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Your ref: Project Number 740
Our ref: DCP/NRC1977

August 21, 2007

Subject: AP1000 COL Responses to Requests for Additional Information (TR #24)

In support of Combined License application pre-application activities, Westinghouse is submitting responses to NRC requests for additional information (RAI) on AP1000 Standard Combined License Technical Report 24, APP-GW-GLR-060, Rev. 0, Reactor Vessel Insulation System – Verification of In-Vessel Retention Design Bases. These RAI responses are submitted as part of the NuStart Bellefonte COL Project (NRC Project Number 740). The information included in the responses is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification.

The responses are provided for requests for additional information RAI-TR24-SPLA-01 through RAI-TR24-SPLA-08. These responses complete all requests received to date for Technical Report 24.

Pursuant to 10 CFR 50.30(b), the responses to requests for additional information on Technical Report 24 is submitted as Enclosure 1 under the attached Oath of Affirmation.

Questions or requests for additional information related to the content and preparation of these responses 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,

Monte D Bartley FOR

A. Sterdis, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

D079

00219psa.doc

NRO

/Attachment

1. "Oath of Affirmation," dated August 21, 2007

/Enclosure

1. Responses to Requests for Additional Information on Technical Report No. 24

cc:	D. Jaffe	-	U.S. NRC	1E	1A
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	M. Moran	-	Florida Power & Light	1E	1A
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	E. Schmiech	-	Westinghouse	1E	1A
	G. Zinke	-	NuStart/Entergy	1E	1A
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ATTACHMENT 1

"Oath of Affirmation"

ATTACHMENT 1

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	)
NuStart Bellefonte COL Project	)
NRC Project Number 740	)

APPLICATION FOR REVIEW OF
"AP1000 GENERAL COMBINED LICENSE INFORMATION"
FOR COL APPLICATION PRE-APPLICATION REVIEW

W. E. Cummins, being duly sworn, states that he is Vice President, Regulatory Affairs & Standardization, for Westinghouse Electric Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this document; that all statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

W E Cummins

W. E. Cummins
Vice President
Regulatory Affairs & Standardization

Subscribed and sworn to
before me this 21st day
of August 2007.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia S. Aston, Notary Public
Murrysville Boro, Westmoreland County
My Commission Expires July 11, 2011
Member, Pennsylvania Association of Notaries

Patricia S. Aston
Notary Public

ENCLOSURE 1

Responses to Requests for Additional Information on Technical Report No. 24

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-TR24-SPLA-01
Revision: 0

Question: The ULPU-2400 Configuration V tests found that the optimal annulus design, from a coolability limit perspective, progresses from ~3" near the bottom of the hemispherical section (after entry through the inlet) gradually widening to ~6" at the upper end of the hemispherical section, and remaining at ~6" through the cylindrical section. However, Figure 4-2 on page 9 of report APP-MN20-Z0R-001 R0 appears to use the neutron shield to form the upper part of the cylindrical section of the Reactor Vessel Insulation System (RVIS) flow path, prior to allowing the External Reactor Vessel Cooling (ERVC) water/steam mixture to exit into the Reactor Pressure Vessel (RPV) cavity. The thickness of the neutron shield in Figure 4-2 appears to "narrow" the annulus (in a stepwise fashion) in this region. Without specific test results for this apparent change to the "as-tested" configuration, it seems possible that the neutron shield's imposed reduction in the annulus could help create a chokepoint and a consequential steam backpressure which could result in a significant reduction in the effectiveness of the ERVC strategy. Please discuss how this neutron shield emplacement comports with the above described optimal annulus design such that we have reasonable confidence that the ERVC system will function as well as the ULPU-2400 Configuration V tests have indicated.

Note: It is recognized that, even at severe accident temperatures, the above report estimates that the annulus in the vicinity of the neutron shield will be wide enough to provide the minimum 12 ft^2 flow area, but the as-tested configuration did not take into account the impact of such a stepwise reduction in flow area.

Westinghouse Response:

The ULPU Configuration V test includes a step-wise constriction at the top of the annulus before the steam-water mixture exits the annulus. This step-wise constriction is called a nozzle in the test report (Reference) and is shown on page 10 of the report. The nozzle is designed to simulate a reduction from the flow area in the 6 inch annulus between the reactor vessel and the reactor vessel insulation to a 12 ft^2 flow area. This is the configuration provided by the Reactor Vessel Insulation System design.

