VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 9, 1993

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

Serial No.	93-182B
NA&F/GLD-CGL	R1
Docket Nos.	50-280
	50-281
	50-338
	50-339
License Nos.	DPR-32
	DPR-37
	NPF-4
	NPF-7

Gentlemen:

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

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 Surry and North Anna Power Stations and demonstrates continued compliance with the acceptance criteria of 10CFR50.46.

Attachment 1 describes two issues reported by Westinghouse involving errors and/or changes to the small break loss of coolant accident (LOCA) evaluation model which are applicable to Surry and North Anna Power Stations. As indicated in Attachment 1, the small break LOCA model changes have been concluded to be significant, based upon the criterion established in 10CFR50.46(a)(3)(i). In addition to these generic issues, there were plant-specific changes associated with the application of the large break LOCA evaluation model for the North Anna units. Attachment 2 provides a report describing these plant-specific evaluation model changes. These changes are not significant, but are being reported at this time for convenience.

Information regarding the effect of the ECCS evaluation model changes upon the reported LOCA analysis of record (AOR) results is provided for Surry and North Anna Power Stations in Attachments 3 and 4, respectively. To summarize the information in

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Attachments 3 and 4, the calculated peak clad temperature (PCT) for the small and large break LOCA analyses for Surry and North Anna are given below. Results which include significant changes, as defined in 10CFR50.46(a)(3)(i), are designated with an asterisk.

Surry Units 1 and 2 - Small break: 1839°F (*)
Surry Units 1 and 2 - Large break: 2094°F
North Anna Units 1 and 2 - Small break: 1860°F (*)
North Anna Unit 1 - Large break: 2025°F
North Anna Unit 2 - Large break: 2060°F

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). The Surry and North Anna small break LOCA issues described in Attachment 1 are significant, as defined in 10CFR50.46(a)(3)(i). However, since the conclusion of the evaluation performed by Westinghouse is that the existing NOTRUMP small break LOCA analysis results remain bounding, no reanalysis is currently scheduled. The changes in the North Anna large break LOCA analysis results are not significant. The need for reanalysis to accommodate additional steam generator tube plugging will be reassessed prior to operation of North Anna 2, Cycle 11. No further action is required to demonstrate compliance with 10CFR50.46 requirements.

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

Very truly yours,

W. L. Stewart
Senior Vice President - Nuclear

Attachments:
1. Report of Westinghouse LOCA Evaluation Model Changes - Surry Units 1and 2 and North Anna Units 1 and 2
2. Report of Changes in Application of ECCS Evaluation Model - North Anna Units 1 and 2
3. Effect of ECCS Evaluation Model Changes - Surry Units 1 and 2
4. Effect of ECCS Evaluation Model Changes - North Anna Units 1 and 2

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ATTACHMENT 1

REPORT OF WESTINGHOUSE LOCA EVALUATION MODEL CHANGES - SURRY UNITS 1 AND 2 AND NORTH ANNA UNITS 1 AND 2

1.0 Background

This report provides a summary of changes in the NOTRUMP small break LOCA evaluation model as reported to Virginia Power by Westinghouse. These changes affect the analysis results from those last reported for Surry Units 1 and 2 (1) and North Anna Units 1 and 2 (2). These changes are described in Section 2.0 below. It has been concluded that these changes are significant, as defined in 10CFR50.46(a)(3)(i).

2.0 Evaluation Model Changes

2.1 NOTRUMP Drift Flux Flow Regime Map Errors

Errors were discovered in both WCAP-10079-P-A (3) and related coding in the portion of NOTRUMP where the improved TRAC-P1 vertical flow regime map is evaluated. This model is used in standard analyses only during counter-current flow conditions in vertical flow links. This change affects Equation G-65 which previously allowed for unbounded values of the parameter C_{∞}, contrary to the intent of the original source of this equation. This error allowed a discontinuity to exist in the flow regime map under some conditions. The correction involved placing an upper limit on the parameter C_{∞}, as reasoned from the discussion in the original source. This correction returned NOTRUMP to consistency with the original source for the affected equation.

