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Our ref: DCP/NRC2335

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Subject: AP1000 Responses to Requests for Additional Information (SRP15)

Westinghouse is submitting responses to the NRC request for additional information (RAI) on SRP Section 15. These RAI responses are submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in the responses is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Enclosure 1 provides the response for the following RAIs:

RAI-SRP15.6.5-SRSB-08
RAI-SRP15.6.5-SRSB-10
RAI-SRP15.6.5-SRSB-12
RAI-SRP15.6.5-SRSB-13
RAI-SRP15.6.5-SRSB-15
RAI-SRP15.6.5-SRSB-16

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk

Robert Sisk, Manager
Licensing and Customer Interface
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/Enclosure

1. Responses to Requests for Additional Information on SRP Section 15

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NRO

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ENCLOSURE 1

Responses to Requests for Additional Information on SRP Section 15

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP15.6.5-SRSB-08
Revision: 0

Question:

Revision 15 of DCD subsection 3.9.2.3 states that "The coolant velocity in the downcomer annulus between the core barrel and the reactor vessel wall is lower in the AP1000 design than in previous three-loop plants because the AP1000 has no thermal shield or neutron pads in the annulus to restrict this flow." On page 4-3 of WCAP-16716, Rev. 2, the statement "because the AP1000 has no thermal shield or neutron pads in the annulus to restrict this flow" is deleted.

(a) Is this statement still true with the addition of the neutron panels? Since the neutron panels were added to the AP1000 design, it would be expected that the coolant velocity would now be increased. Explain why the coolant velocity in AP1000 downcomer annulus is still lower.

(b) Confirm that the neutron panels have been included in the fuel deformation analysis due to combined LOCA/seismic loads to demonstrate compliance with 10 CFR 50.46(b)(4) acceptance criterion of coolable geometry.

Westinghouse Response:

(a) As indicated in WCAP-16716, Rev. 2, the statement "...because the AP1000 has no thermal shield or neutron pads in the annulus to restrict this flow" is no longer correct and was deleted from the DCD. While the AP1000 vessel flow rate is higher than that of similar three loop plants, the AP1000 reactor vessel inner diameter has been increased by two inches to 159 inches. The resulting flow area increase is sufficient to yield an AP1000/similar 3 loop plant downcomer flow velocity ratio of approximately 0.89. Therefore, the AP1000 downcomer flow velocity, including the addition of the neutron panels, is lower than similar three-loop plants.

(b) The fuels structural analysis was not revised to include modeling the neutron panels or to include the LOCA hydraulic forcing functions done with the neutron panels incorporated. The LOCA forcing function is insignificant with respect to the seismic loads for the fuel structural analysis and the addition of the neutron panels has negligible effects on the fuel structural analysis.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP15.6.5-SRSB-10
Revision: 0

Question:

In Section 6.5.2 of the response to RAI-TR29-SRSB-01 (Westinghouse letter DCP/NRC2128, April 29, 2008), Westinghouse provides an assessment of pressure loss due to flow skirt and neutron panel addition. It states that the steady state pressure drop increase due to the flow skirt and neutron panel addition is mostly offset by the reduction in pressure drop through the inlet nozzle. The change that results in the reduction of the inlet nozzle pressure loss (from 10.72 psi to 5 psi) is not identified in APP-GW-GLE-026, and appears to be a post-DCD Revision 15 change.

Provide a discussion of how the revised inlet nozzle pressure drop was obtained. Confirm that this change is included in the WCOBRA/TRAC model for DCD revision 16, and that there are no other design changes that could impact the modeling of the AP1000 for best-estimate LBLOCA analyses for the proposed revision to DCD Revision 16.

