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April 16, 2015

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

10 CFR 50.46

SUBJECT: William States Lee III Nuclear Station (WLS), Units 1 and 2
Docket Nos. 52-018 and 52-019
Ltr#: WLG2015.03-02

Levy Nuclear Plant (LNP), Units 1 and 2
Docket Nos. 52-029 and 52-030
Shearon Harris Nuclear Power Plant (HAR), Units 2 and 3
Docket Nos. 52-022 and 52-023
Serial: NPD-NRC-2015-013

10 CFR 50.46 ANNUAL REPORT

Ladies and Gentlemen:

The purpose of this letter is to provide a required report in accordance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," for William States Lee III Nuclear Station Units 1 and 2, Levy Nuclear Plant, Units 1 and 2, and Shearon Harris Nuclear Power Plant, Units 2 and 3.

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) applicant if the applicant is also affected by the change. The Duke Energy applications for the William States Lee III Nuclear Station Units 1 and 2, Levy Nuclear Plant, Units 1 and 2, and Shearon Harris Nuclear Power Plant, Units 2 and 3, incorporate by reference the AP1000 design certification document (DCD) and thus, also utilize the peak cladding temperature calculations performed by WEC, as such the WEC report is also applicable to the William States Lee Nuclear Station (WLS) Units 1 and 2, Levy Nuclear Plant (LNP) Units 1 and 2, and Shearon Harris Nuclear Power Plant (HAR) Units 2 and 3 AP1000 COL applications as provided in Attachment 1.

If you have any questions or need any additional information, please contact Robert H. Kitchen, Nuclear Development Licensing Director, at (704) 382-4046.

Sincerely,

Robert H. Kitchen
Director – Nuclear Licensing
Nuclear Development

Attachment: Letter from Paul A. Russ, Westinghouse Electric Company (WEC), to the Nuclear Regulatory Commission (NRC), 10 CFR 50.46 Annual Report for the AP1000 Standard Plant Design, Letter No. DCP_NRC_003285, dated March 25, 2015

D084
D093
D094
A002
NRO

cc: (without attachment)
U.S. NRC Region II, Regional Administrator
U.S. NRC Resident Inspector SHNPP Unit 1

cc: (with attachment)
Mr. Donald Habib, U.S. NRC Project Manager
Mr. Brian Hughes, U.S. NRC Project Manager

Attachment 1

TO

NPD-NRC-2015-013

WLG2015.03-02

Letter from Paul A. Russ, Westinghouse Electric Company (WEC), to the Nuclear Regulatory Commission (NRC), 10 CFR 50.46 Annual Report for the AP1000 Standard Plant Design, Letter No. DCP_NRC_003285, dated March 25, 2015

(24 pages attached including this cover page)

Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
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USA

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Direct fax: 724-940-8505
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Your Ref: Docket No. 52-006
Our Ref: DCP_NRC_003285

March 25, 2015

Subject: 10 CFR 50.46 Annual Report for the AP1000®[1] Standard 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 2014 – 2015 model year that affect the peak cladding temperature (PCT) calculations for the ***AP1000*** standard plant design. On December 30th, 2011, the U.S. Nuclear Regulatory Commission amended its regulations to certify an amendment to the Design Certification rule for the ***AP1000*** standard plant. As such, ***AP1000*** Design Control Document (DCD) Revision 19 now documents the analyses of record (AOR).

The limiting Transient for the ***AP1000*** Certified Design as documented in the ***AP1000*** DCD Revision 19 is the Best Estimate Large Break Loss-of-Coolant Accident (BELOCA). Westinghouse last provided an annual reporting letter to the NRC in March, 2014 (DCP_NRC_003262) which presented an estimated PCT of 2010°F for the BELOCA evaluation. There are no ECCS model changes that impact PCT for the 2014 – 2015 model year. The estimated PCT for BELOCA 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 are provided in the Attachment.

