VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 26, 1994

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

Serial No.	94-254A
NA&F/GLD-CGL	R0'
Docket Nos.	50-280
	50-281
License Nos.	DPR-32
	DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY 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 used in existing licensing analyses. Information is also provided which quantifies the effect of these changes upon reported results for Surry Power Station and demonstrates continued compliance with the acceptance criteria of 10CFR50.46.

Attachment 1 contains excerpted portions of a recent Westinghouse report describing three changes to the 1985 Small Break LOCA Evaluation Model which are applicable to Surry. Attachment 2 provides information regarding the effect of the ECCS Evaluation Model changes upon the reported LOCA results for the analysis of record for Surry Power Station. To summarize the information in Attachment 2, the calculated peak clad temperature (PCT) for the small and large break LOCA analyses for Surry are given below. The effect of the three small break LOCA evaluation model issues, when combined with the effects of outstanding issues, constitutes a significant change, as defined in 10CFR50.46(a)(3)(i) and is designated with an asterisk below.

Surry Units 1 and 2 - Small break: 1850°F (*)
Surry Units 1 and 2 - Large break: 2114°F

We have evaluated these issues and the associated changes in the applicable licensing basis PCT results. These results demonstrate continued compliance with the requirements of 10CFR50.46(b). Although the accumulation of changes since performing the Surry small break LOCA analysis of record is significant, the licensing basis PCT result has adequate margin to the limit. Therefore, no further reanalysis is

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scheduled nor is any other action required to demonstrate compliance with 10CFR50.46 requirements. The Surry large break LOCA result is not changed by this 30-day report and remains the same as that previously documented in our April 27, 1994 report by letter Serial No. 94-254.

If you have 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) Report of Westinghouse ECCS Evaluation Model Changes - Surry Units 1 and 2
2) Effect of ECCS Evaluation Model Changes - Surry Units 1 and 2

cc: U. S. Nuclear Regulatory Commission
Region II
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Mr. M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

ATTACHMENT 1

WESTINGHOUSE REPORT OF ECCS EVALUATION MODEL CHANGES (Changes to 1985 Small Break LOCA Evaluation Model)

SURRY UNITS 1 AND 2

BOILING HEAT TRANSFER CORRELATION ERRORS

Background

This closely related set of errors deals with how the mixture velocity is defined for use in various boiling heat transfer regime correlations. The previous definition for mixture velocity did not properly account for drift and slip effects calculated in NOTRUMP. This error particularly affected NOTRUMP calculations of heat transfer coefficient when using the Westinghouse Transition Boiling Correlation and the Dougall-Rohsenow Saturated Film Boiling Correlation.

In addition, a minor typographical error was also corrected in the Westinghouse Transition Boiling Correlation.

This was determined to be a Non-Discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Model

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant calculations for this issue resulted in the estimated PCT effect documented in the attached Margin Utilization Sheet.

STEAM LINE ISOLATION LOGIC ERRORS

Background

This error consists of two portions: a possible plant specific effect which only applies to analyses which assumed Main Feedwater Isolation (FWI) to occur on S-signal, and a generic effect applying to all previous analyses.

The possible plant specific effect was the result of incorrect logic which caused the main steam line isolation to occur on the same signal as FWI. Therefore, when the S-signal was chosen through user input to be the appropriate signal for FWI, it also caused the steam line isolation to occur on S-signal. This is inconsistent with the standard conservative assumption of steam line isolation on Loss of Offsite Power coincident with the earlier Reactor Trip signal.

The generic effect was the result of incorrect logic which always led to the isolation functions occurring at a slightly later time than when the appropriate signal was generated.

This was determined to be a Non-Discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Model

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant calculations for this issue resulted in the estimated PCT effect (+12°F for the plant specific portion, if applicable, and +18°F for the generic portion) documented in the attached Margin Utilization Sheet.

CORE NODE ZIRC OXIDE INITIALIZATION ERROR

Background

NOTRUMP models two regions for each core node analogous to the two (mixture and vapor) regions in adjoining fluid nodes. During the course of a transient, NOTRUMP tracks region specific quantities for each core node. Erroneous logic caused incorrect initialization of the region specific, fuel cladding zirc oxide thickness at times prior to the actual creation of the relevant region during the core boiliff transient.

This was determined to be a Non-Discretionary Change as described in Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

Affected Evaluation Model

1985 Small Break LOCA Evaluation Model

Estimated Effect

Representative plant calculations led to an estimated generic PCT effect of 0°F for this effect.

ATTACHMENT 2

EFFECT OF WESTINGHOUSE ECCS EVALUATION MODEL MODIFICATIONS

SURRY UNITS 1 AND 2

Effect of Westinghouse ECCS Evaluation Model Modifications - Surry

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. Peak clad 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 - Surry Units 1 and 2
(30-day report as required by 10CFR50.46(a)(3)(ii))

A. PCT for Analysis of Record (AOR)	1852°F (1)
B. Prior PCT Assessments Allocated to AOR	**- 13°F**
1. Safety Injection in the Broken Loop	0°F (2)
2. NOTRUMP Drift Flux Flow Regime Map Errors	- 13°F (2)
SBLOCA Augmented PCT for AOR	1839°F
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation {3}	**59°F**
1. Vessel & SG Calculation Errors in LUCIFER	- 16°F (4)
2. Hot Assembly Average Rod Burst Effects	+ 2°F (4)
3. Revised Burst Strain Limit Model	- 2°F (4)
4. SBLOCA Limiting Time in Life-Zirc/Water Oxidation	+ 15°F (4)
5. NOTRUMP Boiling Heat Transfer Correlation Errors {1}	- 6°F
6. NOTRUMP Steam Line Isolation Logic Errors {1} {2}	+ 18°F
7. NOTRUMP Core Node Zirc Oxide Initialization Error {1}	0°F
SBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	1850°F

Section B - Large Break LOCA Margin Utilization - Surry Units 1 and 2
(As previously reported in the April 27, 1994 report (letter Serial No. 94-254))

A. PCT for Analysis of Record (AOR)	2120°F (3)
B. Prior PCT Assessments Allocated to AOR	0°F
LBLOCA Augmented PCT for AOR	2120°F
C. PCT Assessments for 10CFR50.46(a)(3)(i) Accumulation	**6°F**
1. Vessel & SG Calculation Errors in LUCIFER	- 6°F (4)
2. LBLOCA Rod Internal Pressure Issues	0°F (4)
LBLOCA Licensing Basis PCT (AOR PCT + PCT Assessments)	2114°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} Refer to the Report of Westinghouse ECCS Evaluation Model Changes provided in Attachment 1.

{2} The generic assessment (18°F) applies to Surry. The AOR correctly modeled feedline isolation, making the plant-specific assessment (12°F) unnecessary.

{3} The accumulation of changes (sum of the absolute magnitudes) exceeds 50°F and is significant, as defined in 10CFR50.46(a)(3)(i).

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, "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," Serial No. 93-182B, November 9, 1993.

(3) "Surry Power Station Units 1 and 2 - Safety Evaluation for Revised Large Break LOCA Analysis," 10CFR50.59 Safety Evaluation 94-082, March 28, 1994.

(4) Letter from W. L. Stewart (Va. Electric & Power Co.) to NRC, "Report of ECCS Evaluation Model Changes Pursuant to Requirements of 10CFR50.46, Surry Power Station Units 1 and 2," Serial No. 94-254, April 27, 1994.