Westinghouse Response:

The original AP1000 thermal-hydraulic analysis of the reactor internals was based on historical correlations and turbulent flow relationships that have been traditionally used, along with the governing flow rates, to define the pressure drops at various locations in the reactor internals, such as the reactor vessel inlet nozzles and lower plenum. Subsequent to this analysis, a CFD analysis of the vessel inlet, downcomer, lower plenum and lower core support plate was performed to assess the effectiveness of a flow skirt in improving the core inlet flow distribution. Although the two analyses were performed for different purposes, it was decided to compare the calculated CFD pressure drops with the results of the original thermal-hydraulic analysis. This comparison revealed that, while the inlet-to-core pressure drops were approximately the same for both analyses, the distribution of pressure drops was different. In particular, the CFD-calculated inlet nozzle loss was smaller than the original inlet nozzle loss and the CFD-calculated lower plenum loss was larger than the original lower plenum loss.

While the inlet-to-core pressure drop is approximately the same for both calculations, the core bypass flow through the spray nozzles is slightly higher for the CFD-calculated inlet nozzle loss. In the interests of conservatism, therefore, a decision was made to base the revised inlet-to-core pressure losses on the CFD-calculated results.

Westinghouse letter DCP/NRC2128 discusses the impact of the flow skirt and other design changes on the DCD Revision 15/Revision 16 large break LOCA analysis. The revised inlet nozzle pressure drop and addition of the flow skirt and neutron pads are included in the WCOBRA/TRAC model for DCD Revision 17. The AP1000 ASTRUM analysis steady state calculation was demonstrated to meet the steady state acceptance criteria specified in WCAP-

16009-P-A Table 12-6, consistent with the methodology presented in WCAP-16009-P-A Section 12-4-1.

Westinghouse has processes and procedures in place to assess the impact of design changes on the large break LOCA analyses.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP15.6.5-SRSB-12
Revision: 0

Question:

The NRC SER for the AP1000 (NUREG-1793), Section 15.2.6.5.2 "Large Breaks," addressed the peak cladding temperature (PCT) limitation concerning the elimination of the CMT and the PRHR system to identify the PCT sensitivities, and to add the blowdown and reflood PCT impacts as a bias to their respective 95-percent PCT results. Revision 17 of AP1000 DCD section 15.6. 5.4A.5 states that the AP1000 large-break LOCA analysis complies with the restrictions in NUREG-1512 and WCAP-16009-A, and that: "Previous AP1000 sensitivity calculations evaluated the sensitivity to modeling of the CMT and PRHR relative to a baseline case. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the baseline case. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the baseline case. The ASTRUM methodology samples the parameters ranged in the global model matrix of calculations, and the final 95% uncertainty calculations have been performed for AP1000. Further, local and core –wide cladding oxidation values have been determined using the methodology approved in Reference 32 [WCAP-16009-A]." It is not clear if the proposed model described in Appendix C of APP-GW-GLE-026 was used for these studies, and the maximum local oxidation (MLO) and core-wide oxidation (CWO) biases are not addresses.

Address this limitation for the proposed AP1000 model and include the MLO and CWO sensitivities and resulting biases. Perform additional analyses, as necessary, to establish the MLO and CWO biases.

Westinghouse Response:

The results of the CMT and PRHR sensitivity studies which supported the approved DCD Revision 15 analysis were assumed to be applicable to the ASTRUM analysis submitted in APP-GW-GLE-026 because the design changes did not impact the CMT or PRHR, and, per WCAP-15644-P Revision 2 Section 2.1, "The PIRT review of the key LBLOCA phenomena presented in <WCAP-15613> indicates that, as is true for AP600, the unique passive safety systems play almost no role in the plant's response during the PCT excursion of a LBLOCA event because the transient is so rapid."

In response to this RAI, sensitivity calculations on the AP1000 ASTRUM analysis reference transient, as described in Appendix C of APP-GW-GLE-026, were performed. The WCOBRA/TRAC peak cladding temperature (PCT) results of the CMT inoperable study and the reference transient are presented in Table 1. When the PRHR inoperable was modeled, the maximum timestep allowed during the transient was reduced by 0.0001 s (from 0.0008 s to 0.0007 s) in order for the calculation to execute to completion. Therefore, a reference transient sensitivity case was performed with the same maximum allowable timestep during the transient as the PRHR inoperable case. The results of the PRHR inoperable study and the reference transient timestep sensitivity study are presented in Table 2. The results in Table 1 and Table 2 show that with the CMTs isolated the PCT is lower than the reference transient; the PCT effect of the PRHR operation on the reference transient is minimal.

