November 14, 2016

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
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SUBJECT: AUDIT REPORT REGARDING AUDIT TO REVIEW THERMAL HYDRAULIC CODE CASES IN TOPICAL REPORT WCAP-17788-P, “COMPREHENSIVE ANALYSIS AND TEST PROGRAM FOR GSI-191 CLOSURE” (CAC NO. MF6536)

Dear Mr. Nowinowski:

By letter dated July 17, 2015, the Pressurized Water Reactor Owners Group (PWROG) submitted topical report (TR) WCAP-17788-P, “Comprehensive Analysis and Test Program for GSI-191 [Generic Safety Issue 191] Closure” (Agencywide Documents Access and Management System Accession No. ML15210A668) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

The NRC staff conducted a regulatory audit to increase their level of knowledge and understanding of the topic and associated methodologies between January 27 and March 11, 2016, at the Westinghouse Electric Company office in Rockville, MD and AREVA Inc. office in Washington, DC. The audit results will provide additional support for the safety evaluation of the TR being conducted by the NRC staff. The audit report is enclosed.

If you have any questions, please contact me at 301-415-4053 or by e-mail at Jonathan.Rowley@nrc.gov.

/RA/

Jonathan Rowley, Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulations

Project No. 694

Enclosure:
Audit Report

Mr. W. Anthony Nowinowski, Program Manager
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AUDIT REPORT

EVALUATION OF THERMAL HYDRAULIC MODELS AND METHODOLOGIES

USED IN PRESSURIZED WATER REACTOR OWNERS GROUP

TOPICAL REPORT WCAP-17788-P

1. Scope and Purpose

By letter dated July 17, 2015, the Pressurized Water Reactors Owners Group (PWROG) submitted licensing topical report (TR) WCAP-17788-P, "Comprehensive Analysis and Test Program for [Generic Safety Issue] GSI-191 Closure" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15210A668). The TR is an approach to define an in-vessel fibrous debris limit and provides a means for increasing the approved fibrous debris limit used by licensees to resolve GSI-191.

The thermal hydraulic (TH) analysis in TR WCAP-17788-P uses various codes and methodologies to simulate core inlet blockage caused by debris. This audit allowed the U.S. Nuclear Regulatory Commission (NRC) staff to examine and better understand the TH methodologies, models, and approaches presented in Volume 4 of the TR by examining calculation results from code runs and sensitivity studies, reviewing plant model changes and input variable calculations, and confirming major applied modeling conditions and assumptions.

Specific topics discussed during the audit included:

- introduction presentation by the PWROG,
- results of sensitivity studies requested by the NRC staff in the audit plan,
- review of calculation notes for barrel/baffle and Upper Head Spray Nozzle (UHSN) resistances for all four plant categories,
- examination of validity of major assumptions,
- Investigation of the t_{block} error in S-RELAP5 reporting for the Combustion Engineering (CE) cases in Volume 4 Section 10,
- review of results of the code-to-code comparison for the Westinghouse Electric Company (Westinghouse) downflow plant category as described in Volume 4, Section 5.4, and
- review of the quality assurance process for the models used in the analyses as it relates to the regulatory requirements in Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR),

The audit was held at both the Westinghouse office in Rockville, Maryland and the AREVA Inc. (AREVA) office in Washington, DC between January 27 and March 11, 2016. A list of NRC and PWROG staff that were present during the audit is included below.