The information included in this letter is generic and is expected to apply to all COL Holders and COL Applicants referencing the amended Design Certification Rule for the ***AP1000*** nuclear power plant. 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).

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 Design Certification Rule for the ***AP1000*** standard plant. A representative for each COL Holder and COL Applicant is included on the cc: list of this letter.

Very truly yours,

Paul A. Russ
Director, U.S. Licensing and Regulatory Support
Nuclear Power Plants

/Attachment

1. 10 CFR 50.46 Annual Report for the ***AP1000*** Standard Plant Design, 2014 – 2015 Model Year

Cc:	L. Burkhart	- U.S. NRC
	B. Bavol	- U.S. NRC
	D. McGovern	- U.S. NRC
	D. Jaffe	- U.S. NRC
	A. Rice	- SCANA
	B. Whitley	- Southern Company
	R. Kitchen	- Duke/Progress Energy
	D. Stout	- TVA
	S. Franzone	- Florida Power & Light
	T. Geer	- Westinghouse
	D. Weaver	- Westinghouse
	G. Couture	- Westinghouse
	B. McIntyre	- Westinghouse
	R. DeLong	- Westinghouse
	Z. Harper	- Westinghouse
	K. Hosack	- Westinghouse
	L. Guo	- Westinghouse
	J. Meneely	- Westinghouse
	J. Eisenhauer	- Westinghouse
	A. Colussy	- Westinghouse
	M. Cerrone	- Westinghouse

Internal Reference: LTR-LIS-15-78 Revision 1

ATTACHMENT

10 CFR 50.46 Annual Report for the ***AP1000*** Standard Plant Design

2014 – 2015 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 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 [1].

Affected Evaluation Models

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 (PCT) impact of 0°F.

References

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

ERRORS IN DECAY GROUP UNCERTAINTY FACTORS

Background

Errors in the calculation of decay heat were discovered in the WCOBRA/TRAC code. The decay group uncertainty factors for each fissile isotope are provided in Table 8-14 of WCAP-16009-P-A [1]. The uncertainty factors for ^{239}Pu were applied to ^{238}U, and those for ^{238}U were applied to ^{239}Pu. This error causes an over-prediction of the uncertainty in decay power from ^{239}Pu and an under-prediction of the uncertainty in decay power from ^{238}U. Further, the decay group uncertainty factor for Decay Group 6 of ^{235}U was erroneously coded as 2.5% instead of 2.25%. Correction of these errors impacts the application of the sampled decay heat uncertainty, which may result in small changes to the decay heat power. These issues have been evaluated to estimate the impact on Automated Statistical Treatment of Uncertainty Method (ASTRUM) Best-Estimate (BE) Large-Break Loss-of-Coolant Accident (LBLOCA) analysis results. The resolution of these issues represents a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451 [2].

Affected Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Based on inspection of the limiting cases, it was concluded that the decay heat power was conservatively modeled for the limiting case, resulting in an estimated Peak Cladding Temperature (PCT) impact of 0°F for the **AP1000®** plant.

References

1. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005.
2. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

TREATMENT OF BURNUP EFFECTS

Background

A detailed review of the method used to address the impact of pellet thermal conductivity degradation (TCD) indicated that some approximations made in the method to address TCD could introduce non-conservatisms. Fuel pellet TCD and peaking factor burndown were not explicitly considered in the **AP1000**® Large Break Loss-of-Coolant Accident (LBLOCA) Analysis of Record (AOR) presented in **AP1000** Design Control Document Revision 19 [1]. A quantitative evaluation was used to assess the PCT effect of TCD and peaking factor burndown on the **AP1000** LBLOCA analysis presented in DCD Revision 19.

The method to address TCD translated the statistically sampled time-in-cycle into rod average burnups consistent with the NRC-approved ASTRUM methodology [2]. For second-cycle fuel, this approach tends to over-estimate the actual rod burnups in the core. Because the peaking factors assumed in the analysis reduce as rod burnup increases, the analyzed rod power and local peaking could under-estimate the peaking factors of the rods at the sampled time-in-cycle.