These sensitivity studies on the reference transient provide PCT sensitivity results from WCOBRA/TRAC but do not provide a quantitative MLO or CWO sensitivity. In the ASTRUM methodology the HOTSPOT code is also used to establish the MLO and CWO from the 124 uncertainty analysis calculations performed after the reference transient is determined. Note that the AP1000 shows significant margin to the MLO and CWO limits and it is not appropriate to apply penalties to the analysis results for presuming no operation of this safety-related equipment.

Table 1. Summary of Reference Transient PCT Sensitivity to CMT Inoperable

	Reference Transient	CMT Inoperable
WCOBRA/TRAC Hot Rod PCT °F	1704	1666

Table 2. Summary of Reference Transient PCT Sensitivity to PRHR Inoperable

	Reference Transient Timestep Sensitivity	PRHR Inoperable
WCOBRA/TRAC Hot Rod PCT °F	1698	1700

Design Control Document (DCD) Revision:

Revise Section 15.6.5.4A, fifth paragraph:

> For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology is applied as described in Reference 34. The best-estimate large-break LOCA analysis complies with the stipulated applicability limits in ~~the Reference 3 approval and~~ the Reference 32 approval.

Revise Section 15.6.5.4A.5, first paragraph:

> For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology documented in Reference 34 is applied. The AP1000 large-break LOCA analysis complies with the restrictions in ~~Reference 3 and~~ Reference 32. ~~Previous~~ AP1000 sensitivity calculations evaluated the sensitivity to modeling of the CMT and PRHR relative to the reference transient configuration. ~~a baseline case.~~ A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. ~~baseline case.~~ Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 2°F higher than the reference transient configuration. ~~lower than the PCT of the baseline case.~~ The ASTRUM methodology samples the parameters ranged in the global model matrix of calculations, and the final 95% uncertainty calculations have been performed for AP1000. ~~Further, l~~Local and core-wide cladding oxidation values have been determined using the methodology approved in Reference 32.

PRA Revision:

None

Technical Report (TR) Revision:

Revise APP-GW-GLE-026 Section 2, eleventh paragraph:

> This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of WCAP-16009-P-A [3] as applicable to the ASTRUM methodology. ~~This analysis is in accordance with the applicability limits and usage conditions for AP1000 Best Estimate LBLOCA analysis defined in NUREG-1793 Section 21.6.3.5 and Appendix 21.A.12 [6] as applicable to the ASTRUM methodology.~~ Section 13-3 of WCAP-16009-P-A [3] was found to make acceptable disposition of each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this WCAP. Per Section 3, the

best-estimate large-break LOCA analysis and associated model for AP1000 is applicable to the AP1000 standard design.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP15.6.5-SRSB-13
Revision: 0

Question:

Table 15.6.5-4 of Revision 17 to AP1000 DCD provides the major plant parameter assumptions used in the best-estimate large break LOCA analysis. Please address the following apparent inconsistencies:

(a) Accumulator pressure (P_{ACC}): 670.0 psia $\leq P_{ACC} \leq$ 765.8 psia against DCD technical specification (TS) 3.5.1 "nitrogen cover gas pressure in each accumulator is ≥ 637 psig (651.7 psia) and ≤ 769 psig (783.7 psia)."

(b) Accumulator water volume (V_{ACC}): 1680 $ft^3 \leq V_{ACC} \leq$ 1720 ft^3 against TS 3.5.1 "borated water volume in each accumulator is ≥ 1667 cu. ft., and ≤ 1732 cu. ft."

(c) The accumulator volume range, presented in Table 4 of APP-GW-GLE-026), is not included in Table 15.6.5-4. Since this range is specified in the TS, this table should be updated to include the range to support the LCO.