NRC Audit Team:

- Ashley Smith, Safety Issues Resolution Branch Technical Reviewer, NRR
- Steve Smith, Safety Issues Resolution Branch Technical Reviewer, NRR
- Andrea Russell, Safety Issues Resolution Branch Technical Reviewer, NRR

- Victor Cusumano, Chief, Safety Issues Resolution Branch, NRR
- Vesselin Palazov, Contractor, Information Systems Laboratories
- Jonathan Rowley, PWROG Project Manager, NRR

PWROG Staff:

- Tim Croyle, Manager SEEI, Westinghouse
- Jeff Brown, PWROG GSI-191 Team Lead, APS
- James Spring, Senior Engineer, Westinghouse
- Gordon Wissinger, Consulting Engineer, Areva
- Kurt Flaig, PWROG GSI-191 Team
- Adam Spontarelli, Areva
- Phil Grissom, PWROG/Southern Nuclear
- Dana Nee, PWROG/Dominion
- Ken Greenwood, Areva
- David White, Areva
- Jim Andrachek, Westinghouse
- Brett Kellerman, Westinghouse
- Jeff Kobelak, Westinghouse

2. Documents Audited

The documents listed in this section were viewed by the NRC staff during the audit.

1. CN-LIS-06-217, "Byron/Braidwood (CAE/CBE/CCE/CDE) BELOCA ASTRUM: Unit 2 Initial Steady State and Unit 1 and Unit 2 Confirmatory Studies," April 2007.

2. CN-LIS-07-69, "Byron/Braidwood (CAE/CBE/CCE/CDE) BELOCA ASTRUM: Model Update through Unit 1 Initial Transient," April 2007.

3. CN-LIS-00-7, Rev. 0, "Byron/Braidwood Units 1&2 Uprate BELOCA: Model Development," March 2000.

4. CN-LIS-10-5, "North Anna Unit 2 (VGB) ASTRUM BELOCA – WCOBRA/TRAC Vessel Model Development and Steady State Shakedown," May 2010.

5. CN-LIS-10-12, "1-D Loop Model Inputs for North Anna Units 1 and 2 (VRA/VGB) ASTRUM BELOCA," February 2010.

6. RPVSA-86-733, "CAE THRIVE Analysis for Byron Unit 1," 1986.

7. CN-LIS-14-19, Rev. 0, "WCOBRA/TRAC Core Blockage Sensitivity Studies to Support GSI-191," August 2014.

8. LTR-FSE-13-52, Rev. 1, "Barrel/Baffle and Upper Head Spray Nozzle Loss Coefficient Data in Support of Task 1, Activity 1 of PA-SEE-1090: GSI-191 Thermal-Hydraulic Core Blockage Study," March 2015.

9. LTR-LIS-15-235, “CE Barrel-Baffle Loss Coefficient Data in Support of Task 1, Activity 1 of PA-SEE-1090:G GSI-191, Thermal-Hydraulic Core Blockage Study,” June 2015.

10. LTR-FSE-13-54, “Meeting Minutes from the June 18, 2013 GSI-191 Peer Review of the Westinghouse Upflow Barrel/Baffle Plant Design WCOBRA/TRAC Thermal-Hydraulic Analysis in Support of Activity 1, Task 1 of PA-SEE-1090,” August 2013.

11. CN-LIS-13-25, “WCOBRA/TRAC Core Blockage Study for Downflow Plant Design to Support GSI-191,” June 2014.

12. CN-LIS-13-20, “WCOBRA/TRAC Core Blockage Study for Upflow Plant Design to Support GSI-191,” October 2013.

13. ANP-LIS-13-7, “Analysis Plan Supporting PA-SEE-1090 Task 1, Activity 1: System Evaluations to Support GSI-191 Closure.”

14. EMF-CC-097, “S-RELAP5 User’s Manual.”

15. EMF-2103, Rev. 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” August 2001.

16. EMF-2100, “S-RELAP5 Models and Correlations Code Manual.”

17. 32-9240737-001, “GSI-191 Core Blockage Calculations for CE Upflow Baffle Plants.”

18. 32-9150341-000, “SONGS Unit 3 Cycle 17 S-RELAP5 Base Deck Input Development for RLBLOCA.”

19. 51-9142152-000, “AREVA Support for GSI-191 Upflow Analysis.”

20. 32-9194826-000, “GSI-191 Alternate Flow Paths for B&W Plants.”