Also, the treatment of gamma energy deposition assumes that the modeled hot assembly is surrounded by assemblies with power consistent with the core-average. However, second-cycle assemblies are often face-adjacent with feed assemblies, which can have power higher than the core-average. This can result in an over-estimation of the energy deposited away from the hot rod and hot assembly.

Note that pursuant to 10 CFR 50.46, reporting of this issue is not required.

Affected Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A quantitative evaluation was performed to assess the PCT effect with other considerations of burnup and concluded that the estimated PCT impact on the **AP1000** LBLOCA analysis presented in DCD Revision 19 is 0°F for 10 CFR 50.46 reporting purposes.

References

1. APP-GW-GL-700, Tier 2, Chapter 15, Rev. 19, "AP1000 Design Control Document: Accident Analyses," June 2011.
2. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005.

PRESSURIZER ERROR CORRECTION

Background

An error in the upper head height was discovered in the original model input calculation which impacted the total pressurizer height. When this change was accounted for, the as-modeled pressurizer total height and volume were less than 1% different than the design values. The modeled pressurizer fluid volume was not affected by the changes.

The change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 [1].

Affected Evaluation Models

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Considering that the magnitude of the changes in pressurizer height and volume are less than 1%, this change to the pressurizer was evaluated to have a negligible impact on peak clad temperature (PCT), leading to an estimated impact of 0°F.

References

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

PRESSURIZER SURGE LINE RESISTANCE ERROR CORRECTION

Background

An error was discovered in the original model input calculation of the pressurizer surge line resistance because the resistance of the surge line inlet basket was not accounted for; accounting for the basket resistance increases the pressurizer surge line resistance by 23% from the as-modeled value.

The change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 [1].

Affected Evaluation Models

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The increased pressurizer surge line resistance would result in the pressurizer draining more slowly, which impacts the flow rate out of the vessel during blowdown similar to the broken loop resistance. However, this change does not impact the safety system actuation because in the large break LOCA analysis, the safety injection signal is actuated by the high-2 containment pressure, before the pressurizer low pressure for safety injection is reached. Considering these factors, the change to the pressurizer surge line resistance was evaluated to have a negligible impact on peak clad temperature (PCT), leading to an estimated impact of 0°F.

References

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

NOTRUMP BUBBLE RISE/DRIFT FLUX MODEL INCONSISTENCIES

Background
The NOTRUMP-AP600 computer code was updated to resolve inconsistencies in several drift flux models as well as the nodal bubble rise/droplet fall models. These issues were previously reported in LTR-NRC-14-22 (Reference 1) and include the following:

- Bubble rise and droplet fall model calculations were made more consistent with flow link calculations.
- Corrections were made to limits employed in the vertical counter-current flooding models.
- Checking logic was added to correct situations where drift flux model inconsistencies could result (i.e. prevent liquid flow from an all-vapor node and vapor flow from all-liquid node).

In addition, the following changes were made:
- Selection of bubble rise model drift flux related input parameters (donoring scheme).
- Utilization of mass-based solution formulation for bubble rise/droplet fall.

The revised bubble rise model was implemented to obtain more consistent behavior between the nodal bubble rise model and the flow links connected to them thereby resulting in improved code behavior/stability. During application to the advanced plant model's solution formulation (volumetric based), validation efforts indicated the necessity to utilize the standard plant solution formulation (mass based) for bubble rise/droplet fall to prevent non-physical behavior under nodal boundary crossings. In addition, the bubble rise/droplet fall for the SG tube region of the model was revised to utilize the standard plant model donoring scheme. The validation work performed to support this implementation indicates the model is behaving as expected.

These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451 (Reference 3).

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

Estimated Effect
Due to the nature of the simulations performed in the **AP1000®** plant Design Control Document (Reference 2), the relatively minor changes in results associated with Revision 1 of the **AP1000** plant Core Reference Report and engineering judgment, the estimated effect previously transmitted in Reference 1 of 0°F continues to be assessed.