Westinghouse Response:

The incorrect accumulator pressure and liquid volume ranges were assumed in the AP1000 best-estimate large break LOCA ASTRUM analysis. An evaluation was performed via the reanalysis of the top 10 HOTSPOT peak cladding temperature (PCT) cases from the ASTRUM runset. Maintaining the seed used in the ASTRUM analysis, the accumulator pressures and liquid volumes for the evaluation calculation cases were determined based on the original sampling for each run and the revised ranges. The evaluation showed that the PCT, maximum local oxidation (MLO) and core-wide oxidation (CWO) results reported in APP-GW-GLE-026 are applicable over the technical specification ranges. A 0° F PCT penalty is assessed for this set of closely-related errors.

Design Control Document (DCD) Revision:

Table 15.6.5-4 of Revision 17 to the AP1000 DCD will be updated as follows.

Table 15.6.5-4. Major Plant Parameter Assumptions Used in the Best-Estimate Large-Break LOCA Analysis

Parameter	Value
Plant Physical Configuration	
• Steam generator tube plugging level	≤ 10% (10% tube plugging bounds 0%)
• Hot assembly location	Under support column (Bounds under open hole or guide tube)
• Pressurizer location	In intact loop (Bounds location in broken loop)
Initial Operating Conditions	
• Reactor power	Core Power ≤ 1.01* 3400 MWt
• Peak linear heat rate	$F_Q \leq 2.6$
• Hot rod assembly power	$F_{\Delta H} \leq 1.75$
• Hot assembly power	$P_{HA} \leq 1.683$
• Axial power distribution	See Figure 15.6.4A-13
• Peripheral assembly power	$0.2 \leq P_{LOW} \leq 0.8$
Fluid Conditions	
• Reactor coolant system average temperature	573.6 –7.5°F ≤ T_{AVG} ≤ 573.6 + 7.5°F
• Pressurizer pressure	2250 ± 50 psia
• Pressurizer level (water volume)	1000 ft^3 (nominal)
• Accumulator temperature	50°F ≤ T_{ACC} ≤ 120°F
• Accumulator pressure	651.7 psia < P_{ACC} < 783.7 psia~~670.0 psia < P_{ACC} < 765.8 psia~~
• Accumulator water volume	1667 ft^3 ≤ V_{ACC} ≤ 1732 ft^3
Reactor Coolant System Boundary Conditions	
• Single failure assumption	Failure of one CMT isolation valve to open

• Offsite power availability	Available (bounds loss of offsite power at time zero)
• Reactor coolant pump automatic trip delay time after receiving S-signal	4 s
• Containment Pressure	Bounded (minimum)

PRA Revision:

None

Technical Report (TR) Revision:

Table 5 of APP-GW-GLE-026 will be updated as follows.

Table 5. Major Plant Parameter Assumptions Used in the Best-Estimate Large-Break LOCA Analysis for AP1000

Parameter	Value
Plant Physical Description	
• Steam Generator Tube Plugging Level	≤ 10%
Plant Initial Operating Conditions	
• Reactor Power	Core Power ≤ 1.01* 3400 MWt
• Peaking Factors	$F_Q \leq 2.6$ $F_{\Delta H} \leq 1.75$
• Axial Power Distribution	See Figure 1
• Low Power Region Relative Power (P_{LOW})(1)	$0.2 \leq P_{LOW} \leq 0.8$
• Typical Cycle Length	18 months
Fluid Conditions	
• Vessel Average Fluid Temperature	573.6 –7.5°F ≤ T_{AVG} ≤ 573.6 + 7.5°F
• Pressurizer Pressure	2250 ± 50 psia
• Reactor Coolant Flow	≥ 148,000 gpm
• Accumulator Temperature	50°F ≤ T_{ACC} ≤ 120°F
• Accumulator Pressure	651.7 psia < P_{ACC} < 783.7 psia~~670.0 psia ≤ P_{ACC} ≤ 765.8 psia~~
• Accumulator Water Volume	1667 ft^3 ≤ V_{ACC} ≤ 1732 ft^3~~1680 ft^3 ≤ V_{ACC} ≤ 1720 ft^3~~
• Accumulator Boron Concentration	≥ 2600 ppm
• CMT Temperature	T_{CMT} < 120°F
Accident Boundary Conditions	
• Single Failure Assumption	Failure of one CMT isolation valve to open
• Reactor Coolant Pump Automatic Trip Delay Time After Receipt of S-Signal	4 s
• Offsite Power Availability	Available (bounds loss of offsite power at time zero)
• Containment Pressure	Bounded (minimum)
Note: 1. Average relative power of the 28 peripheral assemblies depicted in Figure 2.	