An additional closely related logic error was also found in the affected NOTRUMP subroutine. This error, which could cause discontinuities in certain other circumstances, was also corrected and made consistent with the description in Reference (3).

This change, which affects the 1985 small break LOCA evaluation model (4), was determined to be a non-discretionary change as described in Section 4.1.2 of WCAP-13451 (5) and was corrected in accordance with Section 4.1.3 of that reference.

Westinghouse performed representative plant calculations which indicated PCT effects of this change which ranged from -13°F to -55°F. For tracking PCT effects, the minimum benefit of -13°F has been assigned to these changes. However, for reportability under 10CFR50.46(a)(3)(i), it has been demonstrated that the effect of these changes may exceed 50°F. Virginia Power is adopting the Westinghouse recommendation that these changes be considered significant with respect to 10CFR50.46(a)(3)(i) requirements.

This benefit is being allocated as a permanent assessment on the AOR results. Attachments 3 and 4 provide the PCT result for the revised AOR, reflecting the margin assessment from this evaluation.

2.2 Safety Injection in the Broken Loop

In Reference (6), Westinghouse identified the resolution of a potential issue concerning the modeling of safety injection (SI) flow into the broken RCS loop for small break LOCA. The existing small break LOCA evaluation model (4) models SI flow into the broken loop as spilling directly into the containment sump. This modeling is based upon the assumption that modeling SI flow into the broken RCS loop would result in a lower calculated PCT, since additional SI flow would be expected to provide additional core cooling. Results from recent evaluations indicate that modeling the SI flow into the broken loop will actually result in a significant increase in the PCT calculated with the Reference (4) model. The PCT increase occurs as a result of competition between the steam venting out the break and the SI to the broken loop, which also exits through the break. The competition between steam and SI results in higher RCS pressures and, thus, lower delivered SI flow rates to the intact RCS loops, which increases PCT. The calculated increase in PCT is significant with respect to 10CFR50.46(a)(3)(i) requirements.

Westinghouse has defined a benefit which more than offsets the increase in PCT from this issue. The benefit involves use of a newer conservative model for condensation of steam in the intact loops by SI water. The newer model is based on data obtained from the COSI test facility which is a 1/100 scale representation of the cold leg and SI injection ports in a Westinghouse-designed PWR. Use of the improved condensation model has demonstrated that the current NOTRUMP evaluation model analyses without the improved condensation model and no SI into the broken loop is more conservative (yields higher calculated PCT) than a case which includes SI into the broken loop and the improved condensation model.

Reference (6) stated Westinghouse's conclusion not to incorporate these changes into the current NOTRUMP evaluation model. Existing analyses performed with the NOTRUMP evaluation model remain valid, since they have a conservative PCT with respect to that which would be obtained from the revised model. In the notification to NRC of this issue (including the conclusions concerning 10CFR50.46 reporting requirements and model changes), Westinghouse stated the belief that the NRC notification by Westinghouse satisfies the reporting requirements of 10CFR50.46 without further reporting on the part of individual utilities (7).

In an October 5, 1993 telephone conference between Westinghouse and the NRC, Mr. Robert C. Jones, Jr., Chief of the Reactor Systems Branch of the Office of Nuclear Reactor Regulation, advised Westinghouse of the NRC position on this issue (7). Among the aspects of the NRC position reported in Reference (7) was the NRC conclusion that Westinghouse notification to the NRC does not relieve individual licensees of their requirements to report these changes as significant under 10CFR50.46(a)(3)(ii). The present letter serves the reporting requirement for this

NOTRUMP evaluation model issue, which Westinghouse has determined to be significant, per the definition in 10CFR50.46(a)(3)(i). Virginia Power will review this issue through our participation in the Westinghouse Owners' Group, which is evaluating the issue and the possible development of a generic program for resolution.

