Framatome ANP Advanced Mark-BW Fuel," Serial Number 02-167E, November 15, 2002.

3) Letter from USNRC to D. A. Christian, "North Anna Power Station, Units 1 and 2 – Request for Additional Information on Small-Break LOCA Evaluation in Support of Proposed Technical Specifications Changes and Exemption Request to Use Framatome ANP Advanced Mark-BW Fuel (TAC NOS. MB4714 and MB4715)," Serial Number 03-245, dated March 21, 2003.

4) Letter from USNRC to J. F. Mallay (Framatome), "Safety Evaluation on Framatome ANP Topical Report EMF-2103 (P), Revision 0, 'Realistic Large Break Loss-of-Coolant Accident Methodology for Pressurized Water Reactors' (TAC NO. MB7554)," dated April 9, 2003.

Attachment 2

Framatome Fuel Transition Program

Mark-up of Proposed Technical Specifications Changes

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. (continued)

3. Moderator Temperature Coefficient,

4. Shutdown Bank Insertion Limits,

5. Control Bank Insertion Limits,

6. AXIAL FLUX DIFFERENCE limits,

7. Heat Flux Hot Channel Factor,

8. Nuclear Enthalpy Rise Hot Channel Factor,

9. Power Factor Multiplier,

10. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,

11. RCS Pressure, Temperature, and Flow DNB Limits, and

12. Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42~~,~~-A "Reload Nuclear Design Methodology."

2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."

3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."

4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."

7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

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

9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."

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

11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

Insert A

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

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

5.6.6 PAM Report

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

Insert A

13. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."

14. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."

15. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."

16. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

17. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."

18. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a ~~large break~~ loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F ~~(Ref. 1);~~ for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The Measured Heat Flux Hot Channel Factor, $F_Q^M(Z)$, shall be limited by the following relationships, as described in Reference 4:

$$F_Q^M(Z) \le \frac{CFQ}{P}\ \frac{K(Z)}{N(Z)} \quad \text{for } P > 0.5$$

$$F_Q^M(Z) \le \frac{CFQ}{0.5}\ \frac{K(Z)}{N(Z)} \quad \text{for } P \le 0.5$$

(continued)

BASES

LCO (continued)

where: CFQ is the $F_Q(Z)$ 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 a cycle dependent function that accounts for power distribution transients encountered during normal operation. N(Z) is included in the COLR; and

P is the fraction of RATED THERMAL POWER defined as

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of CFQ, K(Z), and N(Z) are given in the COLR; however, CFQ is normally approximately 2, K(Z) is a function that looks like the one provided in Figure B 3.2.1-1, and N(Z) is a value greater than 1.0.

An $F_Q^M(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value of $F_Q(Z)$. Then, the measured $F_Q^M(Z)$ is increased by 1.03 which is a factor that accounts for fuel manufacturing tolerances and 1.05 which accounts for flux map measurement uncertainty (Ref. 5).

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during ~~either~~ a ~~large or~~ small break LOCA, and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q^M(Z)$ limit to be exceeded before the next required $F_Q^M(Z)$ evaluation.

If the two most recent $F_Q^M(Z)$ evaluations show an increase in the expression

maximum over z $\left[\frac{F_Q^M(Z)}{K(Z)}\right]$,

it is required to meet the $F_Q^M(Z)$ limit with the last $F_Q^M(Z)$ increased by the appropriate factor, or to evaluate $F_Q^M(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit without detection.

REFERENCES

1. 10 CFR 50.46. ~~, 1974.~~

2. VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."

3. UFSAR, Section 3.1.22.

4. Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications, VEP-NE-1-A, March 1986.

5. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a ~~large break~~ loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F ~~;~~ for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

For transients that may be DNB limited, the Reactor Coolant System flow, temperature, and pressure, and $F^N_{\Delta H}$ are the parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to a value which provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation by using the movable incore detector system to obtain full core flux maps. Between these full core flux maps, the excore neutron detectors are used to monitor QPTR, which is a measure of changes in the radial power distribution. QPTR is defined in Section 1.1 in terms of ratios of excore detector calibrated output. However, the movable incore detector system can measure changes in the relative power of symmetrically located incore locations or changes in the incore tilt, which can be used to calculate an equivalent QPTR.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a ~~large break~~ loss of coolant accident, the peak cladding temperature must not exceed 2200°F, ~~(Ref. 1)~~;

(for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 2); and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

its associated Allowable Value and provided the trip setpoint "as-left" value is adjusted to a value within the calibration tolerance band of the nominal trip setpoint. A trip setpoint may be set more conservative than the nominal trip setpoint as necessary in response to unit conditions. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F); and

2. Boration to ensure recovery and maintenance of SDM.

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Reactor Trip;
- Turbine Trip;
- Feedwater Isolation;
- Start of all auxiliary feedwater (AFW) pumps;

(for small breaks, and a high level of probability that peak cladding temperature does not exceed 2200°F for large breaks (Ref. 10)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.10

SR 3.3.2.10 is the performance of a TADOT as described in SR 3.3.2.7, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB train cycle (RTB and associated bypass breaker must be opened at the same time). A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least one per refueling interval with applicable extensions.

This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 7.
3. UFSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. RTS/ESFAS Setpoint Methodology Study (Technical Report EE-0116).
7. NUREG-1218, April 1988.
8. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990 and WCAP-14333-P-A, Rev. 1, October 1998.
9. Technical Requirements Manual.
10. 10 CFR 50.46.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Head Safety Injection (HHSI) pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the HHSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LBLOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. For small breaks, the accumulator water volume only affects the mass flow rate of water into the RCS since the tanks do not empty for most break sizes analyzed. The assumed water volume has an insignificant effect upon the peak clad temperature. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The safety analysis supports operation with a contained water volume of between 7580 gallons and 7756 gallons per accumulator.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA

(continued)

for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, the accumulators do not discharge following a large main steam line break. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of two accumulators cannot be assumed to

for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the magnitude of post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the maximum flow requirement for the ECCS pumps. The HHSI pumps are credited in a small break LOCA event. This event relies upon the flow and discharge head of the HHSI pumps. The SGTR and MSLB events also credit the HHSI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one LHSI pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one Emergency Diesel Generator.

During the blowdown stage of a large break LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks

for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks

(continued)

Attachment 3

Framatome Fuel Transition Program

Proposed Technical Specification Changes

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. (continued)

3. Moderator Temperature Coefficient,

4. Shutdown Bank Insertion Limits,

5. Control Bank Insertion Limits,

6. AXIAL FLUX DIFFERENCE limits,

7. Heat Flux Hot Channel Factor,

8. Nuclear Enthalpy Rise Hot Channel Factor,

9. Power Factor Multiplier,

10. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,

11. RCS Pressure, Temperature, and Flow DNB Limits, and

12. Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."

3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."

4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."
7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY-REFERENCE CORE REPORT."
8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."
9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."
10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."
11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."
12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."
13. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."
14. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."
15. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."
16. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."
17. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."
18. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

5.6.6 PAM Report

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

5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC by March 1 of each year for the previous calender year. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The Measured Heat Flux Hot Channel Factor, $F_Q^M(Z)$, shall be limited by the following relationships, as described in Reference 4:

$$F_Q^M(Z) \le \frac{CFQ}{P}\frac{K(Z)}{N(Z)} \quad \text{for } P > 0.5$$

$$F_Q^M(Z) \le \frac{CFQ}{0.5}\frac{K(Z)}{N(Z)} \quad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ 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 a cycle dependent function that accounts for power distribution transients encountered during normal operation. N(Z) is included in the COLR; and

P is the fraction of RATED THERMAL POWER defined as

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of CFQ, K(Z), and N(Z) are given in the COLR; however, CFQ is normally approximately 2, K(Z) is a function that looks like the one provided in Figure B 3.2.1-1, and N(Z) is a value greater than 1.0.