Reference: CRSS-03/06, Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility, University of California, T-N Dinh, et al, June 30, 2003

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-TR24-SPLA -02
Revision: 0

Question: There appear to be some other design modifications between what is described in Westinghouse Technical Report Number 24 (TR #24) and either AP1000 Design Control Document (DCD) Revision 15 or in report CRSS-03/06 "Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility", dated June 30, 2003.

a) For example, report CRSS-03/06 "Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility" indicates that the inlet baffle is "plugged" *during normal operation using floating balls. (The report does not state whether* or not such balls were used during the test to check their ability to open the inlet baffle when needed.) However, the proposed design of the Reactor Vessel Insulation System (RVIS) in TR #24 (with elaboration in report APP-MN20-Z0R-001 R0) has been modified to use inlet assemblies with buoyant doors, instead of floating balls.

b) In addition, AP1000 DCD Revision 15, page 5.3-20 indicates that "near the top of the lower insulation segment are four steam vent ducts that provide a flow path for the steam/water within the reactor vessel insulation annular space to flow back to the containment flood-up region..." which will "provide 12 ft^2 minimum flow area for steam/water to exit the annular space...", and that "each of the steam vents is covered with a cap that will be dislodged by the steam/water flow generated under the insulation with the cavity filled with water...". Attachment A to TR #24 now indicates that "Multiple steam vents in the nozzle gallery provide a flow path for the steam/water..." which will "provide 12 ft^2 minimum flow area for steam/water to exit the annular space..." Also, report APP-MN20-Z0R-001 R0, on pages 21-22, suggests that these "multiple" steam exits are covered by "hinged doors", not caps.

Please confirm that the impacts of these design modifications are bounded by the configuration(s) tested in the ULPU-2400 Configuration V Facility, and explain why this is so.

Westinghouse Response:

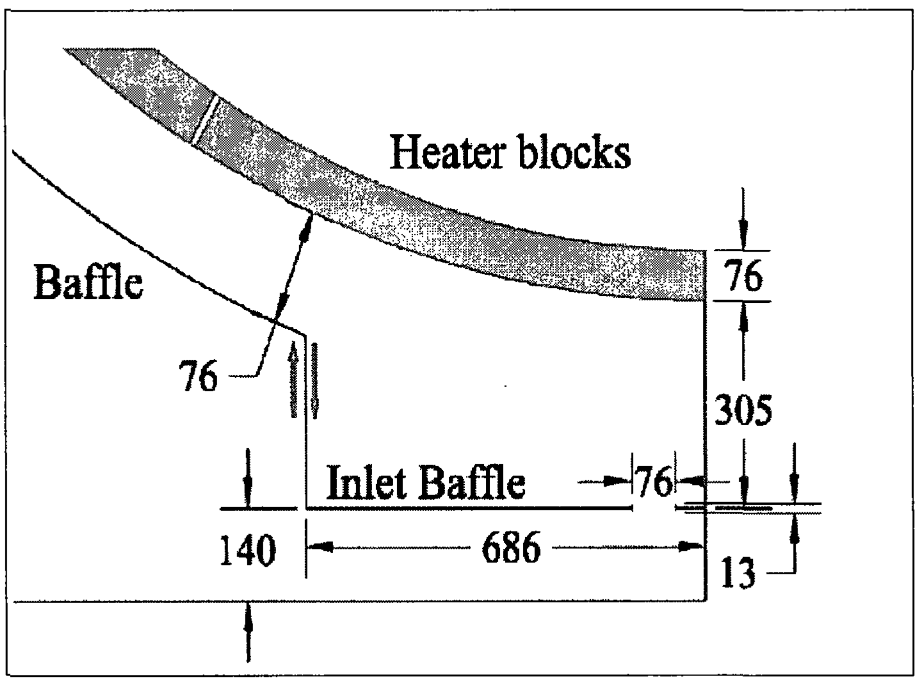
TR 24 is based on the current Reactor Vessel Insulation System (RVIS) design that uses inlet assemblies with doors instead of balls and multiple steam vents instead of ducts. The current design is described in Attachment A of TR24 (Markup of DCD Revision 15), Westinghouse design report APP-MN20-Z0R-001 (referenced in TR 24) and in Revision 16 of the DCD.

The AP1000 RVIS design is consistent with the ULPU tests. The RVIS simulates the test parameters that are important for providing results consistent with those of the tests. In addition the RVIS is designed to requirements (e.g. pressure variations) identified by the tests. Westinghouse incorporated two design enhancements that reduce the flow resistance in the In-Vessel Retention (IVR) water and steam flow paths. Reduced flow resistance increases the mass flow rate and enhances the IVR performance observed in the ULPU Configuration V test.