References
1. LTR-NRC-14-22, "U. S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2013," June 2014.
2. APP-GW-GL-700, Tier 2, Chapter 15, Rev. 19, "AP1000 Design Control Document: Accident Analyses," June 2011.
3. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

FUEL ROD GAP CONDUCTANCE ERROR CORRECTION

Background

An error was identified in the fuel rod gap conductance model in the NOTRUMP-AP600 computer code. The error was associated with the use of an incorrect temperature in the calculation of the cladding emissivity term. This error corresponds to a Non-Discretionary Change as described in Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

An estimated peak cladding temperature effect of 0°F is assessed for the existing **AP1000®** plant small break LOCA analysis results based upon NOTRUMP-AP600 sensitivity calculations.

References

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

RADIATION HEAT TRANSFER MODEL ERROR CORRECTION

Background

Two errors were discovered in the calculation of the radiation heat transfer coefficient within the fuel rod model of the NOTRUMP-AP600 computer code. First, existing logic did not preclude non-physical negative or large (negative or positive) radiation heat transfer coefficients from being calculated. These calculations occurred when the vapor temperature exceeded the cladding surface temperature or when the predicted temperature difference was less than 1°F. Second, a temperature term incorrectly used degrees Fahrenheit instead of Rankine. These errors have been corrected in the NOTRUMP-AP600 code and represent a closely related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

An estimated peak cladding temperature effect of 0°F is assessed for the existing **AP1000®** plant small break LOCA analysis results based upon NOTRUMP-AP600 sensitivity calculations.

References

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

DOWNCOMER CONDENSATION MODEL ERROR CORRECTION

Background
An error in the direct vessel injection (DVI) condensation model was discovered in the advanced plant version of the NOTRUMP code (NOTRUMP-AP600). The error causes the model to become deactivated as a result of improperly tracking the active mixture level location in the vessel downcomer region, thereby resulting in the under-prediction of condensation effects. This error has been corrected in the NOTRUMP-AP600 code and represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 (Reference 1).

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

Estimated Effect
An estimated peak cladding temperature effect of 0°F is assessed for the existing **AP1000®** plant small break LOCA analysis results based upon NOTRUMP-AP600 sensitivity calculations.

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

GATE VALVE LENGTH INPUT ERROR IN THE DETAILED MOMENTUM FLUX MODEL (FLOAD4)

Background

As part of the small-break loss-of-coolant accident (SBLOCA) advanced plant evaluation model using NOTRUMP (NOTRUMP-AP600 EM) used for the **AP1000®** plant, the automatic depressurization system stage 4 (ADS-4) flow path resistances are increased for non-critical flow conditions to accommodate the lack of a detailed momentum flux model in the ADS-4 discharge paths in the NOTRUMP code. The FLOAD4 code is used to perform the detailed momentum flux calculations, from which the ADS-4 flow path resistance increase is determined. An error in a gate valve length input was identified in the FLOAD4 calculations used to support both Revision 19 of the Design Control Document (DCD) and Revision 1 of the Core Reference Report (CRR) analyses. The resolution of this issue represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Models

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

Estimated Effect

Based on a sensitivity run, the correction of this error is found to result in a decrease in the ADS-4 flow path resistance increase factor; therefore, the value used in the SBLOCA analyses is judged to remain bounding leading to an estimated PCT impact of 0°F.