An $F_Q^M(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value of $F_Q(Z)$. Then, the measured $F_Q^M(Z)$ is increased by 1.03 which is a factor that accounts for fuel manufacturing tolerances and 1.05 which accounts for flux map measurement uncertainty (Ref. 5).

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA, and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. An evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q^M(Z)$ limit to be exceeded before the next required $F_Q^M(Z)$ evaluation.

If the two most recent $F_Q^M(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \qquad \left[\frac{F_Q^M(Z)}{K(Z)}\right],$$

it is required to meet the $F_Q^M(Z)$ limit with the last $F_Q^M(Z)$ increased by the appropriate factor, or to evaluate $F_Q^M(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit without detection.

REFERENCES

1. 10 CFR 50.46.

2. VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient."

3. UFSAR, Section 3.1.22.

4. Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications, VEP-NE-1-A, March 1986.

5. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to unirradiated fuel is limited to 225 cal/gm and irradiated fuel is limited to 200 cal/gm (Ref. 1); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 2).

For transients that may be DNB limited, the Reactor Coolant System flow, temperature, and pressure, and $F^N_{\Delta H}$ are the parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to a value which provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation by using the movable incore detector system to obtain full core flux maps. Between these full core flux maps, the excore neutron detectors are used to monitor QPTR, which is a measure of changes in the radial power distribution. QPTR is defined in Section 1.1 in terms of ratios of excore detector calibrated output. However, the movable incore detector system can measure changes in the relative power of symmetrically located incore locations or changes in the incore tilt, which can be used to calculate an equivalent QPTR.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident, the peak cladding temperature must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

its associated Allowable Value and provided the trip setpoint "as-left" value is adjusted to a value within the calibration tolerance band of the nominal trip setpoint. A trip setpoint may be set more conservative than the nominal trip setpoint as necessary in response to unit conditions. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. <u>Safety Injection</u>

   Safety Injection (SI) provides two primary functions:

   1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to ≤ 2200°F for small breaks, and a high level of probability that peak cladding temperature does not exceed 2200°F for large breaks (Ref. 10); and

   2. Boration to ensure recovery and maintenance of SDM.

   These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

   - Phase A Isolation;
   - Reactor Trip;
   - Turbine Trip;
   - Feedwater Isolation;

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.10

SR 3.3.2.10 is the performance of a TADOT as described in SR 3.3.2.7, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB train cycle (RTB and associated bypass breaker must be opened at the same time). A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least one per refueling interval with applicable extensions.

This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 7.
3. UFSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. RTS/ESFAS Setpoint Methodology Study (Technical Report EE-0116).
7. NUREG-1218, April 1988.
8. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990 and WCAP-14333-P-A, Rev. 1, October 1998.
9. Technical Requirements Manual.
10. 10 CFR 50.46.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Head Safety Injection (HHSI) pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the HHSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LBLOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. For small breaks, the accumulator water volume only affects the mass flow rate of water into the RCS since the tanks do not empty for most break sizes analyzed. The assumed water volume has an insignificant effect upon the peak clad temperature. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The safety analysis supports operation with a contained water volume of between 7580 gallons and 7756 gallons per accumulator.

(continued)

BASES

LCO (continued)	For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.
APPLICABILITY	In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist. This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks. In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.
ACTIONS	<u>A.1</u> If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, the

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the magnitude of post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the maximum flow requirement for the ECCS pumps. The HHSI pumps are credited in a small break LOCA event. This event relies upon the flow and discharge head of the HHSI pumps. The SGTR and MSLB events also credit the HHSI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one LHSI pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one Emergency Diesel Generator.

