VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

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United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 95-608
NL&P/MAE R1
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
30-DAY REPORT OF ECCS EVALUATION MODEL CHANGES
PER REQUIREMENTS OF 10CFR50.46

Pursuant to 10CFR50.46(a)(3)(ii) Virginia Electric and Power Company is providing information concerning changes to the ECCS Evaluation Models and their application in existing licensing analyses. Information is also provided which quantifies the effect of these changes upon reported results for North Anna Power Station, and demonstrates continued compliance with the acceptance criteria of 10CFR50.46. In our February 7, 1995 letter (Serial No. 94-436A) we committed to a reanalysis of the small break LOCA. The results of this reanalysis, as well as a reanalysis of the large break LOCA, are included in this report as the revised analysis of record (AOR).

Attachment 1 contains an excerpt of a Westinghouse report describing a specific evaluation model change to the Westinghouse Large Break LOCA ECCS Evaluation Model.

Attachment 2 provides a report describing plant-specific evaluation model changes associated with the application of the large break and small break LOCA evaluation models for the North Anna units.

Information regarding the effect of these ECCS Evaluation Model changes upon the LOCA AOR results is provided for the North Anna Power Station in Attachment 3. To summarize the information in Attachment 3, the calculated peak clad temperatures (PCTs) for the small and large break LOCA analyses for North Anna are given below.

North Anna Units 1 and 2 - Small break: 1704°F
North Anna Units 1 and 2 - Large break: 2053°F

The results represent significant changes, based on the criterion established in 10CFR50.46(a)(3)(i).

A001

We have evaluated these issues and the associated changes in the applicable licensing basis PCT results. These results demonstrate compliance with the requirements of 10CFR50.46(b). Although the North Anna large break and small break LOCA changes described in Attachment 2 are significant, the licensing basis PCT results have adequate margin to the limit. No further action is required to demonstrate compliance with 10CFR50.46 requirements.

If you have further questions or require additional information, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments:

1) Westinghouse Report of a Large Break LOCA ECCS Evaluation Model Change - North Anna Units 1 and 2

2) Report of Changes in Application of ECCS Evaluation Models - North Anna Units 1 and 2

3) Effect of ECCS Evaluation Model Changes - North Anna Units 1 and 2

cc: U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. R. D. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

ATTACHMENT 1

WESTINGHOUSE REPORT OF A LARGE BREAK LOCA
ECCS EVALUATION MODEL CHANGE

NORTH ANNA UNITS 1 AND 2

BASH ACCUMULATOR EMPTY FLAG

BACKGROUND

The error relates to a mismatch in the use of an end-of-accumulator-injection flag which is passed between the REFILL module and the BASH module. This flag in turn affects the application of the steam/water mixing model for safety injection into the cold leg, and is generally manifested as a momentary perturbation in system flows and pressures at the time at which the intact loop accumulator empties.

AFFECTED MODEL

1981 LBLOCA Evaluation Model with BASH v11.0

ESTIMATED EFFECT

Based upon the analysis of several representative plant cases, the net effect is estimated to be in the range of ±10 °F in the predicted peak clad temperature. For margin tracking purposes, a peak clad temperature penalty of +10 °F has been assessed for analyses performed with this version of the BASH code.

ATTACHMENT 2

REPORT OF CHANGES IN
APPLICATION OF ECCS EVALUATION MODELS

NORTH ANNA UNITS 1 AND 2

Revised Large Break and Small Break LOCA Analysis

North Anna Units 1 and 2

1.0 Background

This report provides a summary of changes in LOCA analysis results from those last reported for North Anna Units 1 and 2 (1). These changes are described in Section 2.0 below. It has been concluded that these changes are significant, as defined in 10 CFR 50.46(a)(3)(i).

2.0 Evaluation Model Changes

2.1 Revised Large Break LOCA Analysis (North Anna Units 1 and 2)

Since our previous 10CFR50.46 report (1), a revised analysis of the large break LOCA transient has been performed for North Anna Units 1 and 2. This revised analysis has been implemented as the analysis of record via a station 10CFR50.59 evaluation (2), consistent with the provisions of North Anna Technical Specification 6.9.1.7 (relating to the Core Operating Limits Report). This discussion summarizes the changes incorporated in this analysis, assumptions have been made which reflect operation with the replacement steam generators, and upflow and downflow configurations in addition to changes in other key analysis inputs.

Previous Westinghouse large break LOCA sensitivity studies for a three loop plant similar to North Anna have indicated that the downflow configuration has PCT results which bound that of the upflow design. However, since these results can be somewhat variable between different plants, the North Anna reanalysis was performed for both upflow and downflow configurations. Results and limitations associated with this analysis are applicable to the operation of North Anna Units 1 and 2. The key changes in assumptions from the prior analysis are listed below and discussed further in the following paragraphs.

