Wolf Creek
Nuclear Operating Corporation

Ron Benham
Manager Nuclear and Regulatory Affairs

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RA 20-0025

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

Reference: Westinghouse Letter LTR-LIS-20-34, dated February 11, 2020, "Wolf Creek 10 CFR 50.46 Annual Notification and Reporting for 2019"

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

To Whom It May Concern:

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).

WCNOC has reviewed the above Reference, which addresses 10 CFR 50.46 reporting information pertaining to the Emergency Core Cooling System (ECCS) Evaluation Model changes that were implemented by Westinghouse for 2019. The review concludes that the effect of changes to, or errors in, the Evaluation Models on the limiting transient peak cladding temperature (PCT) is not significant for 2019. Therefore, changes to the ECCS Evaluation Models are being reported as an annual report.

Attachment I provides an assessment of the specific changes and enhancements to the Westinghouse Evaluation Models for 2019. 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 Loss-of-Coolant Accident (LOCA) and Small Break LOCA PCT margin allocations in effect for the 2019 WCGS Evaluation Models. Since WCNOC has penalties that are carried on the tables, both Cycle 23 specific and Cycle 23 independent tables are included. The Cycle 23 independent tables are labeled as current. The PCT values determined in the Large Break and Small Break LOCA analyses 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.

Sincerely,

Ron Benham

Ron Benham

RDB/rlt

Attachments: 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: S. A. Morris (NRC), w/a
N. O'Keefe (NRC), w/a
B. K. Singal (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)

ERROR IN THE LOWER PLENUM HEAT TRANSFER AREA

Background

An error was found in the collection of the lower core plate flow hole details used to calculate the lower plenum heat transfer area for Wolf Creek. The correction of this error led to an approximate 1% increase in the lower plenum heat transfer area used in the Appendix K Small Break LOCA analysis. These changes represent a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The difference in the lower plenum heat transfer area is relatively minor and has been evaluated to have a negligible effect on the small break LOCA analysis results, leading to an estimated PCT impact of 0°F.

WOLF CREEK CYCLE 24 FUEL ROD RECONSTITUTION

Background

Within one fuel assembly, a single rod is being reconstituted for Wolf Creek Cycle 24. A cycle-specific evaluation is performed to support Cycle 24. The evaluation will be reviewed for continued applicability during the LOCA reload evaluation process for each subsequent cycle as the current limit is updated to account for the evaluation of one stainless steel filler rod. This item represents a change in plant configuration or associated set points, distinguished from an evaluation model change in Section 4 of WCAP-13451.

Affected Evaluation Model(s)

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM
1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The Large Break LOCA ASTRUM and Small Break LOCA NOTRUMP analyses of record were evaluated for one reconstituted rod. The estimated effect is 0°F for Small Break LOCA and 0°F for Large Break 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 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.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

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

REMOVAL OF THE VESSEL INTERFACIAL HEAT TRANSFER LIMIT

Background

The Westinghouse Code Qualification Document (CQD) Best-Estimate Large-Break LOCA (BELBLOCA) evaluation model (EM) is documented in WCAP-12945-P-A [1]. A limit on the vessel interfacial heat transfer was implemented into the WCOBRA/TRAC code as presented in Equation 5-12 therein. The implementation of the limit was intended to prevent any extreme conditions which are detrimental to the robustness of the numerical method. During the licensing of the method, the application of the limit was found to have a small impact on predicted results as discussed in the response to RAI1-116 of WCAP-12945-P-A [1].

An error was found in the implementation of the vessel interfacial heat transfer limit which effectively negates the application of the limit. The error was corrected by removing the vessel interfacial heat transfer limit from the WCOBRA/TRAC code (as opposed to a direct correction of the error). Since the WCOBRA/TRAC code validation and sensitivity studies associated with the model from WCAP-12945-PA [1] all contained the error, the removal of the limit preserves the existing validation basis and sensitivity study conclusions that were presented in the topical report. Based on the validation and RAI responses therein, it was concluded that the as-coded interfacial heat transfer models and condensation behavior was acceptable.