References

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

ERROR IN THE ELEVATION TERM CALCULATION IN THE DETAILED MOMENTUM FLUX CODE (FLOAD4)

Background

As part of the small-break loss-of-coolant accident (SBLOCA) advanced plant evaluation model using NOTRUMP (NOTRUMP-AP600 EM) used for the **AP1000®** plant, the automatic depressurization system stage 4 (ADS-4) flow path resistances are increased for non-critical flow conditions to accommodate the lack of a detailed momentum flux model in the ADS-4 discharge paths in the NOTRUMP code. The FLOAD4 code is used to perform the detailed momentum flux calculations, from which the ADS-4 flow path resistance increase is determined. An error was discovered in an elevation calculation performed within the FLOAD4 code resulting in the total elevation not being preserved. This issue exists in the FLOAD4 calculations used to support both Revision 19 of the Design Control Document (DCD) and Revision 1 of the Core Reference Report (CRR) analyses. The resolution of this issue represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Models

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

Estimated Effect

Based on a sensitivity run, the correction of this error is found to result in a decrease in the ADS-4 flow path resistance increase factor; therefore, the value used in the SBLOCA analyses is judged to remain bounding, leading to an estimated PCT impact of 0°F.

References

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

SBLOCTA PRE-DNB CLADDING SURFACE HEAT TRANSFER COEFFICIENT CALCULATION

Background

Two errors were discovered in the pre-departure from nucleate boiling (pre-DNB) cladding surface heat transfer coefficient calculation in the SBLOCTA code (cladding heat-up calculations). The first error is a result of inconsistent time units (hours vs. seconds) in the parameters used for the calculation of the Reynolds and Prandtl numbers, and the second error relates to an incorrect diameter used to develop the area term in the cladding surface heat flux calculation. Both of these issues impact the calculation of the pre-DNB convective heat transfer coefficient, representing a closely related group of Non-Discretionary Changes to the Evaluation Model as described in Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

These errors have been corrected in the SBLOCTA code. Because this condition occurred prior to DNB, it was judged that these errors had no direct impact on the cladding heat-up related to the core uncovery period. A series of validation tests were performed, and confirmed that these errors have a negligible effect on SBLOCA analysis results, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

References

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

AP1000 PLANT ACCUMULATOR TANK VOLUME INPUT

Background

The minimum accumulator tank volume was input into the **AP1000**® plant small-break loss-of-coolant accident (SBLOCA) NOTRUMP analyses (both the **AP1000** Plant Design Control Document (DCD), Revision 19, and the Core Reference Report (CRR), Revision 1, analyses) instead of the nominal accumulator tank volume, which is a 1% increase in volume. This error has been evaluated for impact on existing analyses and represents a Non-Discretionary Change to the Evaluation Model as described in Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

The impact of using a minimum instead of a nominal accumulator tank volume on the **AP1000** plant SBLOCA analyses was qualitatively evaluated. Based on the magnitude of the change, it is concluded that this correction has a negligible impact on the SBLOCA transient timing and results, leading to an estimated peak cladding temperature impact of 0°F for the SBLOCA DCD and CRR analyses.

References

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

AP1000 PLANT PRHR HEAT EXCHANGER SBLOCA INPUT ERRORS

Background

Several errors were discovered in the inputs associated with the **AP1000**® plant small break loss-of-coolant accident (SBLOCA) analysis passive residual heat removal (PRHR) heat exchanger model. The inlet and outlet tubesheets were neglected from input calculations, a portion of tube length was neglected for inertial length calculations, and the loss coefficient distribution was incorrect. The corrected input values have been evaluated for impact on current licensing-basis analysis results described in Revision 19 of the **AP1000** plant Design Control Document (DCD) and Revision 1 of the **AP1000** plant Core Reference Report (CRR). These changes represent a closely-related group of Non-Discretionary Changes to the Evaluation Model as described in Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

Qualitative and quantitative evaluations have been used to conclude that the differences would have a negligible effect on the **AP1000** plant SBLOCA analysis results, leading 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.

AP1000 PLANT CORE MAKEUP TANK DISCHARGE CHECK VALVE INPUT ERROR

Background

The **AP1000**® plant small break loss-of-coolant accident (SBLOCA) analysis core makeup tank (CMT) discharge piping check valve model has historically captured a simplified, normally closed, check valve. It was recently determined that the simplified check valve representation is not reflective of the **AP1000** plant design. The as designed, normally open, check valve may allow reverse flow conditions that are not captured in the current SBLOCA analyses in Revision 19 of the **AP1000** plant Design Control Document (DCD) and Revision 1 of the **AP1000** plant Core Reference Report (CRR). This change represents a Non-Discretionary Change to the Evaluation Model as described in Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

Transient simulations and a qualitative evaluation have been used to conclude that the differences would have a negligible effect on the **AP1000** plant SBLOCA analysis results, leading 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.

