July 3, 2007

10 CFR 50.46

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - EMERGENCY CORE COOLING SYSTEM (ECCS) EVALUATION MODEL CHANGES - 30 DAY REPORT AND ANNUAL NOTIFICATION AND REPORTING FOR 2006

References:

1. Watts Bar Nuclear Plant (WBN) Unit 1 - Emergency Core Cooling System (ECCS) Evaluation Model Changes - April 12, 2006 - Annual Notification and Reporting for 2005
2. Westinghouse Letter WAT-D-11569 to TVA dated May 25, 2007 - 10CFR50.46 Annual Notification and Reporting for 2006
3. Westinghouse Letter WAT-D-11571 to TVA dated June 5, 2007 - 10CFR50.46 Reporting Text for HOTSPOT Fuel Relocation Error and Revised PCT Rackup Sheets

The purpose of this letter is to submit a 30-day report in accordance with 10 CFR 50.46(a)(3)(ii) for a calculated Peak Cladding Temperature (PCT) change of more than 50 degrees for Watts Bar Nuclear Plant. In Reference 3 above, Westinghouse Electric Company identified a change in the calculated PCT of more than 50 degrees involving an error resulting in a 65 degrees F penalty in the calculated PCT. With this change, the PCT is estimated to be 1845 degrees F, which is 355 degrees F below the 2200 degrees F regulatory limit. Due to the large margin remaining, no reanalysis or other actions are planned.

In addition to the error above, other identified changes or errors are summarized in Enclosure 1. The remaining identified changes or errors did not result in any additional penalties to the emergency core cooling system (ECCS) evaluation models used for WBN during 2006. Thus, this letter also serves as the annual report of changes or errors to the WBN ECCS evaluation models required by 10 CFR 50.46(a)(3)(ii).

There are no regulatory commitments associated with this submittal. If you have any questions concerning this matter, please call me at (423) 365-1824.

Sincerely,

Original signed by

J. D. Smith
Manager, Site Licensing
and Industry Affairs (Acting)

Enclosure
cc: See Page 2

JDS:RAS
Enclosures
cc (Enclosures):

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ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
EMERGENCY CORE COOLING SYSTEM (ECCS)
PEAK CLAD TEMPERATURE (PCT) ANNUAL REPORT
SUMMARY OF CHANGES

BEST ESTIMATE LARGE BREAK - CQD QUALIFICATION DOCUMENT (CQD) (1996) RELATED ITEMS

1. General Code Maintenance (Enhancements/Forward-Fit Discretionary Changes)

Background

A number of coding changes were made as part of normal code maintenance. Examples include additional information in code outputs, improved automation in the astrum codes, increased WCOBRA/TRAC code dimensions, and general code cleanup. All of these changes are considered to be discretionary changes in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The nature of these changes leads to an estimated PCT impact of 0 (zero) degree F.

2. HOTSPOT FUEL RELOCATION (Non-Discretionary Change)

Background

In the axial node where burst is predicted to occur, a fuel relocation model in HOTSPOT is used to account for the likelihood that additional fuel pellet fragments above that elevation may settle into the burst region. It was discovered that the effect of fuel relocation on local linear heat rate was being calculated, but then cancelled out later in the coding. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

1996 and 1999 BELOCA EMs analyses were assessed on a plant-specific basis, via the HOTSPOT reanalysis of a representative WCOBRA/TRAC case using the corrected code version at the burst elevation/burst model enabled sub-case. The HOTSPOT 95 percent probability PCT results were used to establish the plant-specific PCT penalty.

2004 ASTRUM EM analyses were assessed on a plant-specific basis, via the reanalysis of all of the burst cases from the original HOTSPOT calculations using the corrected HOTSPOT code version.

Plant-Specific Text

The plant-specific PCT penalty was 0 (zero) degree F for Reflood 1 and 65 degrees F for Reflood 2.

3. STEAM GENERATOR NOZZLE VOLUME ACCOUNTING ERROR (Non-Discretionary Change)

Background

It was discovered that many plant-specific WCOBRA/TRAC calculations shared a common error of double accounting of the volume of one or both SG Plenum Nozzles. The extent of over accounting is plant-specific but would be in the vicinity of 7-9 ft3 per nozzle. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

Estimated Effect

RCS Loop inventory does not significantly contribute to core cooling during blowdown since most of the fluid in both the intact and broken RCS loops will exit the break without entering the core, making RCS Loop volume a tertiary player in system behavior. A small volume error of this nature is anticipated to be negligible throughout the transient, such that an estimated effect of 0 (zero) degree F is assigned for 10 CFR 50.46 reporting purposes.