3.0 References

(1) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Report of ECCS Evaluation Model Changes and 30-Day Report Per Requirements of 10CFR50.46 - Surry Power Station Units 1 and 2," Serial No. 93-182, April 19, 1993.

(2) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Report of ECCS Evaluation Model Changes and 30-Day Report Per Requirements of 10CFR50.46 - North Anna Power Station Units 1 and 2," Serial No. 93-182A, July 16, 1993.

(3) "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

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

(5) "Westinghouse Methodology for Implementation of 10CFR50.46 Reporting," WCAP-13451, October 1992.

(6) Letter from H. A. Sepp (Westinghouse-Strategic Licensing Issues), "Safety Injection (SI) in the Broken Loop," NSAL-93-018, September 21, 1993; received via letter VRA-93-139, from D. R. Beynon (W Projects) to J. P. O'Hanlon, dated September 22, 1993.

(7) Letter from D. R. Beynon (W Projects) to J. P. O'Hanlon, "Surry Units 1 and 2, North Anna 1 and 2 - NRC Feedback on the Safety Injection in the Broken Loop," VRA-93-155, October 11, 1993.

ATTACHMENT 2

REPORT OF CHANGES IN APPLICATION OF ECCS EVALUATION MODEL - NORTH ANNA UNITS 1 AND 2

Extended Steam Generator Tube Plugging for Large Break LOCA Analysis and Large Break LOCA/Seismic Steam Generator Tube Collapse

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 not significant, as defined in 10CFR50.46(a)(3)(i). They are being reported here for convenience and to reset to zero the PCT changes presently accumulated toward the 50°F threshold.

2.0 Evaluation Model Changes

2.1 Extended SG Tube Plugging Evaluation for Large Break LOCA Analysis (Applicable to North Anna Unit 2 only)

Our previous 10CFR50.46 report (1), reported the results of a revised large break LOCA transient analysis for North Anna Units 1 and 2. This revised analysis was implemented as the analysis of record (AOR) by a station 10CFR50.59 evaluation, in conjunction with the provisions of North Anna Technical Specification 6.9.1.7 (relating to the Core Operating Limits Report). This analysis remains the analysis of record (AOR). It was performed with the 1981 evaluation model with BASH and assumed 20% uniform steam generator tube plugging (SGTP). Results and limitations associated with the AOR are applicable to the operation of North Anna Units 1 and 2. This report describes a change (applicable to North Anna Unit 2) which has been assessed for its effect upon the analysis of record PCT results.

North Anna Unit 2 has recently completed a refueling outage and has entered operation of Cycle 10. The 'C' steam generator total tube plugging after the outage exceeded the 20% value assumed in the large break LOCA AOR.

The following are the actual values of North Anna Unit 2 tube plugging for Cycle 10 operation:

Steam Generator A: 9.18%
Steam Generator B: 7.02%
Steam Generator C: 23.11%

The effects of this actual SGTP upon the large break LOCA AOR results have been assessed by incorporation into the 10CFR50.59 reload safety evaluation for operation of North Anna Unit 2, Cycle 10 (2). This approach is justified since the evaluation has shown that the requirements of 10CFR50.46(b) continue to be met and that no changes in Technical Specifications or evaluation models are required to demonstrate such compliance. The evaluation approach is summarized below.