(continued)

Attachment 4

Framatome Fuel Transition Program
Proposed Technical Specification Change

Significant Hazards Consideration Determination
Environmental Assessment

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

SIGNIFICANT HAZARDS CONSIDERATION

Virginia Electric and Power Company (Dominion) is requesting a revision to TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," to provide additional analytical methodologies to be used to determine acceptable core designs and provide inputs to develop the core operating limits contained in the COLR. These methodologies, the realistic LBLOCA (RLBLOCA) and the small LBLOCA (SBLOCA), will be used for analysis of the Framatome fuel product in the North Anna Power Station reactors. The proposed change adds references for the RLBLOCA topical report, a Framatome core design topical report, and a SBLOCA topical report.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of the standards as they relate to this supplementary amendment request follows:

1. The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased.

   The proposed methodology has been generically reviewed and approved for use by the NRC for determining core operating limits prior to its use by Dominion. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The core operating limits developed in accordance with the new methodologies will be bounded by any limitations in the NRC safety evaluation report (SER) for the new methodologies. Application of the topical reports associated with the new methodologies will demonstrate that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. The proposed changes do not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence of any accident previously evaluated is not significantly increased.

   The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed changes do not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose resulting from an accident. The proposed changes do not affect setpoints that initiate protective or mitigative actions. The proposed changes ensure that plant structures, systems, and components are maintained consistent with the safety analysis and licensing basis. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created.

   The proposed changes do not involve any physical alteration of plant systems, structures, or components, other than allowing for fuel design in accordance with NRC approved methodologies. The proposed methodologies continue to meet applicable criteria for LBLOCA and SBLOCA analyses. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized in response to plant transients changed. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced.

   The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within safety analyses. The proposed changes in the methodologies used in the LBLOCA and SBLOCA analyses do not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed changes in the analysis methodologies comply with the requirements of 10CFR50.46 paragraph (a)(1)(i) (i.e., not exceeding a peak cladding temperature of 2200°F for small break LOCA and a high probability that peak cladding temperature will remain below 2200° F for large break LOCA). Therefore, the margin of safety as defined in the Bases to the North Anna Units 1 and 2 Technical Specifications is not significantly reduced.

Based upon the above discussion Dominion has determined that the requested changes do not involve a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

These Technical Specification changes to allow the use of RLBLOCA methodology meet the eligibility criteria for categorical exclusion from an environmental assessment set forth in 10 CFR 51.22(c)(9), as discussed below:

(i) The license condition and associated exemptions from the Code of Federal Regulations involve no Significant Hazards Consideration.

As discussed in the attached evaluation of the Significant Hazards Consideration, the use of RLBLOCA and SBLOCA analysis methods at North Anna will not involve a significant increase in the probability or consequences of an accident previously evaluated. The possibility of a new or different kind of accident from any accident previously evaluated is not created, and the proposed use of RLBLOCA and SBLOCA methods does not involve a significant reduction in a margin of safety. Therefore, the proposed use of the RLBLOCA and SBLOCA analysis methods meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The RLBLOCA and SBLOCA analysis methods continue to meet applicable criteria for LBLOCA and SBLOCA analyses respectively. The application of the topical reports associated with the new methodologies will demonstrate that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. Therefore, no effect on the isotopic levels in the coolant will result from the use of RLBLOCA and SBLOCA analysis methodologies, and no effect on normal operating plant releases will occur. It is concluded that the existing radiological consequences analyses remain applicable for operation of North Anna with the use of RLBLOCA and SBLOCA analysis methods. Therefore, use of RLBLOCA and SBLOCA methods for analyzing the Advanced Mark-BW fuel will not significantly change the types, or significantly increase the amounts, of effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

Analysis of the Framatome fuel product using RLBLOCA and SBLOCA methods will not affect the plant operating conditions. Cycle specific reload evaluations will verify that fuel rod design criteria are satisfied, ensuring that cladding integrity is maintained. Analysis of Advanced Mark-BW fuel using RLBLOCA and SBLOCA methods will not significantly increase radiation levels compared to the current NAIF fuel, so individual and cumulative occupational exposures are unchanged.

Based on the above, the proposed use of the RLBLOCA and SBLOCA analysis methods for Advanced Mark-BW fuel does not have a significant effect on the environment, and meets the criteria of 10 CFR 51.22(c)(9). It is concluded that the proposed Technical Specification changes qualify for a categorical exclusion from a specific environmental review by the Commission, as described in 10 CFR 51.22.