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Serial: RA-19-0173
April 10, 2019

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, 2019:

- no changes, tests or experiments were implemented pursuant to 10 CFR 50.59(c), and
- no plant-specific departures were implemented under 1 O 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

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_000146, dated March 26, 2019.

cc: C. Haney, U.S. NRC Region II Administrator
B. Hughes, 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_000146, dated March 26, 2019.

Westinghouse

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USA

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Your Ref: Docket No. 52-006
Our Ref: DCP_NRC_000146

March 26, 2019

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 2018 Model Year (i.e., 01/01/2018 – 12/31/2018) that affect the peak cladding temperature (PCT) calculations for the AP1000 plant design.

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

<u>AP1000 Design Certification AOR:</u>
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, 2018 (DCP_NRC_003328) which presented an estimated PCT of 2010°F for the LBLOCA evaluation. There are no new ECCS model changes that impact PCT for the 2018 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.

<u>AP1000 V.C. Summer Units 2 & 3 AOR:</u>
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 V.C. Summer Units 2 & 3 licenses via license amendment request (LAR). There are no new ECCS model changes that impact PCT for the 2018 model year. The estimated PCT for LBLOCA remains at 1970°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 V.C. Summer Units 2 & 3 AOR are provided in the Attachment 2.

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 containment pressure analysis included in LAR-17-043 have been approved for incorporation into the Vogtle Units 3&4 licenses via NRC License Amendments 147 (Unit 3) and 146 (Unit 4). As such, the estimated PCT for LBLOCA is 2024°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 3.

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 V.C. Summer Units 2 & 3 and 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, 2018 Model Year
2. 10 CFR 50.46 Annual Report for the AP1000 V.C. Summer Units 2 & 3 AOR, 2018 Model Year
3. 10 CFR 50.46 Annual Report for the AP1000 Vogtle Units 3 & 4 AOR, 2018 Model Year

Cc:

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

Attachment 1

10 CFR 50.46 Annual Report for the AP1000 Design Certification AOR, 2018 Model Year

UO_2 FUEL PELLET HEAT CAPACITY

Background

A typographical error was discovered in the implementation of the UO_2 fuel pellet heat capacity as described by Equation C-4 of WCAP-8301 [1] for fuel rod heat-up calculations within the Appendix K Large Break and Small Break LOCA evaluation models. The erroneous formulation results in an over-prediction of heat capacity that increases with fuel temperature. The corrected formulation results in a maximum decrease in heat capacity on the order of approximately 1.2% for existing analyses of record. This represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 [2].

Affected Evaluation Model(s)

1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP.

Estimated Effect

The small over-prediction in UO_2 fuel pellet heat capacity has been evaluated to have a negligible effect on existing large and small break LOCA analysis results due to the small magnitude of the change, leading to an estimated PCT impact of 0°F.

References

1. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," June 1974.
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 V.C. Summer Units 2 & 3 AOR, 2018 Model Year

UO_2 FUEL PELLET HEAT CAPACITY

Background

A typographical error was discovered in the implementation of the UO_2 fuel pellet heat capacity as described by Equation C-4 of WCAP-8301 [1] for fuel rod heat-up calculations within the Appendix K Large Break and Small Break LOCA evaluation models. The erroneous formulation results in an over-prediction of heat capacity that increases with fuel temperature. The corrected formulation results in a maximum decrease in heat capacity on the order of approximately 1.2% for existing analyses of record. This represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 [2].

Affected Evaluation Model(s)

1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP.

Estimated Effect

The small over-prediction in UO_2 fuel pellet heat capacity has been evaluated to have a negligible effect on existing large and small break LOCA analysis results due to the small magnitude of the change, leading to an estimated PCT impact of 0°F.

References

1. WCAP-8301, “LOCTA-IV Program: Loss-of-Coolant Transient Analysis,” June 1974.
2. WCAP-13451, “Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting,” October 1992.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	V. C. Summer Unit 2 and Unit 3
Utility Name:	South Carolina Electric & Gas
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

AOR + ASSESSMENTS **PCT =** 1970.0 °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.

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.