21. 32-9223655-000, “GSI-191 Additional Alternate Flow Path Studies for B&W Plants.”

22. 32-1118279-00, “Baffle Slot/Hole Velocity Predictions for 177 FA Plants.”

23. 4310192-00, “BWNT LOCA- BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants.”

Quality Assurance Procedures

24. WEC 3.2.6, “Design Analysis”
25. WEC 3.3.3, “Design Verification”
26. WEC 3.6.6, “Simple Application Computer Programs”
27. WEC 6.1, “Document Control”
28. ES 3.2.1, “Creating and Verifying a Design Document”
29. WEC 3.6.1, “Computer Software Development”
30. WEC 3.6.2, “Validation of Computer Software”
31. CN-DWGVER-01-17
32. ANP 0402-01, “Calculations”

33. ANP 0902-30, “Management and Use of Engineering Applications Software”
34. ANP 0902-28, “Development of Engineering Applications Software”

Drawings

35. Westinghouse 675C89
36. Westinghouse 685J102_1
37. Westinghouse 685J102_2
38. Westinghouse 685J102_3
39. Westinghouse 685J102_4
40. Westinghouse 685J102_5
41. Westinghouse 685J102_6
42. Westinghouse 685J102_7
43. Westinghouse 685J326_1
44. Westinghouse 685J326_2
45. Westinghouse 685J326_3
46. Westinghouse 6117E01_1
47. Westinghouse 6117E01_2
48. Westinghouse 6117E01_3
49. Westinghouse 6118E09
50. ANP J-23866-164-018, Rev. 62, “Core Support Plate Assy.”
51. ANP J-23866-164-025, Rev. 05, “Core Shroud Assemblies”
52. ANP J-23866-164-003, Rev. 08, “Reactor Internal Assembly”
53. ANP J-23866-164-016, “Core Support Barrel”
54. ANP J-23866-164-033, Rev. 06, “Fuel Alignment Plate”
55. ANP 32-1178279-000, “Lower Grid Top Rib Section”

The materials presented by PWROG during the audit were both preliminary (not quality assured) and proprietary. Therefore, they will not be included with this audit report.

3. <u>Audit Activities and Observations</u>

3.1 <u>Introduction presentation by the PWROG</u>

The PWROG presented an overview of TR WCAP-17788 as an introduction to the audit. The TR was developed following Elements 1-4 of Regulatory Guide 1.203, “Transient and Accident Analysis Methods.” Of significant importance, the new concepts in the TR were presented, which included the reliance on alternate flow paths (AFPs) to cool the core if the core inlet became blocked with debris and simulation of a debris bed due to changes in pressure.

The PWROG described the bases for selection of the TH codes chosen for the proposed GSI-191 TR methodology. The selection and development of the models used in the methodology went through peer reviews internally, across vendors (Westinghouse and AREVA), and externally (Texas A&M University and the PWROG TIGER team).

Modifications to WCOBRA/TRAC and S-RELAP5 were discussed briefly. The application of the transient form-loss coefficient at the first node of the specified core channels was modified in WCOBRA/TRAC to allow ramping of the applied coefficient (described in Section 5.1 of Volume 4). While S-RELAP5 did not require the same modification as WCOBRA/TRAC since it includes such capability, it did require a modification to determine the pressure drop across locations at which high resistances are applied to simulate the build-up of debris.

Documentation regarding this modification was requested through the request for additional information (RAI) process. RELAP5/MOD2-B&W (Babcock & Wilcox Company) did not have any modifications for the analyses in this TR.

The plant models selected for the analyses for each plant category were chosen based on their pedigree and currently updated status. The power and power density were also taken into account to create a bounding representation of each plant category.

The PWROG described how previous code assessments were relied on and that the steady state AFP bypass fractions calculated at full power for the TR were compared to plant-specific values and, also, independently between vendors and outside sources (i.e., Texas A&M University). To further assess model adequacy, the PWROG performed sensitivity studies that are discussed in Section 3.2 of this audit report.

