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Serial: RA-20-0060
April 14, 2020

10 CFR 52, Appendix D, X.B
10 CFR 50.59
10 CFR 52.97
10 CR 50.46

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

WILLIAM STATES LEE III NUCLEAR STATION, UNITS 1 AND 2
COMBINED LICENSE NOS NPF-101 AND NPF-102
DOCKET NOS. 52-018 AND 52-019

SUBJECT: Submission of Periodic Reports

REFERENCES:

1. Letter from Joseph W. Donahue (Duke Energy) to U.S. Nuclear Regulatory Commission (NRC), dated October 8, 2018, “Submission of Periodic Reports and Annual UFSAR Update,” Serial: NPD-NRC-2018-010.

The purpose of this letter is to submit periodic reports for William States Lee III Nuclear Station (WLS), Units 1 and 2 as required by NRC regulations and/or license conditions for a Part 52 combined license holder. These reports address various annual or semi-annual reporting requirements. The following reports are addressed by this letter:

- Semi-Annual Changes, Tests, and Experiments Report
- Semi-Annual Departures Report
- Semi-Annual Schedule for Implementation of Operational Programs
- Annual 10 CFR 50.46 Report

Semi-Annual Departures Report and Semi-Annual Changes, Tests, and Experiments Report. For the WLS Units 1 and 2, in accordance with the requirements of 10 CFR 50.59(d)(2) and 10 CFR 52, Appendix D, paragraphs X.B.1 and X.8.3.b, during the period of October 1 through March 31, 2020:

- no changes, tests or experiments were implemented pursuant to 10 CFR 50.59(c), and
- no plant-specific departures were implemented under 10 CFR 52, Appendix D, Section VIII.

Semi-Annual Schedule for Implementation of Operational Programs. Pursuant to the WLS COL Section 2.D.(11), a schedule for implementation of operational programs is required to be submitted within one year of the date of COL issuance, with subsequent reports submitted on a semi-annual basis until the 10 CFR 52.103(g) finding. There are no changes to the schedule previously sent in Reference 1. Therefore, the previously submitted schedule continues to be current.

Annual 10 CFR 50.46 Report. In accordance with 10 CFR 59.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," for the WLS Units 1 and 2. A design certification holder is required to report to the NRC in accordance with 10 CFR 50.46(a)(3). This same regulation requires a similar report from any combined license (COL) holder and COL applicant. The Duke Energy COL for WLS Units 1 and 2 incorporate by reference the AP1000 design certification document and thus, also utilize the peak cladding temperature calculations performed by Westinghouse Electric Company (WEC). As such, the WEC report, provided in the Enclosure, is applicable to the WLS Units 1 and 2.

This letter contains no new regulatory commitments.

Please address any comments or questions regarding this matter to Art Zaremba, Manager – Fleet Licensing at (980) 373-2062.

Sincerely,

M. Christopher Nolan
Vice President, Nuclear Regulatory Affairs, Policy & Emergency Preparedness

Enclosure:
Letter from Zachary S. Harper, Westinghouse Electric Company (WEC), to the U. S. Nuclear Regulatory Commission, 10 CFR 50.46 Annual Report for the AP1000 Plant Design, Letter No. DCP_NRC_003340, dated March 23, 2020.

cc: L. Dudes, U.S. NRC Region II Administrator
D. Murray, U.S. NRC Project Manager

Enclosure

Letter from Zachary S. Harper, Westinghouse Electric Company (WEC), to the U. S. Nuclear Regulatory Commission, 10 CFR 50.46 Annual Report for the AP1000 Plant Design, Letter No. DCP_NRC_003340, dated March 23, 2020.

Westinghouse

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Your Ref: Docket No. 52-006
Our Ref: DCP_NRC_003340

March 23, 2020

Subject: 10 CFR 50.46 Annual Report for the AP1000® Plant Design

Pursuant to 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors", Westinghouse Electric Company, LLC is submitting this report to document emergency core cooling system (ECCS) evaluation model changes or errors for the 2019 Model Year (i.e., 01/01/2019 – 12/31/2019) that affect the peak cladding temperature (PCT) calculations for the AP1000 plant design.