- Improved BASH Evaluation Model codestream

- Assumption of 7% uniform steam generator tube plugging

- Peak Heat Flux Hot Channel Factor, F(Q), of 2.19 (current limit)

- Peak Enthalpy Hot Channel Factor, $F\Delta h$ of 1.60 (changed from 1.55)

- Hot Assembly Relative Power Factor of 1.45 (changed from 1.40)

- Upflow and downflow baffle/barrel designs

- Hot Leg Nozzle Gap flow is modelled during reflood

- North Anna Improved Fuel (NAIF) with ZIRLO™ cladding and PERFORMANCE+ design features

- Incorporation of skewed axial power distribution evaluation methodology

This analysis was performed using the Westinghouse 1981 large break LOCA evaluation model with BASH (3). Technical Specification 6.9.1.7 lists this as an acceptable reference methodology for determination of relevant power distribution limits in the Core Operating Limits Report.

An improvement to the BASH Evaluation Model codestream is described in Reference (4). The improved codestream provides for an interactive calculation between the BASH and COCO codes. The containment pressure methodology during the blowdown phase of the transient has not been changed. The refill transient portion of the WREFLOOD code, which calculates RCS behavior during vessel lower plenum refill following the end of blowdown, has been reprogrammed as a separate, but identical code (REFILL), which also runs interactively with the COCO code.

A further improvement in the BASH Evaluation Model codestream is described in Reference (5). With the improved codestream, the REFILL and LOCTA codes have been incorporated directly into the BASH code as subroutine modules. This eliminates all external transfer of data between these codes. In conjunction with this merging of codes, efforts were made to minimize any remaining code-to-code data transfer and to streamline and optimize some internal operations in the coding. The newly combined codes are configured as a single code which is identified as a new version of BASH.

With the improvements to the BASH codestream in References (4) and (5), no changes have been made to any of the approved physical models or basic techniques which form the basis of the methodology. However, these revisions to the BASH codestream can produce small changes to the results calculated by the improved codestream.

Large Break LOCA analyses have been traditionally performed using a symmetric, chopped cosine, core axial power distribution. In Reference (6), Westinghouse informed the NRC of the withdrawal of the Westinghouse Power Shape Sensitivity Model (PSSM) topical (Reference 7) effective October 30, 1995. This power shape methodology had been employed to support Reload Safety Evaluations (RSEs). Westinghouse further indicated that future large break LOCA analysis with the 1981 model with BASH would incorporate the explicit analysis approach to skewed power shapes as described in Reference (8). The analysis described herein employs the Reference (8) explicit analysis methodology.

The analysis assumed an RCS total flowrate of 278,400 gpm. Reference (9) evaluated the effect of a reduction in RCS flow from 278,400 gpm to 269,800 gpm. It was concluded therein that the large break LOCA results are insensitive to this magnitude of change in the RCS flow. Therefore, the analysis results are applicable to the reduced RCS flow of 269,800 gpm.

The analysis employs the Westinghouse methodology for incorporating the hot leg nozzle gaps into BASH (10). In a standard Westinghouse plant design, a relatively small gap exists at cold conditions between the core barrel and the reactor vessel at the hot leg nozzles, to allow for installation and removal of the lower internals assembly independent of the vessel. This gap diminishes at hot isothermal conditions due to the difference in thermal expansion between the core barrel and the vessel. Following a LBLOCA, the core barrel and vessel temperatures decrease significantly. The core barrel cools more quickly since it is covered by water on two sides and is thinner than the vessel. Hence, the gap between the outlet nozzle and core barrel expands, and is larger than the cold as-built dimension. This gap provides a flow path for the vapor in the upper plenum to be relieved through the break. This flowpath can be modelled in BASH within the existing capabilities of the code through a simple change to the standard nodalization of the approved Evaluation Model. The use of this methodology is a change to the BASH Evaluation Model (EM), and has been implemented without prior NRC review consistent with Westinghouse practice for this issue on other plants. The process for addressing model changes is documented in WCAP-13451 (11).

The analysis assumed a peak Heat Flux Hot Channel Factor, FQ(z), value of 2.19 and a peak Nuclear Enthalpy Hot Channel Factor, FΔh, value of 1.60. As required by Technical Specification 6.9.1.7, the Core Operating Limits Report (COLR) documents the applicable limit values of key core-related parameters for each reload core. These values bound the limits in the current cycle specific COLR's. For future reload cycles, the COLR will specify the appropriate limits which account for all design considerations, particularly large and small break LOCA effects.