The input parameters in Tables 6-1 through 6-4 were chosen by the PWROG to encompass the relevant plant parameters for the units considered in each plant category. When a licensee submits an application to use the methodology in WCAP-17788 (after it is approved), that licensee will need to determine which plant category they fall under, justify being inside the bounds of the plant parameters for that plant category, and determine which t_{block} value from Volume 1 will be used in the analyses of their plant. Further justification regarding the bases for choosing the parameters in Tables 6-1 through 6-4 was requested in the formal review process as an RAI in the Volume 4 RAIs at ADAMS Accession No. ML16161A416.

3.2 Results for sensitivity studies requested by the NRC staff in the audit plan

Prior to the audit, the NRC staff requested that the PWROG run sensitivity studies for certain analysis items such as timestep, convergence criteria, and time of core blockage (t_{block}) to determine the effect they had on the results of t_{block}, m_{split}, K_{split}, and K_{max}. Tables 1 and 2 show the studies that Westinghouse and AREVA ran as a result of this request. Note that there were not any sensitivity studies run for the B&W plant category.

Table 1: WEC Sensitivity Matrix (Upflow and downflow plant categories)

Run #	Plant Category	Base Case	Modification	Action
1	W Upflow	1B	Nominal Decay Heat (American Nuclear Society (ANS) 1979 Standard)	Compare to Base Case
2	W Upflow	1A	Uniform Radial Power Profile	Compare to Base Case
3	W Upflow	1B	Uniform Radial Power Profile	Compare to Base Case
4	W Upflow	2B	Modeled Loss-of Coolant Accident (LOCA) Holes	Compare to Base Case
5	W Upflow	1B	Change max timestep to 0.001 sec	Compare to Base Case
6	W Upflow	1B	Change max timestep to 0.003 sec	Compare to Base Case

7	W Upflow	3 (Rev. 1)	Reduce convergence criteria by an order of magnitude	Compare to Base Case
8	W Upflow	1B	t_{block} = 113 min	Compare to Base Case
9	W Upflow	1B	t_{block} = 173 min	Compare to Base Case
10	W Upflow	1B	t_{block} = 203 min	Compare to Base Case
11	W Downflow	1A	Nominal Decay Heat (ANS 1979 Standard)	Compare to Base Case
12	W Downflow	2A	K_inlet = 8e5	Compare to Base Case
13	W Downflow	2A	K_inlet = 9e5	Compare to Base Case
14	W Downflow	2A	K_inlet = 10e5	Compare to Base Case
15	W Downflow	--	Areva Confirmatory Case	Compare to Areva Case
16	W Downflow	1A	Change max timestep to 0.001 sec	Compare to Base Case
17	W Downflow	1A	Change max timestep to 0.003 sec	Compare to Base Case
18	W Downflow	3 (Rev. 1)	Reduce convergence criteria by an order of magnitude	Compare to Base Case
19	W Downflow	1A	tblock = 140 min	Compare to Base Case
20	W Downflow	1A	tblock = 290 min	Compare to Base Case
21	W Downflow	1A	tblock = 320 min	Compare to Base Case

The base cases for the Westinghouse plant categories presented in the TR used the decay heat x 1.2 (ANS 1971) model, peak power profile, convergence criteria of 1x10^{-3}, and a timestep of 0.002 seconds.