As described below, two AP1000 analyses of record (AORs) are reported:

AP1000 Design Certification AOR:
On December 30th, 2011, the U.S. Nuclear Regulatory Commission certified an amendment to the Design Certification Rule for the AP1000 plant. As such, AP1000 Design Control Document (DCD) Revision 19 documents the AOR for the AP1000 Design Certification. The limiting transient for the AP1000 Design Certification is the Best Estimate Large Break Loss-of-Coolant Accident (LBLOCA). Westinghouse last provided an annual reporting letter to the NRC in March 2019 (DCP_NRC_000146) which presented an estimated PCT of 2010°F for the LBLOCA evaluation. There are no new ECCS model changes that impact PCT for the 2019 model year. The estimated PCT for LBLOCA remains at 2010°F and does not exceed the 10 CFR 50.46 (b)(1) acceptance criterion of 2200°F.

The summary of the PCT margin allocations and their bases for the AP1000 Design Certification AOR are provided in the Attachment 1.

AP1000 Vogtle Units 3 & 4 AOR:
In addition to the AOR for the AP1000 Design Certification, the NRC has approved the AP1000 Core Reference Report (WCAP-17524-P-A), a generic topical which includes an ECCS "reanalysis" in the context of 10 CFR 50.46. The AOR contained in the Core Reference Report (CRR) has also been approved for incorporation into the Vogtle Units 3 & 4 licenses via NRC License Amendment 52. Additionally, changes as part of the reactor coolant system flow coastdown changes included in LAR-18-025 have been approved for incorporation into the Vogtle Units 3&4 licenses via NRC License Amendments 155 (Unit 3) and 154 (Unit 4). As

such, the estimated PCT for LBLOCA is 2046°F and does not exceed the 10 CFR 50.46 (b)(1) acceptance criterion of 2200°F.

The summary of the PCT margin allocations and their bases for the AP1000 Vogtle Units 3 & 4 AOR are provided in the Attachment 2.

By copy of this letter, COL Holders and COL Applicants are hereby notified of any changes or errors in the AP1000 standard plant design PCT calculations as required by 10 CFR 50.46(a)(3)(iii). This letter contains site-specific evaluations for Vogtle Units 3 & 4.

Questions or requests for additional information related to content and preparation of this information should be directed to Westinghouse. Please send copies of such questions or requests to the respective COL Holders and COL Applicants referencing the amended AP1000 Design Certification Rule for the AP1000 nuclear power plant. A representative for each COL Holder and COL Applicant is included on the cc: list of this letter.

Very truly yours,

Zachary S. Harper
Manager, AP1000 Licensing

/Attachments

1. 10 CFR 50.46 Annual Report for the AP1000 Design Certification AOR, 2019 Model Year
2. 10 CFR 50.46 Annual Report for the AP1000 Vogtle Units 3 & 4 AOR, 2019 Model Year

Cc:

J. Dixon-Herrity	- U.S. NRC	A. Zaremba	- Duke/Progress	M. Corletti	- Westinghouse
A. Bradford	- U.S. NRC	S. Franzone	- FP&L	A. Schoedel	- Westinghouse
D. Habib	- U.S. NRC	R. Orthen	- FP&L	M. Yuan	- Westinghouse
A. Chamberlain	- SNC	L. Oriani	- Westinghouse	D. McDevitt	- Westinghouse
M. Humphrey	- SNC	D. Weaver	- Westinghouse	M. Sheaffer	- Westinghouse
E. Grant	- SNC	A. Konig	- Westinghouse	M. Barca	- Westinghouse
Y. Arafeh	- SNC	K. Hosack	- Westinghouse	M. Patterson	- Westinghouse

Attachment 1

10 CFR 50.46 Annual Report for the AP1000 Design Certification AOR, 2019 Model Year

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 [1].

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.

References

1. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

REMOVAL OF THE VESSEL INTERFACIAL HEAT TRANSFER LIMIT

Background

The Westinghouse Code Qualification Document (CQD) Best-Estimate Large-Break LOCA (BE LBLOCA) 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-P-A [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 [2].

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.

References

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.
2. WCAP-13451, “Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting,” October 1992.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	AP1000
Utility Name:	Westinghouse Nuclear Power Plants
EM:	ASTRUM (2004)
AOR Description:	Best Estimate Large Break
Summary Sheet Status:	DCD

	PCT (°F)	**Reference #**	**Note #**
ANALYSIS-OF-RECORD	1837	1	

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Evaluation of Pellet Thermal Conductivity Degradation and Peaking Factor Burndown	139	2		2012
2. Revised Heat Transfer Multiplier Distributions	11	3		2013
3. Error in Burst Strain Application	23	4		2013

AOR + ASSESSMENTS **PCT =** 2010.0 °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 APP-GW-GL-700, Revision 19, "AP1000 Design Control Document," June 2011.