As discussed above, the two key design enhancements from the original concept are in the inlet assembly and the steam vents. The flow annulus on the bottom head and sidewall of the reactor vessel is the same for the ULPU Configuration V test, for the original concept up to the steam vents, and for the current RVIS design up to the neutron shield region. The following provides additional information on the other regions.

a) Inlet Assembly

Since simulating the buoyant balls passively floating open was not one of the purposes of the ULPU testing, the balls were not included in the test. Rather the test assumed that the balls would always open and water would freely flow into the AP1000 inlet assembly. The passive actuation feature of buoyancy initiating the IVR flow is retained with the current four door design.



From Figure 2.4 in ULPU Configuration V Test Report

The figure above is from the ULPU Configuration V test report and shows the test configuration that simulates the inlet assembly. In the test, water entered the inlet assembly through a 76 mm hole in the inlet baffle. The hole was sized to provide a flow area that matched the test scale. The test provides representative IVR results for any inlet assembly design that does not impose a greater flow resistance than the test. A lower flow resistance than the test improves mass flow which enhances in-vessel retention (IVR) performance over that predicted by the test.

The AP1000 design provides a flow area through four doors which exceeds the minimum required area by 33-1/3%. In addition the door design provides a lower flow resistance than modeled in the ULPU Configuration V test.

At the top of the inlet assembly, the bottom head insulation has a circular opening to allow water from the inlet assembly to flow into the annulus between the reactor vessel and the RVIS. The radius of the opening in the insulation panels (686 mm shown in the above figure) is the same for the ULPU Configuration V test and for the AP1000 RVIS design.

The lower flow resistance of the AP1000 inlet assembly design enhances the IVR performance observed in the ULPU Configuration V testing.

c) Steam Vents

Calculations show that the flow resistance of the current design is lower than that of the previous duct concept or of the ULPU Configuration V test configuration. This is reasonable. The ULPU Configuration V test and the previous duct concept both had multiple 90^0 mitre bends. The resistance coefficient of a mitre bend is generally larger than the resistance coefficient of a sudden contraction. In these configurations, the resistance coefficient of one mitre bend is roughly twice as large as that of the sudden contraction.

The current steam vent design therefore promotes increased mass flow and provides enhanced IVR performance over that shown to be acceptable in the ULPU Configuration V testing.

Summary

With a lower flow resistance for the inlet assembly and steam vent designs, and with the remainder of the RVIS having the same configuration as tested in ULPU Configuration V, the current RVIS provides enhanced IVR performance compared with the ULPU Configuration V test and the previous duct concept.

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-TR24-SPLA-03
Revision: 0

Question: The RVIS inlet and steam vent doors (as well as the Reactor Coolant Drain Tank Room Ventilation Damper) are, per Commission Policy as explained in SECY-94-084, approved in a Staff Requirements Memorandum dated June 30, 1994, considered active components. In addition, these devices are listed in the previously approved revision to the AP1000 DCD as Risk-Significant structures, systems, and components (SSCs) within the scope of the Design Reliability Assurance Program (D-RAP).

Section 5.3.5.2 of the current revision of the AP1000 DCD indicates that "Periodic verification of the vessel insulation moving parts can be performed during refueling outages." Also, Section 16.3.1 of the AP1000 DCD discusses a set of Investment Protection Short-Term Availability Controls, explaining that the rationale for these controls derive, in part, from PRA insights that identify SSCs that are important in protecting the utilities investment and for preventing and mitigating severe accidents.

a) Please confirm that these devices remain in the D-RAP of the Westinghouse-controlled version of the AP1000 DCD.

b) Given that "Periodic verification of the vessel insulation moving parts can be performed during refueling outages", and that "SSCs that are important in protecting the utilities investment and for preventing and mitigating severe accidents" should be subject to the above-noted availability controls, please propose such controls for the RVIS flowpath doors and dampers, including operability definition(s), action statements, and specific surveillance requirements, or otherwise justify why such controls are not needed. It is suggested that appropriate markups to Chapters 5 and 16 of the AP1000 DCD be included into Attachment A of TR #24.