Table 2: Areva Sensitivity Matrix (CE plant category)

Run #	**Status**	**Base Case**	**Modification**	**Result/ Expectation**
1	In WCAP	--	None • t_block = 20,000 sec • DH = 1.2*ANS71 • Timestep = 0.002 sec during uncovering	Peak Cladding Temperature (PCT) = 746°F
4	Not documented	1	Changed t_block to 15,000 sec	PCT = 954°F

Run #	Status	Base Case	Modification	Result/ Expectation
5	Not documented	1	Changed t_block to 16,000 sec	PCT = 890°F
6	Not documented	1	Changed t_block to 17,000 sec	PCT = 857°F
7	Not documented	1	Changed t_block to 18,000 sec	PCT = 734°F
8	Not documented	1	Changed t_block to 19,000 sec	PCT = 852°F
8	Not documented	1	Changed t_block to 19,200 sec	PCT = 808°F
10	Not documented	1	Changed t_block to 22,000 sec	No heatup
11	Not documented	1	Changed t_block to 24,000 sec	No heatup
31	Not documented	1	Increased timestep to 0.005 sec during uncovering	PCT =680°F
32	Not documented	1	Reduced timestep to 0.001 sec during uncovering	PCT =700°F
33	Not documented	1	Reduced timestep to 0.0005 sec during uncovering	PCT =650°F
25	Not documented	4	Changed DH to 1.0*ANS71	No heatup
27	Not documented	4	Changed DH to 1.0*ANS71, reduced t_block to 8000 sec	PCT = 613°F

The base cases for the Combustion Engineering, Inc. (CE) plant category presented in the TR used the decay heat x 1.2 (ANS 1971) model and peak power profile. The maximum timestep used was 0.002 seconds.

Results of the sensitivity study for the decay heat model confirmed its expected impact on the predictions. Results of the peak power profile, convergence, and timestep sensitivity studies were discussed at the audit. The results showed that the cases used in the submittal were conservative in comparison to corresponding cases examining sensitivities to selected inputs. For t_{block} sensitivity studies, the PCT generally decreased as the time of blockage increased. As a note, when determining the blockage time to use in the submittal, a time was chosen which resulted in a PCT close to, but below, the acceptance criteria of 800°F. The PWROG explained that it was not necessary to optimize the blockage time to get the longest blockage time that resulted in a PCT less than 800°F because of the amount of conservatism in other parts of the analyses.

Selected computation results related to the prediction of entrainment, void fraction, and quality of the flow out of the break were examined and discussed during the audit. Further explanation and assessment of this topic was asked through the RAI process.

3.2.1 Review calculation notes for barrel/baffle and UHSN resistances for all four plant categories

The NRC staff reviewed the calculation notes for the barrel/baffle and UHSN resistances for each of the plant categories. This was done by first examining plant drawings for a single unit in order to determine the flow area for each type of AFP defined in the calculation. Then, the assumptions for the loss coefficients for each type of AFP were reviewed. Discussions with the respective Westinghouse and AREVA engineers regarding the inputs and assumptions for the calculation notes were useful for gathering background information and understanding the calculation documents and applied assumptions. Lastly, spot checks of the resulting resistances were completed.

The PWROG performed testing to show that fibrous debris would not block a flow path using two hole sizes over a range of flow conditions. Based on plant drawings used for development of the AFP resistances, some of the credited flow paths are smaller than what was tested. The NRC staff asked the PWROG to explain how the hole dimensions tested were representative of the AFP geometries in all four plant categories through the RAI process.

RAIs were developed as a result of reviewing the barrel/baffle and UHSN resistance calculation notes. These RAIs were included in the Volume 4 RAIs at ADAMS Accession No. ML16161A416 and the Volume 6 RAIs at ADAMS Accession No. ML16075A200.

3.3 Examine validity of major assumptions

The hot leg break methodology in Volume 4 includes assumptions for the inputs in Tables 6-1 through 6-4. During the audit, the NRC staff reviewed documentation related to the calculation of the AFP resistances and discussed how other inputs were developed and the assumptions that were made.

Flow exchange patterns between parallel core channels along both the axial and radial directions were examined for selected calculations. It was noted that the effect of the presence of grid spacers was not accounted for in modeling of the core cross-flows in the TH models using WCOBRA/TRAC. This can be non-conservative with respect to crossflows. This issue was included in the RAI process with other Volume 4 RAIs.