4. ERRORS IN REACTOR VESSEL NOZZLE DATA COLLECTIONS (Non-discretionary Change)

Background

Some minor errors were discovered in the reactor vessel nozzle data collections that potentially affect the vessel inlet and outlet nozzle fluid volume, metal mass, and surface area. The corrected values have been evaluated for impact on current licensing-basis analysis results and will be incorporated into the plant-specific input databases on a forward-fit basis. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

These errors were evaluated to have a negligible impact on the Large Break LOCA analysis results, leading to an estimated PCT impact of 0 (zero) degree F for 10 CFR 50.46 reporting purposes.

Appendix K Small Break - NOTRUMP RELATED ITEMS

1. General Code Maintenance (Enhancements/Forward-Fit Discretionary Changes)

Background

Various changes in code input and output format have been made to enhance usability and help preclude errors in analyses. This includes both input changes (e.g., more relevant input variables defined and more common input values used as defaults) and input diagnostics designed to preclude unreasonable values from being used, as well as various changes to code output which have no effect on calculated results. In addition, various updates were made to eliminate inactive coding, improve active coding, and enhance commenting, both for enhanced usability and to facilitate code debugging when necessary. These changes represent discretionary changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Models

1981 Appendix K Large Break LOCA Evaluation Model with BASH

1985 Appendix K Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The nature of these changes leads to an estimated PCT impact of 0 (zero) degree F.

2. NOTRUMP-EM Refined Break Spectrum

Background

During the course of reviewing several extended power uprate and replacement steam generator Small Break LOCA (SBLOCA) analyses, the Nuclear Regulatory Commission (NRC) questioned the break spectrum analyzed in the NOTRUMP evaluation model (EM). The NRC was concerned that the resolution of the break spectrum used in the NOTRUMP EM (1.5, 2, 3, 4, and 6 inch cases) may not be fine enough to capture the worst break with regard to limiting peak clad temperature as per 10 CFR 50.46. That is, the plant could be SBLOCA limited with regard to overall LOCA results. In response to this, Westinghouse performed some preliminary work indicating that in some cases more limiting results could be obtained from non-integer break sizes; however, the magnitude of the impact was far less than that shown in preliminary work performed by the NRC. Based on this, Westinghouse performed evaluations to determine if all currently operating plants would maintain compliance with the 10 CFR 50.46 acceptance criteria when considering a refined

SBLOCA break spectrum. It should be noted that use of a refined break spectrum is not an error, but a change, since evaluating only integer break sizes has been the standard practice since the initial licensing of NOTRUMP.

<u>Affected Evaluation Model(s)</u>

1985 Appendix K Small Break LOCA Evaluation Model

<u>Estimated Effect</u>

Consistent with the method described in Reference 1, for plants with low SBLOCA peak cladding temperatures (PCTs) (i.e., less than 1700°F) and overall SBLOCA results that are significantly non-limiting when compared with large break LOCA (LBLOCA) results, no explicit refined break spectrum calculations were performed, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes. For plants with high SBLOCA PCTs (i.e., equal to or greater than 1700°F), explicit refined break spectrum calculations were performed, and PCT penalties were assessed, if necessary.

<u>Reference</u>

1. LTR-NRC-06-44, "Transmittal of LTR-NRC-06-44 NP-Attachment, 'Response to NRC Request for Additional Information on the Analyzed Break Spectrum for the Small Break Loss of Coolant Accident (SBLOCA) NOTRUMP Evaluation Model (NOTRUMP EM), Revision 1,' (Non-Proprietary)," July 14, 2006.

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
EMERGENCY CORE COOLING SYSTEM (ECCS)
PEAK CLAD TEMPERATURE (PCT) ANNUAL REPORT
RACKUP SHEETS

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
EMERGENCY CORE COOLING SYSTEM (ECCS)
PEAK CLAD TEMPERATURE (PCT) ANNUAL REPORT
RACKUP SHEETS

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Cycle 8, RSG Composite

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 6/4/2007

Analysis Information

EM:	CQD (1996)	**Analysis Date:**	8/1/1998	**Limiting Break Size:**	Guillotine
FQ:	2.5	**FdH:**	1.65		
Fuel:	Vantage +	**SGTP (%):**	10		

Notes: Mixed Core - Vantage + / Performance + / RFA-2, RSG (12% SGTP)