The removal of the vessel interfacial heat transfer limit 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
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The removal of the vessel interfacial heat transfer limit was found to have negligible impact on the WCOBRA/TRAC code validation results. The validation results in combination with pressurized water reactor large break LOCA transient calculations and engineering judgement support an estimated peak cladding temperature impact of 0°F.

Reference

1) WCAP-12945-P-A, Volume I, Revision 2, Volumes II through V, Revision 1, “Code Qualification Document for Best Estimate LOCA Analysis,” March 1998

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

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	WOLF CREEK
Utility Name:	Wolf Creek NOC
EM:	NOTRUMP
AOR Description:	Appendix K Small Break
Summary Sheet Status:	Current

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	936	1	

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Loose Part Evaluation	45	2	(a)	1990

AOR + ASSESSMENTS **PCT =** 981.0 °F

* The licensee should determine the reportability of these assessments pursuant to 10 CFR 50.46.

** The "Reporting Year" refers to the annual reporting year in which this assessment was included.

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.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	WOLF CREEK
Utility Name:	Wolf Creek NOC
EM:	ASTRUM (2004)
AOR Description:	Best Estimate Large Break
Summary Sheet Status:	Current

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1900	1	

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Containment Fan Cooler Capacity	0	2,4	(a)	2014
2. Decay Group Uncertainty Factors Errors	-10	3		2014

AOR + ASSESSMENTS	**PCT =** 1890.0 °F

* The licensee should determine the reportability of these assessments pursuant to 10 CFR 50.46.

** The "Reporting Year" refers to the annual reporting year in which this assessment was included.

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.

4 LTR-LIS-19-282, "Wolf Creek 10 CFR 50.46 PCT Summary Sheet Updates for Replacement Fan Cooler Tube Bundles Installation and Planned Retirement of Cycle 23 Sheets," August 2019.

NOTES:

(a) The estimated effect includes the corrected fan cooler heat removal rates and implementation of replacement tube bundles in the containment fan coolers, which were installed for Cycle 24.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	WOLF CREEK
Utility Name:	Wolf Creek NOC
EM:	NOTRUMP
AOR Description:	Appendix K Small Break
Summary Sheet Status:	Retired Cycle 23

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	936	1	

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Loose Part Evaluation	45	2	(a)	1990
2. Evaluation of One Reconstituted Fuel Rod (Limit 3.02 Violation)	0	3		2018

AOR + ASSESSMENTS **PCT =** 981.0 °F

* The licensee should determine the reportability of these assessments pursuant to 10 CFR 50.46.

** The "Reporting Year" refers to the annual reporting year in which this assessment was included.

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.

3 LTR-LIS-18-64 Revision 1, "Wolf Creek (SAP) Cycle 23 LOCA Safety Evaluation Input and Potential Issue (PI) Notification," April 2018.

NOTES:

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

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	WOLF CREEK
Utility Name:	Wolf Creek NOC
EM:	ASTRUM (2004)
AOR Description:	Best Estimate Large Break
Summary Sheet Status:	Retired Cycle 23

	PCT (°F)	**Reference #**	**Note #**
ANALYSIS-OF-RECORD	1900	1	

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Containment Fan Cooler Capacity	0	2		2014
2. Decay Group Uncertainty Factors Errors	-10	3		2014
3. Containment Fan Cooler Capacity	0	2	(a)	2014
4. Evaluation of One Reconstituted Fuel Rod (Limit 3.02 Violation)	0	4		2018

AOR + ASSESSMENTS **PCT =** 1890.0 °F

* The licensee should determine the reportability of these assessments pursuant to 10 CFR 50.46.

** The "Reporting Year" refers to the annual reporting year in which this assessment was included.

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.

4 LTR-LIS-18-64, Revision 1, "Wolf Creek (SAP) Cycle 23 LOCA Safety Evaluation Input and Potential Issue (PI) Notification," April 2018.

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.

10 CFR 50.46 Reporting SharePoint Site Check:

EMs applicable to Wolf Creek:
Realistic Large Break – ASTRUM (2004)
Appendix K Small Break – NOTRUMP

2019 Issues

Transmittal Letter	Issue Description
LTR-LIS-19-282	10 CFR 50.46 PCT Summary Sheet Updates for Replacement Fan Cooler Tube Bundles Installation and Planned Retirement of Cycle 23 Sheets