The analysis assumes a full core of North Anna Improved Fuel (NAIF) with ZIRLO™ cladding and PERFORMANCE+ design features, which is similar and compatible to Westinghouse Vantage 5 Hybrid (V5H) fuel and the 17x17 Standard fuel. This modeling is applicable to full or mixed cores of either fuel product. The only mechanism available to cause a transition core to have a greater calculated large break LOCA PCT than a full core of either fuel product is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. It has been shown that the mixed core hydraulic resistance mismatches are not a significant factor, and it is not necessary to apply a LOCA analysis transition core penalty.

Employing these assumptions in the current version of the 1981 ECCS Evaluation Model with BASH, it has been demonstrated that operation at an assumed core thermal power of 2893 MWt with SGTP up to 7% in any SG will comply with all of the acceptance criteria specified in 10 CFR 50.46. Attachment 3 provides the PCT result for the revised analysis of record, in conjunction with appropriate margin assessments which address BASH evaluation model issues.

The revised analysis of record PCT is 2013°F for the limiting baffle/barrel configuration (downflow). This result does not include the PCT adjustment associated with the issue discussed in Attachment 1. This additional effect is presented on the summary table in Attachment 3. Since the revised analysis of record is more than 50°F different from the

existing analysis of record, implementation of this analysis represents a significant change, as defined in 10CFR50.46(a)(3)(i). The resulting licensing basis PCT demonstrates that operation at the rated thermal power of 2893 MWt will comply with all of the acceptance criteria specified in 10CFR50.46. Attachment 3 provides the PCT result for the revised analysis of record, in conjunction with appropriate margin assessments which address BASH evaluation model issues.

2.2 Revised Small Break LOCA Analysis (North Anna Units 1 and 2)

Since our previous 10CFR50.46 report (1), a revised analysis of the small break LOCA transient has been performed for North Anna Units 1 and 2. This revised analysis has been implemented as the analysis of record for both units via a station 10CFR50.59 evaluation (2), consistent with the provisions of North Anna Technical Specification 6.9.1.7 (relating to the Core Operating Limits Report).

This discussion presents the results of a revised small break LOCA transient for North Anna Unit 1 operation with the upflow modification. Since prior Westinghouse and Virginia Power small break LOCA analyses have demonstrated that the upflow configuration bounds downflow, this analysis models only the Unit 1 upflow design. The key analysis input changes required to provide acceptable results for the current small break LOCA analysis are listed below and discussed further in the following paragraphs.

- Assumption of 7% uniform steam generator tube plugging
- Peak Heat Flux Hot Channel Factor, F(Q), of 2.32* (changed from 2.20)
- Peak value for Enthalpy Hot Channel Factor, $F\Delta h$ of 1.65* (changed from 1.55)
- A minimum delivered HHSI flow rate calculated for LOCA analysis
- A full core of North Anna Improved Fuel (NAIF) with ZIRLO™ cladding and PERFORMANCE+ features (bounds operation with 17x17 Standard and NAIF mixed cores)
- Upflow baffle/barrel design (from Downflow baffle/barrel design)
- Safety Injection in all loops (from safety injection occurring in intact loop only)
- COSI Condensation Model (from homogeneous equilibrium condensation model)

This analysis was performed using the 1985 Small Break LOCA Evaluation Model with NOTRUMP (12). Technical Specification 6.9.1.7 lists this as an acceptable reference

methodology for determination of relevant power distribution limits in the Core Operating Limits Report.

In the previous NOTRUMP evaluation model, safety injection is delivered only in the intact loop, and the least resistance safety injection line is assumed to spill on the containment floor. This modeling was assumed to be conservative since the additional safety injection was considered to be a benefit. This assumption was based on older models which employed a homogenous equilibrium assumption for the mixing of different phases. Sensitivity studies with the previous evaluation model determined that safety injection in the broken loop, in conjunction with the existing condensation model, resulted in a PCT penalty. Reference (13) has documented this change to the NOTRUMP evaluation model. This modeling is used in this reanalysis.

To offset the penalties associated with the revised safety injection assumption, Westinghouse has incorporated a new condensation model in the NOTRUMP evaluation model. This model, referred to as the COSI model, is based on tests which modelled the configuration of the SI piping to the RCS cold leg. Use of this more realistic model for condensation of steam by pumped SI is demonstrated to provide a benefit larger than any penalty associated with injecting into the broken loop (13).