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1892**	**1,2**	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . Vessel Channel DX Error	-4	3	
2 . MONTECF Decay Heat Uncertainty Error	4	6	
3 . Input Error Resulting in Incomplete Solution Matrix	0	7	
4 . Tavg Bias Error	8	7	
5 . Revised Blowdown Heatup Uncertainty Distribution	5	8	
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . Accumulator Line/Pressurizer Surge Line Data Evaluation	-131	4	
2 . Increased Accumulator Temperature Range Evaluation	4	5	
3 . 1.4% Uprate Evaluation	12	5	
4 . Increased Stroke Time for the ECCS Valves	0	9	
5 . Replacement Steam Generators (D3 to 68AXP)	-10	10	
C. 2007 ECCS MODEL ASSESSMENTS			
1 . HOTSPOT Fuel Relocation Error	65	11	
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	1845		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . WCAP-14839, Rev. 1, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant," August 1998.

2 . WAT-D-10499, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 1997," February 27, 1998.

3 . WAT-D-10618,"Tennessee Valley Authority, Watts Bar Nuclear Plant Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 1998," March 5, 1999.

4 . WAT-D-10725,"Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, 10 CFR 50.46 Annual Notification and Reporting for 1999," February 23, 2000.

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 6/4/2007

Cycle 8, RSG Composite

5 . WAT-D-10840, "Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, Final Deliverables for 1.4% Uprate Program," August 31, 2000.

6 . WAT-D-10904, "10 CFR 50.46 Annual Notification and Reporting for 2000," February 2001.

7 . WAT-D-11225, "10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.

8 . WAT-D-11334, "10 CFR 50.46 Annual Notification and Reporting for 2004, " April 2005.

9 . WAT-D-11285, "Evaluation of Proposed Changes to the Stroke Time for the ECCS Valves," November 2004.

10 . WTV-RSG-06-015, "LOCA & Non-LOCA Analysis Summary for Replacement Steam Generator," February 2006.

11 . LTR-LIS-07-378, "10 CFR 50.46 Reporting Text for HOTSPOT Fuel Relocation Error and Revised PCT Rackup Sheets for Watts Bar Unit 1," June 2007.

Notes:

None

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Cycle 8, RSG

Reflood 1

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 6/4/2007

Analysis Information

EM:	CQD (1996)	**Analysis Date:**	8/1/1998	**Limiting Break Size:**	Guillotine
FQ:	2.5	**FdH:**	1.65		
Fuel:	Vantage +	**SGTP (%):**	10		
Notes:	Mixed Core - Vantage + / Performance + / RFA-2, RSG (12% SGTP)				

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1656**	**1,2**	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . Vessel Channel DX Error	56	3	
2 . MONTECF Decay Heat Uncertainty Error	4	6	
3 . Input Error Resulting in Incomplete Solution Matrix	60	7	
4 . Tavg Bias Error	8	7	
5 . Revised Blowdown Heatup Uncertainty Distribution	5	8	
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . Accumulator Line/Pressurizer Surge Line Data Evaluation	-37	4	
2 . Increased Accumulator Temperature Range Evaluation	4	5	
3 . 1.4% Uprate Evaluation	12	5	
4 . Increased Stroke Time for the ECCS Valves	0	9	
5 . Replacement Steam Generators (D3 to 68AXP)	-50	10	
C. 2007 ECCS MODEL ASSESSMENTS			
1 . HOTSPOT Fuel Relocation Error	0	11	
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	1718		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . WCAP-14839, Rev. 1, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant," August 1998.

2 . WAT-D-10499, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 1997," February 27, 1998.

3 . WAT-D-10618,"Tennessee Valley Authority, Watts Bar Nuclear Plant Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 1998," March 5, 1999.

4 . WAT-D-10725,"Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, 10 CFR 50.46 Annual Notification and Reporting for 1999," February 23, 2000.

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Cycle 8, RSG

Reflood 1

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 6/4/2007

5 . WAT-D-10840, "Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, Final Deliverables for 1.4% Uprate Program," August 31, 2000.

6 . WAT-D-10904, "10 CFR 50.46 Annual Notification and Reporting for 2000," February 2001.

7 . WAT-D-11225, "10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.

8 . WAT-D-11334, "10 CFR 50.46 Annual Notification and Reporting for 2004, " April 2005.

9 . WAT-D-11285, "Evaluation of Proposed Changes to the Stroke Time for the ECCS Valves," November 2004.