Virginia Power has applied a direct PCT penalty of 10°F per % SGTP to the limiting case PCT from the large break LOCA analysis of record. The penalty magnitude conservatively represents prior North Anna results with the BASH evaluation model for SGTP in the range of 30-35%. The magnitude of the penalty associated with increased SGTP is highly dependent upon the level of SGTP and the PCT in the AOR. For example, an increase in SGTP from 30% to 35% would result in a larger expected change in PCT than an increase in SGTP from 20% to 25%. In addition, the penalty increases as the PCT approaches 2200°F and the contribution of zirconium/water reaction heat becomes significant. Since the 10°F per % increase in SGTP is based upon studies performed for North Anna with the BASH evaluation model for SGTP in the range of 30% to 35%, it is appropriate and conservative for use in evaluating SGTP increases in the region of 20%. Also, since the AOR PCT of 2044°F is less than the PCT obtained from the sensitivity cases for SGTP between 30% and 35%, the sensitivity employed adequately accounts for the zirconium/water reaction effects.

For this evaluation, a uniform SGTP of 23.5% has been assumed (2), which is 3.5% greater than the SGTP in the AOR. This bounds the actual SGTP in 'C' steam generator for North Anna Unit 2, Cycle 10 operation. Using the assumed PCT sensitivity of 10°F per % increase in uniform tube plugging, results in a 35°F PCT penalty for the effects of this change. This penalty is being allocated as a permanent assessment on the AOR results. The resulting licensing basis PCT demonstrates that operation at the rated thermal power of 2893 MWt with SGTP up to 23.5% in any steam generator will comply with all of the acceptance criteria specified in 10CFR50.46. Attachment 4 provides the revised large break LOCA licensing basis PCT, reflecting margin assessments from this evaluation.

2.2 Large Break LOCA/Seismic Steam Generator Tube Collapse

The existing North Anna Units 1 and 2 large break LOCA licensing basis PCT accommodates the effect of 0.57% uniform SGTP to account for potential steam generator tube collapse which may occur at the time of a LOCA due to combined LOCA and seismic loads. A Westinghouse site-specific steam generator LOCA/seismic load analysis has quantified this effect. When this issue was previously reported (3), it was addressed by allocating 0.57% SGTP margin (i.e., the limit upon actual SGTP was obtained by subtracting 0.57% from the value of SGTP assumed in the AOR). In order to more clearly reflect the treatment of the LOCA/seismic and North Anna Unit 2, Cycle 10 SGTP-related penalties, the method of accounting for the LOCA/seismic issue penalty has been changed. The PCT sensitivity used for calculating this penalty is the same as described above in Section 2.1 (10°F per % increase in SGTP). A permanent penalty of 6°F has been allocated for both North Anna units. This penalty appears in Attachment 4 and is included in the revised large break LOCA licensing basis PCT.

3.0 References

(1) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Report of ECCS Evaluation Model Changes and 30-Day Report Per Requirements of 10CFR50.46-North Anna Power Station Units 1 and 2," Serial No. 93-182A, July 16, 1993.

(2) "North Anna Unit 2 Cycle 10 Reload Safety Evaluation - Pattern RE," 10CFR50.59 Safety Evaluation 93-SE-OT-081, October 15, 1993.

(3) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "North Anna Power Station Units 1 and 2 Report of Errors/Changes in Application of ECCS Evaluation Models Per Requirements of 10CFR50.46 (30 Day Report)," Serial No. 92-091, February 10, 1992.

ATTACHMENT 3

EFFECT OF ECCS EVALUATION MODEL CHANGES - SURRY UNITS 1 AND 2

The information provided herein is applicable to Surry 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. 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 - Surry Units 1 and 2

A. PCT for AOR	1852°F (1)
B. PCT Assessments Allocated to AOR	0°F
1. NOTRUMP Drift Flux Flow Regime Map Errors {1}	- 13°F
SBLOCA Augmented PCT for AOR	1839°F
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	
1. Safety Injection in the Broken Loop {1}	0°F
SBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	1839°F

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

A. PCT for Analysis of Record (AOR)	1969°F (2)
B. PCT Assessments Allocated to AOR	
1. Fuel Rod Initial Condition Inconsistency	+ 25°F (3)
2. SG Tube Seismic/LOCA Assumption	+ 30°F (3)
3. 15x15 Grid Increased Pressure Drop	+ 26°F (4)
4. Evaluation of LHSI Flow Measurement	+ 69°F (5)
5. Structural Metal Heat Modeling	- 25°F (5)
LBLOCA Augmented PCT for AOR {2}	2094°F
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	0°F
LBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	2094°F

Notes { } and References ( ) on the following page.