The analysis assumed a peak Heat Flux Hot Channel Factor, FQ(z), value of 2.32 and a peak Nuclear Enthalpy Hot Channel Factor, FΔh, value of 1.65. As required by Technical Specification 6.9.1.7, the Core Operating Limits Report (COLR) documents the applicable limit values of key core-related parameters for each reload core. These analytical values bound the limits in the current cycle specific COLR's.

A new revised hot rod axial power shape was used in the analysis. This power shape was chosen from a generic database of potential shapes achievable during power operation by assessing the characteristics which yield limiting small break LOCA results. The selected shape has been identified as the most limiting within the bounds of the proposed K(z) curve.

The flow rates for the HHSI are provided by an engineering model of the HHSI subsystem that is based on the system configuration and measured data from the plant. This model includes allowances for imbalance between the separate injection lines, HHSI pump degradation, and instrument accuracy. The HHSI pump curves used in the model are based on the actual measured plant data for the installed HHSI pumps in each unit. For the calculated HHSI flows, it is assumed that the HHSI flow recirculation line is open above RCS pressures of 1000 psig and that it is closed below that RCS pressure. This is consistent with previous assumptions used to calculate HHSI flow rates versus RCS pressure for small break LOCAs. Other assumptions regarding HHSI system configuration, such as water levels and back pressures, are set to provide limiting conditions for the specified test condition. HHSI flow testing performed during refueling outages can be used to assess the condition of the HHSI pumps and maintain the required flow balance between the individual HHSI injection lines to meet small break LOCA requirements.

The analysis assumes a full core of North Anna Improved Fuel (NAIF) with ZIRLO™, PERFORMANCE+, which is similar and compatible to Westinghouse Vantage 5 Hybrid (V5H) fuel. This modeling is applicable to full or mixed cores of either fuel product. The only mechanism available to cause a transition core to have a greater calculated small break LOCA PCT than a full core of either fuel product is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. The small break evaluation model assumes only one core channel. This assumption is acceptable, since the flowrate during a small break LOCA is low, providing enough time to maintain flow equilibrium and eliminate crossflow effects. As stated in Reference (14), mixed core hydraulic resistance mismatches are not a significant factor for small break analyses and it is not necessary to apply a LOCA analysis transition core penalty.

For this analysis of the small break LOCA, cases were run assuming 2 inch, 3 inch, and 4 inch effective diameter cold leg breaks. The 6 inch break case was not run but previous evaluations have demonstrated its peak clad temperature (PCT) to be less limiting than the 2, 3, and 4 inch breaks. The 6 inch break produces a more rapid depressurization and accumulator actuation, which results in primary core recovery sooner than for the smaller break cases. The PCT occurs during this initial, deep uncovery. Analyses of 6 inch break cases also typically exhibit a second, more shallow core uncovery, but fuel rod heatup is limited during this period by three factors: 1) greater accumulator and safety injection flows limit the uncovery to the top portion (approximately 2 feet) of the core, 2) the larger break size allows more energy removal from the core and 3) the duration of the second uncovery is ultimately limited by significant additional flow from the low head safety injection pumps, which provide for full core recovery. The 3 inch cold leg break was found to be the most limiting break size for the small break LOCA.

The revised analysis of record PCT is 1704°F. Since this result is more than 50°F different from the existing analysis of record, implementation of this analysis represents a significant change, as defined in 10CFR50.46(a)(3)(i). The resulting licensing basis PCT demonstrates that operation at the rated thermal power of 2893 MWt will comply with all of the acceptance criteria specified in 10CFR50.46. Attachment 3 provides the PCT result for the revised analysis of record.

3.0 References

(1) Letter from J. P. O'Hanlon (Va. Electric & Power Co.) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Report of ECCS Evaluation Model Changes Pursuant to Requirements of 10CFR50.46" Serial No. 95-198, April 25, 1995.

(2) North Anna Station 10CFR50.59 Safety Evaluation, 95-SE-OT-035, "North Anna Power Station Units 1 and 2 - Safety Evaluation for Revised Large Break Loss of Coolant Analysis (LBLOCA) and Small Break Loss of Coolant Analysis (SBLOCA)", October 31, 1995.

(3) WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model using the BASH Code," March 1987.

(4) Letter from N. J. Liparulo (Westinghouse) to USNRC, "Change in Methodology for Execution of BASH Evaluation Model," NTD-NRC-94-4143, May 23, 1994.

(5) Letter from N. J. Liparulo (Westinghouse) to USNRC, "Change in Methodology for Execution of BASH Evaluation Model," NTD-NRC-95-4540, August 29, 1995.

(6) Letter from N. J. Liparulo (Westinghouse) to USNRC, "Withdrawal of WCAP-12909-P on Power Shape Sensitivity Model (PSSM)," NTD-NRC-95-4518, August 7, 1995.