Westinghouse Response:

a) The Reactor Vessel Insulation System (RVIS) Water Inlet and Steam Vent Devices are listed as being within the scope of D-RAP in Revision 16 of the AP1000 DCD.

b) The RVIS water inlet assembly and steam vents are simple, passive components. They are made from radiation resistant materials and open passively when flood water rises in the reactor cavity. The moving parts are hinged doors.

Proper fit and freedom of motion of the doors are confirmed at cold, severe accident, and operating temperatures during hot functional testing of the plant. The plant therefore begins operations with doors that were confirmed to swing freely at the relevant temperature conditions. Periodic testing need only confirm that the door hinges rotate freely and that the doors and surrounding structure have not been visibly damaged. Performing this test at refueling temperatures is most appropriate.

There are no identified plant scenarios that would damage the doors or cause them to fail. The doors are passive components which function during accidents beyond the design basis. The RVIS is not a safety system. For all of these reasons, it is reasonable for the doors to be tested every ten years.

Testing of the inlet assembly doors consists of a visual inspection for signs of physical damage that might preclude the doors from swinging freely and confirming that no more than 45 lbs of force in the vertical direction applied at the approximate center of each door is required to move a closed door to the vertical position.

Testing of the steam vent doors consists of a visual inspection for signs of physical damage that might preclude the doors from swinging freely and confirming that no more than 15 lbs of force perpendicular to the outer door surface at the approximate center of each door is required to move a closed door to the open position.

Periodic testing is performed during refueling, concurrently with other operations in these areas, such as testing of the loose parts sensors or replacement of the ex-core detectors.

This information on the testing requirements is to be added to TR24.

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-TR24-SPLA-04
Revision: 0

Question:

TR #24 lists a "DCD commitment" that the minimum flow area around the reactor vessel shall be 12 ft^2 with an external pressure of 12.95 feet of water on the RVIS. It also states a "DCD commitment" that there shall be a minimum gap of 2 inches between the RVIS and the reactor vessel sidewall due to dynamic loads during a severe accident. (The specification of this (minimum 2 inch) gap at the "reactor vessel sidewall" is interpreted to mean the cylindrical portion of the RVIS, not the hemispherical portion. These stated commitments in TR #24 raise a couple of comments/questions:

a) AP1000 DCD Revision 15 does not appear to discuss the "12.95 feet of water on the RVIS" commitment. Please include a discussion about this in Attachment A of TR #24. This discussion should elaborate on the conditions within the RPV cavity when this differential pressure of 12.95 feet of water across the RVIS is to be expected and the basis for this value, including any connection to the ULPU-2400 Configuration V tests. Also, due to the varying pressure exerted by water with depth, please identify specifically where this differential pressure will be exerted on the RVIS.

b) As stated, it appears that the minimum flow area around the reactor vessel of 12 ft^2 is required everywhere, when a simple calculation near the bottom of the hemispherical section, assuming an ~3 inch annulus, clearly shows a flow area less than 12 ft^2. (And, in fact, the inlet flow area requirement is only 6 ft^2.) Please specify those portions of the RVIS that must provide a minimum flow area around the reactor vessel of 12 ft^2 and the rationale for the flow area requirement(s), and revise Attachment A of TR #24 to reflect this.

c) It is not clear how the minimum gap of 2 inches at the reactor vessel sidewall during dynamic loading during a severe accident comports with the requirement to have a minimum flow area around the reactor vessel of 12 ft^2, particularly when the optimal configuration per the ULPU-2400 Configuration V test program calls for a roughly 6 inch annular gap in the cylindrical portion of the RVIS. Please provide the rationale for allowing such a significant reduction in this gap during these loading periods, and revise Attachment A of TR #24 to reflect this.

Note: If this information is specified in other references, it is sufficient to cite these references in Attachment A of TR #24. (e.g. - the first listed reference in the next RAI may, in fact, contain such information.)

Westinghouse Response:

It is correct to assume that the 2 inch gap refers to the cylindrical sidewall of the reactor vessel but not to the hemispherical bottom head.

a) Westinghouse response RAI RAI-TR24-EMB2-05 identifies the source of the 12.95 value and where it appears in the DCD. This value is conservatively higher than the calculated value at the worst location on the outside of the insulation panels. As shown in Reference 1 (Section 7.0 and Appendix D Section 4.1.5), this pressure was conservatively assumed to apply to the entire outside surface of the RVIS between the inlet assembly and the steam vents.

b) As shown in Reference 1 (Sections 4.2.2.1, 4.2.3.3, 4.2.3.4, Appendix D Section 4.1.5 and 4.1.8 and Appendix D Attachment 1 Sections 4.1.5 and 4.1.8), 12 ft^2 is the minimum flow area for venting the steam-water mixture above the hemispherical bottom head of the reactor vessel.