Effect of Errors/Changes in Application of ECCS Evaluation Models -
Surry Units 1 and 2

Notes:

{1} The current report is the initial quantification of effects for this issue.

{2} Reference 6 advised the NRC that we are in the process of reanalyzing the large break LOCA for Surry Units 1 and 2 with completion planned by the end of first quarter of 1994.

References:

(1) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Surry Power Station Units 1 and 2 - Proposed Technical Specifications Changes - FΔH Increase/Statistical DNBR Methodology," Serial No. 91-374, July 8, 1991.

(2) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Surry Power Station Units 1 and 2 - Proposed Technical Specifications Change - Surry Improved Fuel Assembly, " Serial No. 87-188, May 26, 1987.

(3) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "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.

(4) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2- Report of ECCS Evaluation Model Changes Pursuant to Requirements of 10CFR50.46," Serial No. 92-560, August 31, 1992.

(5) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Report of ECCS Evaluation Model Changes and 30-Day Report Per Requirements of 10CFR50.46 -Surry Power Station Units 1 and 2," Serial No. 93-182, April 19, 1993.

(6) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Emergency Core Cooling System Analysis - Surry Power Station Units 1 and 2," Serial No. 93-642, October 20, 1993

ATTACHMENT 4

EFFECT OF ECCS EVALUATION MODEL CHANGES - NORTH ANNA UNITS 1 AND 2

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. 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.

<u>Section A - Small Break LOCA Margin Utilization - North Anna Units 1 and 2</u>

A. PCT for AOR	1873°F (1)
B. PCT Assessments Allocated to AOR	0°F
1. NOTRUMP Drift Flux Flow Regime Map Errors {1}	- 13°F
SBLOCA Augmented PCT for AOR	1860°F
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	
1. Safety Injection in the Broken Loop {1}	0°F
SBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	1860°F

<u>Section B - Large Break LOCA Margin Utilization - North Anna Units 1 and 2</u>

	Unit 1	Unit 2
A. PCT for Analysis of Record (AOR)	2044°F (2)	2044°F (2)
B. PCT Assessments Allocated to AOR		
1. 1993 - Structural Metal Heat Modeling	- 25°F (3)	- 25°F (3)
2. LBLOCA/Seismic SG Tube Collapse {1}	+ 6°F (4)	+ 6°F (4)
3. N2C10 Extended SGTP Evaluation {1}	n/a	+ 35°F (4)
LBLOCA Augmented PCT for AOR	2025°F	2060°F
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	0°F	0°F
LBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	2025°F	2060°F

Notes { } and References ( ) are on the following page.

Effect of Errors/Changes in Application of ECCS Evaluation Models
North Anna Units 1 and 2

Notes:

{1} The current report is the initial quantification of effects for this issue.

References:

(1) "North Anna Power Station Units 1 and 2 - Implementation of Extended SGTP Small Break LOCA Analysis," 10CFR50.59 Safety Evaluation 92-SE-OT-005, January 21, 1992.

(2) "North Anna Power Station Units 1 and 2 - Large Break LOCA Analysis for 20% SGTP," 10CFR50.59 Safety Evaluation 93-SE-OT-048, June 17, 1993.

(3) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Report of ECCS Evaluation Model Changes and 30-Day Report Per Requirements of 10CFR50.46 - North Anna Power Station Units 1 and 2," Serial No. 93-182A, July 16, 1993.

(4) "North Anna Unit 2 Cycle 10 Reload Safety Evaluation - Pattern RE," 10CFR50.59 Safety Evaluation 93-SE-OT-081, October 15, 1993.