10 . WTV-RSG-06-015, "LOCA & Non-LOCA Analysis Summary for Replacement Steam Generator," February 2006.

11 . LTR-LIS-07-378, "10 CFR 50.46 Reporting Text for HOTSPOT Fuel Relocation Error and Revised PCT Rackup Sheets for Watts Bar Unit 1," June 2007.

Notes:

None

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Cycle 8, RSG
Reflood 2

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 6/4/2007

Analysis Information

EM:	CQD (1996)	**Analysis Date:**	8/1/1998	**Limiting Break Size:**	Guillotine
FQ:	2.5	**FdH:**	1.65		
Fuel:	Vantage +	**SGTP (%):**	10		

Notes: Mixed Core - Vantage + / Performance + / RFA-2, RSG (12% SGTP)

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1892**	**1,2**	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . Vessel Channel DX Error	-4	3	
2 . MONTECF Decay Heat Uncertainty Error	4	6	
3 . Input Error Resulting in Incomplete Solution Matrix	0	7	
4 . Tavg Bias Error	8	7	
5 . Revised Blowdown Heatup Uncertainty Distribution	5	8	
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . Accumulator Line/Pressurizer Surge Line Data Evaluation	-131	4	
2 . Increased Accumulator Temperature Range Evaluation	4	5	
3 . 1.4% Uprate Evaluation	12	5	
4 . Increased Stroke Time for the ECCS Valves	0	9	
5 . Replacement Steam Generators (D3 to 68AXP)	-10	10	
C. 2007 ECCS MODEL ASSESSMENTS			
1 . HOTSPOT Fuel Relocation Error	65	11	
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	1845		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References:

1 . WCAP-14839, Rev. 1, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for the Watts Bar Nuclear Plant," August 1998.

2 . WAT-D-10499, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 1997," February 27, 1998.

3 . WAT-D-10618,"Tennessee Valley Authority, Watts Bar Nuclear Plant Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 1998," March 5, 1999.

4 . WAT-D-10725,"Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, 10 CFR 50.46 Annual Notification and Reporting for 1999," February 23, 2000.

Westinghouse LOCA Peak Clad Temperature Summary for Best Estimate Large Break

Cycle 8, RSG

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 6/4/2007

Reflood 2

5 . WAT-D-10840, "Tennessee Valley Authority, Watts Bar Nuclear Plant Unit 1, Final Deliverables for 1.4% Uprate Program," August 31, 2000.

6 . WAT-D-10904, "10 CFR 50.46 Annual Notification and Reporting for 2000," February 2001.

7 . WAT-D-11225, "10 CFR 50.46 Annual Notification and Reporting for 2003," March 2004.

8 . WAT-D-11334, "10 CFR 50.46 Annual Notification and Reporting for 2004, " April 2005.

9 . WAT-D-11285, "Evaluation of Proposed Changes to the Stroke Time for the ECCS Valves," November 2004.

10 . WTV-RSG-06-015, "LOCA & Non-LOCA Analysis Summary for Replacement Steam Generator," February 2006.

11 . LTR-LIS-07-378, "10 CFR 50.46 Reporting Text for HOTSPOT Fuel Relocation Error and Revised PCT Rackup Sheets for Watts Bar Unit 1," June 2007.

Notes:

None

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Small Break

Cycle 8, RSG

Plant Name: Watts Bar Unit 1
Utility Name: Tennessee Valley Authority
Revision Date: 5/7/2007

Analysis Information

EM:	NOTRUMP	**Analysis Date:**	5/17/2004	**Limiting Break Size:**	4 inch
FQ:	2.5	**FdH:**	1.65		
Fuel:	RFA-2	**SGTP (%):**	12		
Notes:	Mixed Core - Vantage + / Performance + / RFA-2				

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1132**	**1**	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . Increased Stroke Time for the ECCS Valves	0	2	
C. 2006 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . Leaking SIS Relief Valve	120	3	
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	1252		

* It is recommended that the licensee determine if these PCT allocations be considered with respect to 10 CFR 50.46 reporting requirements.

References

1 . WTV-RSG-06-015, "LOCA & Non-LOCA Analysis Summary for Replacement Steam Generator," February 2006.

2 . WAT-D-11285, "Evaluation of Proposed Changes to the Stroke Time for the ECCS Valves," November 2004.

3 . WAT-D-11360, "Safety Injection Pump Discharge Relief Valve Leakage Evaluation," July 2005.

Notes:

None