2 LTR-LIS-12-288, "Information Regarding the Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown Including Analysis Input Changes for AP1000 Large Break LOCA Analysis," June 2012.

3 LTR-LIS-13-357, "AP1000 Plant 10 CFR 50.46 Report for Revised Heat Transfer Multiplier Distributions," July 2013.

4 LTR-LIS-14-41, "AP1000 Plant 10 CFR 50.46 Report for the HOTSPOT Burst Strain Error Correction," January 2014.

NOTES:

(a) None

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name: AP1000
Utility Name: Westinghouse Nuclear Power Plants
EM: NOTRUMP-AP
AOR Description: Appendix K Small Break
Summary Sheet Status: DCD

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1370	1	(a)

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Adiabatic Heat-up Calculation	264	2	(a)	2010

AOR + ASSESSMENTS **PCT =** 1634.0 °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 APP-GW-GL-700, Revision 19, "AP1000 Design Control Document," June 2011.

2 LTR-LIS-10-373, "10 CFR 50.46 Report for the Evaluation of AP1000 SBLOCA 10-inch Transient Adiabatic Heat-up Calculation," June 2010.

NOTES:

(a) This is an adiabatic heat-up calculated PCT.

Attachment 2

10 CFR 50.46 Annual Report for the AP1000 Vogtle Units 3 & 4 AOR, 2019 Model Year

AP1000 PLANT EVALUATION OF RCP DESIGN CHANGES

Background

The design changes associated with the reactor coolant pump (RCP) have been evaluated against the **AP1000**® Core Reference Report large break loss-of-coolant accident (LBLOCA) analysis performed in [1]. The evaluated design changes included updated homologous curves and small changes to the pump rated conditions and rotor inertia.

These items represent changes in plant configuration, distinguished from an evaluation model change in Section 4 of WCAP-13451 [2].

Affected Evaluation Model(s)

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A quantitative LBLOCA evaluation was performed to assess the impacts of the RCP design changes on the LBLOCA analysis. Based on the plant-specific simulations, it was concluded that the design changes result in an estimated impact of 22°F on the AP1000 plant Core Reference Report analysis [1]. Therefore, the estimated Peak Cladding Temperature (PCT) penalty is 22°F for the AP1000 plant.

References

1. WCAP-17524-P-A, Revision 1, "AP1000 Core Reference Report," May 2015.
2. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

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 [1].

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.

References

1. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

REMOVAL OF THE VESSEL INTERFACIAL HEAT TRANSFER LIMIT

Background

The Westinghouse Code Qualification Document (CQD) Best-Estimate Large-Break LOCA (BE LBLOCA) 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-P-A [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 [2].

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.

References

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.
2. WCAP-13451, “Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting,” October 1992.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	Vogtle Unit 3 and Unit 4
Utility Name:	Southern Nuclear Operating Company
EM:	ASTRUM (2004)
AOR Description:	Best Estimate Large Break
Summary Sheet Status:	Current

	PCT (°F)	Reference #	Note #
ANALYSIS-OF-RECORD	1936	1	(a)

ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. Revised Heat Transfer Multiplier Distributions	11	2		2013
2. Error in Burst Strain Application	23	3		2013
3. Design Change Rebaseline Evaluation	54	4,5	(b)	2018
4. Evaluation of RCP Design Changes – LAR 189	22	6,7	(c)	2019

AOR + ASSESSMENTS **PCT =** 2046.0 °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-17524-P-A, Revision 1, “AP1000 Core Reference Report,” May 2015.

2 LTR-LIS-13-357, “AP1000 Plant 10 CFR 50.46 Report for Revised Heat Transfer Multiplier Distributions,” July 2013.

3 LTR-LIS-14-41, “AP1000 Plant 10 CFR 50.46 Report for the HOTSPOT Burst Strain Error Correction,” January 2014.

4 LTR-LIS-18-393, “Update to the Vogtle Units 3 & 4 LBLOCA and SBLOCA 10 CFR 50.46 PCT Summary Sheets for LAR-79,” November 2018.

