

WOLF CREEK NUCLEAR OPERATING CORPORATION

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Manager Regulatory Affairs

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RA 16-0025

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

- Reference: 1) Letter RA 15-0080, dated November 4, 2015, from C. R. Hafenstine, WCNOC to USNRC
- 2) Westinghouse Letter LTR-LIS-16-53, dated February 18, 2016, "Wolf Creek 10 CFR 50.46 Annual Notification and Reporting for 2015"

Subject: Docket No. 50-482: 10 CFR 50.46 Annual Report of Emergency Core Cooling System (ECCS) Evaluation Model Changes

Gentlemen:

In accordance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," paragraph (a)(3)(ii), Wolf Creek Nuclear Operating Corporation (WCNOC) is submitting the attached information to fulfill the annual reporting requirement for the Wolf Creek Generating Station (WCGS).

In Reference 1, WCNOC submitted a 30 Day Report of Emergency Core Cooling System (ECCS) Evaluation Model Changes to the Nuclear Regulatory Commission (NRC) to report changes of greater than 50°F in the peak cladding temperature (PCT) from those previously reported for a large break loss-of-coolant accident (LOCA). Those changes were due to implementation of a new best-estimate large break LOCA methodology known as Automated Statistical Treatment of Uncertainty Method (ASTRUM) that was approved by the NRC for WCGS.

WCNOC has reviewed Reference 2, which addresses 10 CFR 50.46 reporting information pertaining to the ECCS Evaluation Model changes that were implemented by Westinghouse for 2015. The review concludes that with the exception of the above changes that were already addressed in a 30 day ECCS model change report (Reference 1), the additional effect of changes to, or errors in, the Evaluation Models on the limiting transient PCT is not significant for 2015. Therefore, changes to the ECCS Evaluation Model are being reported as an annual report.


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Attachment I provides an assessment of the specific changes and enhancements to the Westinghouse Evaluation Models for 2015. Except for the exceptions noted above, these model changes and enhancements do not have impacts on the PCT and, generally, will not be presented on the PCT rack-up forms.

Attachment II provides PCT rack-up forms for the calculated Large Break LOCA and Small Break LOCA PCT margin allocations in effect for the 2015 WCGS Evaluation Models. The PCT values determined in the Large Break and Small Break LOCA analysis of record, combined with all of the PCT allocations, remain below the 10 CFR 50.46(b)(1) regulatory limit of 2200 °F. Therefore, WCGS is in compliance with 10 CFR 50.46 requirements and no reanalysis or other action is required.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4204 or Bill Muilenburg at 620-364-4186.

Sincerely,



Cynthia R. Hafenstine

CRH/rit

- Attachment I Assessment of Changes to the Westinghouse Emergency Core Cooling System (ECCS) Evaluation Models for Large and Small Break Loss-of-Coolant Accidents (LOCA)
- II Emergency Core Cooling System (ECCS) Evaluation Model Peak Cladding Temperature (PCT) Margin Utilization Rack-up Forms

cc: M. L. Dapas (NRC), w/a
C. F. Lyon (NRC), w/a
N. H. Taylor (NRC), w/a
Senior Resident Inspector (NRC), w/a

ASSESSMENT OF CHANGES TO THE WESTINGHOUSE EMERGENCY CORE COOLING SYSTEM (ECCS) EVALUATION MODELS FOR LARGE AND SMALL BREAK LOSS-OF-COOLANT ACCIDENTS (LOCA)

GENERAL CODE MAINTENANCE

Background

Various changes have been made to enhance the usability of codes and to streamline future analyses. Examples of these changes include modifying input variable definitions, units and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451 "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting."

Affected Evaluation Model

1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The nature of these changes leads to an estimated peak cladding temperature (PCT) impact of 0 °F.

LOWER SUPPORT PLATE, CORE BARREL, AND VESSEL WALL UNHEATED CONDUCTOR ERRORS

Background

Modeling errors were discovered in the lower support plate, core barrel, and vessel cladding unheated conductors in the Best-Estimate Large Break Loss-of-Coolant Accident (BELBLOCA) analysis-of-record. The modeling errors impacted the volume and surface area of the core barrel, the surface area and thermal resistance of the lower support plate, and the thermal resistance of the vessel wall.

The resolution of these issues represents a closely-related group of Non-Discretionary Changes in the application of the Evaluation Model as described in Section 4.1.2 of WCAP-13451.