The Westinghouse upflow, Westinghouse downflow, and CE plant categories used a large-break loss-of-coolant accident (LBLOCA) evaluation model for the analysis. Conversely, the B&W plant category used a small-break loss-of-coolant accident (SBLOCA) evaluation model. The adequacy of using a SBLOCA model to evaluate long term core cooling was discussed by the audit team and PWROG. The PWROG explained that a SBLOCA model was used because of computational difficulties associated with using a LBLOCA model to analyze extended transients using RELAP5/MOD2-B&W. Further justification and examination of this assumption was requested by the NRC staff through the RAI process.

3.4 Investigate the t_{block} error in SRELAP5 reporting for the CE cases in Volume 4 Section 10

During a conference call on November 18, 2015, the PWROG provided the NRC staff with information regarding an error in the analysis for the CE plant category. More information about the updated results due to this error were discussed at the audit.

The error was found by the PWROG while attempting to improve margin in the analysis for the CE plant category. A condition report was opened once the error was found. While re-examining the analysis to determine where margin could be gained, the PWROG noticed that flow through the average channel was occurring after the timing of core was assumed to be blocked. The original input value for the core inlet loss coefficient (1E9) for CE plants did not result in a condition of complete core inlet blockage at t_{block}. Selection of this value for the loss coefficient was previously verified to produce a condition of very low flow (essentially zero for TH analyses purposes). However, after further review by the PWROG, the core inlet flow rate is not close enough to zero to not have an effect on the core temperature.

Analysis completed using a higher loss coefficient (1E20) effectively produced a condition of no flow at the core inlet. This, however, affects the t_{block} time. An erratum to include the updated results due to this error was submitted to the NRC staff on February 12, 2016 (ADAMS

Accession No. ML16088A036). The draft version of this erratum was discussed during the audit. The calculation notebooks had not been updated to include the correction and associated results.

3.4.1 Review results of the code-to-code comparison for the Westinghouse downflow plant category as described in Volume 4, Section 5.4

The PWROG summarized TH analysis completed independently by AREVA and by Westinghouse for the downflow plant category during the initial development of the hot leg break methodology. Two different codes were used for this analysis: S-RELAP5 and WCOBRA/TRAC. The plant and transient condition analyzed in both studies were identical; however, the applied plant models were developed independently following different methods and techniques. To better understand the effects from using different methodologies on the prediction of key results from the analyses, additional information was requested by the NRC staff through the RAI process.

3.4.2 Review the quality assurance process for the models used in the analyses as it relates to the regulatory requirements in 10 CFR Appendix B

The implementation of changes in WCOBRA/TRAC and RELAP5 are performed using a defined process for single application computer programs including development, implementation, and validation processes. The NRC staff concluded that the quality assurance process used by the PWROG was acceptable for meeting the associated regulatory requirements. With respect to approved methods, the NRC staff confirmed that the PWROG was in compliance with the approved procedures and processes.

4. Closing Briefing

Throughout the audit, the PWROG presented information regarding the hot leg break methodology in WCAP-17788-P. The NRC staff has prepared RAIs pertaining to information discussed during the audit that will be required to make safety conclusions and items to which a clear conclusion was not reached between the NRC audit team and the PWROG during the audit. Some items in this regard include:

- Results of additional sensitivity studies
- Clear documentation related to calculation notes for AFPs
- Basis for inputs and assumptions of the hot leg break methodology
- AFP testing used to bound plant configuration
- Adequacy of B&W plant model using a SBLOCA evaluation model
- Capabilities of applied codes to predict voids and entrainment in the core post-LOCA for large breaks
- Adequacy of modeling crossflows in Westinghouse, CE, and B&W plants models

Overall, the audit was effective, informative, and productive and the objectives defined in the audit plan were accomplished. The information, knowledge, and understanding obtained during the audit enhanced and supported the process of RAI development and will assist in review of the TR.