5 ND-17-2074 (ML18029A243), “Containment Pressure Analysis (LAR-17-043),” December 2017. Approved by NRC November 7, 2018 as Amendments 147 (VEGP Unit 3) and 146 (VEGP Unit 4) (ML18289A742).

6 LTR-LIS-17-39, “AP1000 Plant Suggested 10 CFR 50.46 Reporting Text and Updated LBLOCA PCT Summary Sheet for Evaluation of Reactor Coolant Pump (RCP) Design Changes,” January 2017.

7 ND-18-1147 (ML18243A459), “Reactor Coolant System (RCS) Flow Coastdown (LAR-18-025),” August

2018. Approved by NRC February 25, 2019 as Amendments 155 (VEGP Unit 3) and 154 (VEGP Unit 4) (ML19038A450).

NOTES:

(a) Value contains 2°F bias for PCT sensitivity to PRHR isolation, per Reference 1 response to CRR-008, Table 2 and Table 15.6.5-8.

(b) The design change rebaseline evaluation used current code versions and accounts for design changes up to May 5, 2014 and plant model error corrections.

(c) The RCP design changes evaluation assesses the impact of DCP 5338 (APP-GW-GEE-5338), which is tied to DCP 4880 (APP-GW-GEE-4880). The evaluated changes include updated homologous curves and small changes to the pump rated conditions and rotor inertia.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	Vogtle Unit 3 and Unit 4
Utility Name:	Southern Nuclear Operating Company
EM:	NOTRUMP-AP
AOR Description:	Appendix K Small Break
Summary Sheet Status:	Current

	PCT (°F)	Reference #	Note #	
ANALYSIS-OF-RECORD	663.5	1		
ASSESSMENTS*	**Delta PCT (°ΔF)**	**Reference #**	**Note #**	**Reporting Year****
1. NOTRUMP Bubble Rise/Drift Flux Model Inconsistencies	32	2		2014
2. LAR-114 Evaluation	13	3,4,5	(a)	2016
3. LAR-133 Evaluation	144	6,7	(b)	2018
4. Design Change Rebaseline Analysis	243.5	8,9	(c)	2018
AOR + ASSESSMENTS	**PCT =** 1096.0 °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-17524-P-A, Revision 1, "AP1000 Core Reference Report," May 2015.

2 LTR-LIS-15-5, "Updates to the AP1000 Plant SBLOCA 10 CFR 50.46 PCT Rackups," January 2015.

3 LTR-LIS-16-144, "Update to the AP1000 Plant SBLOCA 10 CFR 50.46 PCT Rackups for LAR-114," January 2017.

4 LTR-LIS-17-59, "AP1000 Plant 10 CFR 50.46 Annual Notification and Reporting for 2016," March 2017.

5 ND-16-0984 (ML16207A340), "Automatic Depressurization System (ADS) Stage 2, 3 & 4 Valve Flow Area Changes and Clarifications (LAR-16-012)," July 2016. Approved by NRC December 29, 2016 as Amendment 62 (ML16357A640).

6 LTR-LIS-16-429, "Update of the AP1000 Plant SBLOCA 10 CFR 50.46 PCT Rackups for LAR-133," January 2017.

7 ND-17-0443 (ML17090A209), "PXS/ADS Line Resistance Changes (LAR-17-009)," March 2017. Approved by NRC February 28, 2018 as Amendments 111 (VEGP Unit 3) and 110 (VEGP Unit 4) (ML18026A566/ML18026A571).

8 LTR-LIS-18-393, "Update to the Vogtle Units 3 & 4 LBLOCA and SBLOCA 10 CFR 50.46 PCT

Summary Sheets for LAR-79," November 2018.

9 ND-17-2074 (ML18029A243), "Containment Pressure Analysis (LAR-17-043)," December 2017. Approved by NRC November 7, 2018 as Amendments 147 (VEGP Unit 3) and 146 (VEGP Unit 4) (ML18289A742).

NOTES:

(a) The LAR-114 evaluation assesses the impact of reduced automatic depressurization system (ADS) Stage 2, 3, and 4 flow areas described in design change proposals (DCPs) 5051 and 5054.

(b) The LAR-133 evaluation assesses the impact of updated ADS Stages 1 – 4 and In-Containment Refueling Water Storage Tank (IRWST) line resistances described in DCPs 4903 and 5138.

(c) The design change rebaseline analysis used current code versions and accounts for design changes up to August 4, 2014 and plant model error corrections.