Affected Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A qualitative evaluation was completed concluding that the modeled net stored energy and heat transfer rate of the vessel wall, core barrel, and lower support plate unheated conductors were adequate. This error is estimated to have a PCT impact of 0°F.

CORE CHANNEL GAP ERROR

Background

A modeling error was discovered in the BELBLOCA analysis-of-record. The modeling error over-represented the flow area between the guide tube and non-guide tube core average channels.

The resolution of this issue represents a Non-Discretionary Change in the application of the Evaluation Model as described in Section 4.1.2 of WCAP-13451.

Affected Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A qualitative evaluation was completed concluding the magnitude of the error is negligible. The error is estimated to have a PCT impact of 0°F.

EMERGENCY CORE COOLING SYSTEM (ECCS) EVALUATION MODEL PEAK CLADDING TEMPERATURE (PCT) MARGIN UTILIZATION RACK-UP FORMS

***** LARGE BREAK LOCA PCT MARGIN UTILIZATION *****

Evaluation Model:	ASTRUM (2004)
Fuel:	RFA-2
Peaking Factor:	FQ=2.50, FdH=1.65
SG Tube Plugging:	10%
Power Level:	3565 MWth
Limiting Break Size:	DEG

LICENSING BASIS

	Clad Temp (°F)	Ref.	Notes
Analysis of Record (AOR) PCT	1900 °F	1	

MARGIN ALLOCATIONS (ΔPCT)

A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS

1. Containment Fan Cooler Capacity	0	2	
2. Decay Group Uncertainty Factors Errors	-10	3	

B. PLANNED PLANT CHANGE EVALUATIONS

1. Containment Fan Cooler Capacity	0	2	(a)
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C. 2015 PERMANENT ECCS MODEL ASSESSMENTS

1. None	0		
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D. OTHER

1. None	0		
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LICENSING BASIS PCT + MARGIN ALLOCATIONS PCT = 1890 °F

CUMULATIVE ABSOLUTE MAGNITUDE OF PCT CHANGES $\Sigma |\Delta PCT| = 0 \text{ °F}$
SINCE LAST 30-DAY REPORT (LETTER RA 15-0080)

References:

1. WCAP-17107-P, Revision 1, "Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for the Wolf Creek Nuclear Power Plant Using the ASTRUM Methodology," January 2014.
2. LTR-LIS-14-400, "10 CFR 50.46 Report for the Wolf Creek Large Break LOCA Evaluation of the Change in Containment Cooling Capacity," August 2014.
3. LTR-LIS-14-492, "Wolf Creek Unit 1 10 CFR 50.46 Report for the Correction of the Decay Group Uncertainty Factors Errors," November 2014.

Notes:

- (a) This effect was estimated based on a cooling capacity intended to bound future implementation of replacement tube bundles in the containment fan coolers.

***** SMALL BREAK LOCA PCT MARGIN UTILIZATION *****

Evaluation Model:	1985 EM with NOTRUMP
Fuel:	17x17 RFA-2 w/IFM
Peaking Factor:	FQ=2.50, FdH=1.65
SG Tube Plugging:	10%
Power Level:	3565 MWth
Limiting transient:	4-inch Break

LICENSING BASIS

	Clad Temp (°F)	Ref.	Notes
Analysis of Record PCT	936 °F	1	

MARGIN ALLOCATIONS (ΔPCT)

A. PRIOR PERMANENT ECCS MODEL ASSESSMENTS			
1. None	0		
B. PLANNED PLANT CHANGE EVALUATIONS			
1. Loose Part Evaluation	45	2	(a)
C. 2015 PERMANENT ECCS MODEL ASSESSMENTS			
1. None	0		
D. TEMPORARY ECCS MODEL ISSUES			
1. None	0		
E. OTHER			
1. None	0		

LICENSING BASIS PCT + MARGIN ALLOCATIONS PCT = 981 °F

CUMULATIVE ABSOLUTE MAGNITUDE OF PCT CHANGES $\Sigma |\Delta PCT| = 0 \text{ °F}$

References:

1. WCAP-16717-P, Rev. 0, "Wolf Creek Generating Station (SAP), MSIV/MFIV Replacement Project, Small Break Loss of Coolant Accident Analysis Engineering Report," January 2007.
2. SAP-90-148/NS-OPLS-OPL-I-90-239, "Wolf Creek Nuclear Operating Corporation, RCS Loose Part Evaluation," April 1990.

Notes:

(a) This penalty will be carried to track the loose part which has not been recovered.