(7) WCAP-12909-P, "Westinghouse ECCS Evaluation Model: Revised Large Break LOCA Power Distribution Methodology," June 1991.

(8) WCAP-10266-P-A, Addendum 1, Revision 2-P-A, "The 1981 Version of Westinghouse ECCS Evaluation Model Using the BASH Code, Addendum 1; Power Shape Sensitivity Studies," March, 1987.

(9) Letter from W. L. Stewart (VEPCO) to the USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 1, Proposed Technical Specification Change, Reduced Minimum RCS Flow Rate Limit to Support Increased Steam Generator Tube Plugging Level," Serial No. 92-018, January 8, 1992.

(10) Letter from N. J. Liparulo (Westinghouse-Manager, Nuclear Safety & Regulatory Activities) to USNRC, NTD-NRC-95-4477, July 26, 1995, "Transmittal of Topical Reports WCAP-14404-P and WCAP-14405-NP, "Methodology for Incorporating Hot Leg Nozzle Gaps into BASH."

(11) Letter from N. J. Liparulo (Westinghouse-Manager, Nuclear Safety & Regulatory Activities) to USNRC, "Westinghouse Methodology for Implementation of 10CFR50.46 Reporting," ET-NRC-92-3755, October 30, 1992; transmits WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting."

(12) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(13) WCAP-10054, 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," August 1994.

(14) WCAP-10444, Addendum 2, "Vantage 5H Fuel Assembly," April 1988.

ATTACHMENT 3

EFFECT OF ECCS EVALUATION MODEL CHANGES

NORTH ANNA UNITS 1 AND 2

Effect of ECCS Evaluation Model Changes - North Anna

The information provided herein is applicable to North Anna Power Station, Units 1 and 2. It is based upon reports from Westinghouse Electric Corporation for issues involving the ECCS evaluation models and plant-specific application of the models in the existing analyses. Peak cladding temperature (PCT) values and margin allocations represent issues for which permanent resolutions have been implemented. Section A presents the detailed assessment for small break LOCA. The large break LOCA details are given in Section B.

Section A - Small Break LOCA Margin Utilization - North Anna Units 1 and 2

A. PCT for Analysis of Record {1}	1704°F	(1)
B. Prior PCT Assessments Allocated to AOR	**0°F**	
SBLOCA Augmented PCT for AOR	1704°F	
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	**0°F**	
SBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	1704°F	

Section B - Large Break LOCA Margin Utilization - North Anna Units 1 and 2

A. PCT for Analysis of Record {2}	2013°F	(1)
B. Prior PCT Assessments Allocated to AOR	**40°F**	
1. LBLOCA/Seismic SG Tube Collapse {3}	+30°F	
2. BASH Accumulator Empty Flag {4}	+10°F	
LBLOCA Augmented PCT for AOR	2053°F	
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	**0°F**	
LBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	2053°F	

Notes { } and References ( ) on the following page

Notes:

{1} The previous SBLOCA AOR reported in the last annual report (Reference 2) was 1880°F. The associated Licensing Basis PCT reported was 2092°F.

{2} The previous LBLOCA AOR reported in the last annual report (Reference 2) was 2066°F. The associated Licensing Basis PCT reported was 2041°F.

{3} A generic steam generator LOCA/seismic load evaluation was performed by Westinghouse to quantify the potential steam generator tube collapse which may occur at the time of a LOCA due to combined LOCA and seismic loads. Based on this analysis, a PCT penalty of +30°F (equivalent to a total steam generator tube reduction of 5%) was allocated as a permanent assessment (Reference 3) for those plants which do not have a detailed analysis.

{4} The current report is the initial quantification of effects for this issue. The PCT assessment (+10°F) for this issue is the maximum penalty from Attachment 1.

References:

(1) North Anna Station 10CFR50.59 Safety Evaluation, 95-SE-OT-035, "North Anna Power Station Units 1 and 2 - Safety Evaluation for Revised Large Break Loss of Coolant Analysis (LBLOCA) and Small Break Loss of Coolant Analysis (SBLOCA)", October 31, 1995.

(2) Letter from J. P. O'Hanlon (Va. Electric & Power Co.) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Report of ECCS Evaluation Model Changes Pursuant to Requirements of 10CFR50.46" Serial No. 95-198, April 25, 1995.

(3) Letter from W. L. Stewart (VEPCO) to Document Control Desk (USNRC), "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Report of ECCS Evaluation Model Changes Per Requirements of 10CFR50.46," Serial No. 91-428, August 23, 1991.