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP15.6.5-SRSB-15
Revision: 0

Question:

Revision 17 of DCD Subsection 15.6.5.4A.6 states that at 2.2 seconds, credit is taken for receipt of an "S" signal due to High-2 containment pressure. DCD Table 15.0-4a indicates a time delay of 2.2 seconds for "S" signal on High-2 containment pressure assumed for LBLOCA analysis.

(a) It appears that the LBLOCA analyses assume the containment pressure reaches the high-2 pressure setpoint coincident with the initiation of the event. Is it a correct interpretation?

(b) Verify that the minimum containment backpressure and the coincident High-2 signal timing are applicable to the entire break spectrum. Given the lower mass and energy releases for smaller breaks it would seem that these parameters would vary with the break size and type.

Westinghouse Response:

(a) The LBLOCA analysis assumes that the containment pressure high-2 pressure setpoint is reached by 2.2 s after the initiation of the event. Then a time delay of 2 s is assumed for signal processing. Therefore, the core makeup tank (CMT) isolation valves are assumed to begin opening 4.2 s after the initiation of the event; as discussed in DCD Section 15.6.5.4A.3, "A safeguards "S" signal occurs due to containment high pressure at 2.2 seconds of large-break LOCA transients. As a consequence of this signal, after appropriate delays, the PRHR and core makeup tank isolation valves open and containment isolation occurs."

For cases with offsite power available, and the assumed automatic reactor coolant pump trip time delay of 4 s, the reactor coolant pumps are assumed to trip 8.2 s after the initiation of the event.

(b) The AP1000 ASTRUM analysis followed the WCAP-16009-P-A methodology for determination of the conservative containment backpressure approved for standard Westinghouse pressurized water reactors. The reference transient was used to establish the containment pressure response that was applied as a boundary condition in the uncertainty analysis calculations (see WCAP-16009-P-A pg 11-13). The inputs to the containment pressure calculation were skewed in order to obtain a conservative (low) pressure transient.

As discussed in response to part (a), it is assumed that by 2.2 s after break inception, the containment high-2 pressure setpoint is reached. The 2.2 s overestimates the time to reach the high-2 setpoint containment pressure in the event of a nominal double-ended cold leg guillotine large break LOCA. For a nominal double-ended cold leg guillotine large break LOCA, containment pressure calculations skewed to obtain a conservatively low pressure transient show that by 2.2 s after break the containment pressure is more than 24 psia. Although 2.2 s may be insufficient time to reach the high-2 containment pressure setpoint for the smallest breaks sampled as part of the ASTRUM methodology due to the reduced mass and energy release, the small break sizes are non-limiting for the AP1000 as shown in APP-GW-GLE-026 Figure 27.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP15.6.5-SRSB-16
Revision: 0

Question:

Revision 17 of DCD Table 15.6.5-6 shows that the reactor coolant pumps trip at 8.2 seconds into the transient for the LBLOCA limiting PCT/MLO case, and Revision 17 of DCD Subsection 15.6.5.4A.6 states that the reactor coolant pumps automatically trip after a 4 s delay from the actuation of the core makeup tank isolation valves at 8.2 seconds into the transient. This time line appears to be inconsistent with DCD Table 15.0-4a, Revision 17, and Table 5 of APP-GW-GLE-026, which indicate that a time delay of 4.0 seconds of LBLOCA for the reactor coolant pump trip following "S", and therefore the RCPs would trip at 6.2 seconds.

Clarify and update the RCS pump trip delay description as necessary in Table 5, Table 15.0-4a , and Section 15.6.5.4A.6, accordingly.

Westinghouse Response:

See response to RAI-SRP15.6.5–SRSB-15(a) for clarification of the signal logic credited in the LBLOCA analysis for reactor coolant pump trip.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None