AP1000 PLANT INCORRECT DVI LINE RESISTANCE FOR DEDVI LINE BREAK

Background

An error was discovered in the input associated with the **AP1000**® plant small break loss-of-coolant accident (SBLOCA) analysis direct vessel injection (DVI) line resistance for the double-ended DVI (DEDVI) line break scenarios. The line resistance associated with the faulted DVI side piping inadvertently included items which were no longer connected to this piping segment as a result of the break location modeled. The corrected inputs related to the broken side DVI line resistance have been evaluated for impact on current licensing-basis analysis results described in Revision 19 of the **AP1000** plant Design Control Document (DCD) and Revision 1 of the **AP1000** plant Core Reference Report (CRR). This change represents a Non-Discretionary Change to the Evaluation Model as described in Section 4.1.2 of WCAP-13451 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

Qualitative evaluations have been used to conclude that the broken side DVI line resistance error would have a negligible effect on the **AP1000** plant DEDVI line break SBLOCA analysis results, leading 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.

GENERAL CODE MAINTENANCE

Background

Various changes have been made to enhance the usability of the codes and to help preclude errors in analyses. This includes items such as 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 (Reference 1).

Affected Evaluation Model(s)

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

Estimated Effect

The nature of the code enhancements implemented leads to an estimated peak cladding temperature impact of 0°F for the **AP1000®** plant.

References

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

AP1000 PLANT PASSIVE CORE COOLING SYSTEM CHECK VALVE INPUT UPDATE

Background
Historically, generic input values have been used to describe the behavior of the **AP1000**® plant passive core cooling system check valves for the small break loss-of-coolant accident (SBLOCA) analyses. In order to ensure that the behavior of the **AP1000** plant continues to be accurately captured, the check valves will be modeled in more detail (e.g. opening/closing characteristics, resistances, etc.) moving forward as needed.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451 (Reference 1).

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

Estimated Effect
This forward fit change is being implemented as needed on an analysis specific basis and as such, it is judged to have a PCT impact of 0°F on existing analyses.

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

Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break

Plant Name: AP1000
Utility Name: Westinghouse Nuclear Power Plants
Revision Date: 2/5/2015

Analysis Information

EM:	ASTRUM (2004)	**Analysis Date:**	5/9/2008	**Limiting Break Size:**	Split
FQ:	2.6	**FdH:**	1.75		
Fuel:	RFA	**SGTP (%):**	10		
Notes:					

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	1837	1	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . Evaluation of Pellet Thermal Conductivity Degradation and Peaking Factor Burndown	139	2	
2 . Revised Heat Transfer Multiplier Distributions	11	3	
3 . Error in Burst Strain Application	23	4	
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2014 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	2010		

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

References

1 . APP-GW-GL-700, Tier 2, Chapter 15, Rev. 19, "AP1000 Design Control Document: Accident Analyses," 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:

None

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Small Break

Plant Name: AP1000
Utility Name: Westinghouse Nuclear Power Plants
Revision Date: 2/5/2015

Analysis Information

EM:	NOTRUMP-AP	**Analysis Date:**	8/23/2002	**Limiting Break Size:**	10 Inch
FQ:	2.6	**FdH:**	1.65		
Fuel:	RFA	**SGTP (%):**	10		
Notes:					

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	**1370**	**1**	**(a)**
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . Adiabatic Heat-up Calculation	264	2	(a)
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2014 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	1634		

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

References

1 . APP-GW-GL-700, Tier 2, Chapter 15, Rev. 19, "AP1000 Design Control Document: Accident Analyses," 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.