LOCA Peak Cladding Temperature (PCT) Summary

Plant Name:	V. C. Summer Unit 2 and Unit 3
Utility Name:	South Carolina Electric & Gas
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)	2017
AOR + ASSESSMENTS	**PCT =** 708.5 °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-18-53, "AP1000 Plant 10 CFR 50.46 Annual Notification and Reporting for 2017," March 2018.

5 NND-16-0336 (ML16246A214), "Automatic Depressurization System (ADS) Stage 2, 3 & 4 Valve Flow Area Changes and Clarifications," September 2016. Approved by NRC March 17, 2017 as Amendment 64 (ML17039B008/ML17039B058).

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.

Attachment 3

10 CFR 50.46 Annual Report for the AP1000 Vogtle Units 3 & 4 AOR, 2018 Model Year

LARGE BREAK LOCA DESIGN CHANGE REBASELINE EVALUATION

Background
Numerous **AP1000®** plant design changes have occurred since the AP1000 Core Reference Report large break loss-of-coolant accident (LBLOCA) analysis was performed [1]. The design changes impacting the LBLOCA analysis included numerous passive core cooling system (PXS), vessel model, containment pressure, reactor coolant system (RCS), and secondary system changes.

These items represent changes in plant configuration or associated set points, 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
An LBLOCA rebaseline evaluation was performed using the latest code versions with model corrections to determine an updated limiting peak cladding temperature (PCT) considering the AP1000 plant design changes up to May 5, 2014. The updated evaluation resulted in a PCT penalty of 54°F against the AP1000 plant Core Reference Report analysis [1].

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.

SMALL BREAK LOCA DESIGN CHANGE REBASELINE ANALYSIS

Background
Numerous **AP1000®** plant design changes have occurred since the AP1000 Core Reference Report small break loss-of-coolant accident (SBLOCA) analysis was performed [1]. The design changes impacting the SBLOCA analysis included numerous passive core cooling system (PXS), automatic depressurization system (ADS), reactor coolant system (RCS), containment pressure, and secondary system changes.

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

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP.

Estimated Effect
An SBLOCA rebaseline analysis was performed using the latest code versions with model corrections to determine an updated limiting peak cladding temperature (PCT) considering the AP1000 plant design changes up to August 4, 2014. The updated analysis resulted in a PCT penalty of 243.5°F against the AP1000 plant Core Reference Report analysis [1].

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.

AP1000 PLANT LAR-133 EVALUATION

Background
The design changes associated with License Amendment Request (LAR) 133 are changes to automatic depressurization system (ADS) Stages 1, 2, 3, and 4 line resistances and in-containment refueling water storage tank (IRWST) line resistances. These changes impact the **AP1000®** plant small break loss- of-coolant accident (SBLOCA) analysis. These items represent changes in plant configuration or associated set points, distinguished from an evaluation model change in Section 4 of WCAP-13451 [1].

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP.

Estimated Effect
Updated SBLOCA calculations using the latest limiting SBLOCA transient have been performed to assess the impact of the changes to the ADS Stages 1 – 4 and IRWST injection line resistances associated with LAR-133. The updated calculations resulted in a 144°F PCT penalty, which is assessed against the AP1000 Core Reference Report analysis [2].

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

UO_2 FUEL PELLET HEAT CAPACITY

Background

A typographical error was discovered in the implementation of the UO_2 fuel pellet heat capacity as described by Equation C-4 of WCAP-8301 [1] for fuel rod heat-up calculations within the Appendix K Large Break and Small Break LOCA evaluation models. The erroneous formulation results in an over-prediction of heat capacity that increases with fuel temperature. The corrected formulation results in a maximum decrease in heat capacity on the order of approximately 1.2% for existing analyses of record. This represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 [2].

Affected Evaluation Model(s)

1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP.

Estimated Effect

The small over-prediction in UO_2 fuel pellet heat capacity has been evaluated to have a negligible effect on existing large and small break LOCA analysis results due to the small magnitude of the change, leading to an estimated PCT impact of 0°F.

References

1. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," June 1974.
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

AOR + ASSESSMENTS **PCT =** 2024.0 °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).

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.

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.