The 12 ft^2 area for the flow path that vents steam is shown in DCD Tier 1 Table 2.2.3-4 9.a) ii and Tier 2 Section 5.3.5.1, previously approved by the NRC. It was scaled up from the AP600 design and modeled in the ULPU Configuration V testing.

Along the hemispherical bottom head, ULPU Configuration V testing showed that the RVIS should provide a curved annulus that varied from nominally 3 inches at the bottom to nominally 6 inches at the top. This configuration improved heat removal for enhanced in-vessel retention (IVR) performance. This was implemented in the RVIS design as shown in many places in Reference 1, including Sections 4.2, 4.2.3.1, and Appendix D.

The testing also showed that a smooth transition between the bottom head and the sidewall annuluses improved IVR performance. The RVIS therefore conservatively continued the 6 inch annulus up the sidewall of the reactor vessel, as shown in Reference 1.

Reference 1 is identified in TR 24 as a reference.

c) Please refer to the Westinghouse response RAI-TR24-EMB2-03. Reference 1 shows that the design requirement was conservatively met by never allowing the steam-water vent path go below a 3.12 inches annulus width in any location at any condition. This ensures the path will always have a minimum 12 ft^2 flow area.

Reference:

1. APP-MN20-Z0R-001, Reactor Vessel Insulation System Design Report

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-TR24-SPLA-05
Revision: 0

Question:

TR24-SPLA-5 Please make the following references available to the staff for review:

a) APP-RXS-M3C-030 - "AP1000 RVIS Functional Requirements" (apparently not part of report APP-MN20-Z0R-001 R0)
b) Transco Dwg RV-49558-1 "RVI Key Elevation View"
c) Transco Dwg RV-49558-2 "RVI Key Plan View"
d) Transco Dwg RV-49558-3 "RVI Shell Insulation Details 1"
e) Transco Dwg RV-49558-3 "RVI Shell Insulation Details 2"

Note: Items b-e are in report APP-MN20-Z0R-001 R0, but they were apparently reduced onto 8.5 X 11" sheets of paper, and are not very legible (particularly the dimension data).

Westinghouse Response:

Document "a" is referenced in report APP-MN20-Z0R-001. A copy of APP-RXS-M3C-030, "AP1000 Reactor Vessel Insulation System Functional Requirements" and a larger copy of each of the requested drawings will be available in our Rockville office for your review.

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-TR24-SPLA-06
Revision: 0

Question:

Page 19.39-6 of AP1000 DCD Revision 15 indicates that "Testing of the ULPU-2000 Configuration III concluded that the aged painted surface did not inhibit the wetability of the surface of the lower head". There are also other statements in that chapter of the AP1000 DCD Revision 15 supportive of leaving the paint applied to the RPV external surface prior to shipment alone.

However, page 35 of report CRSS-03/06 "Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility" states, "Also, in preliminary BETA tests (not reported here) we found that typical paints and coating used to protect the reactor vessel during shipping (and normally left on it during operation) can have detrimental effects on CHF performance. Thus we recommended that in AP1000 such a protective layer is strippable, and that it be removed during installation, leaving bare steel as the surface of interest under IVR." Also, on page 41, one of the overall conclusions reaffirms this, stating a recommendation that a peelable protective coating be used in shipment (of the RPV), and that it be removed on installation.

How were these apparently contradictory conclusions about the paint on the external surface of the RPV reconciled, and what was the rationale? Please indicate the preferred configuration and provide supporting rationale to Attachment A of TR #24 as markups for Chapter 19 of the AP1000 DCD, making the appropriate references to report CRSS-03/06, as necessary.

Westinghouse Response:

Westinghouse has processed a design change that removed the paint from the reactor vessel. The changes to the DCD are in Sections 5.3.4.5, 19.34.2.1, 19.39.10.3 and Appendix 19B.

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-TR24-SPLA-07
Revision: 0

Question:

In amplification of RAI TR24-SPLA-2, please address the following potential concerns:

a) Diagrams in report APP-MN20-Z0R-001 R0 suggest that the inlet assemblies are not at the very bottom of the RVIS structure, but appear to be located some number of degrees of arc away from the very bottom, side-by-side, circumferentially around the structure. The buoyant doors appear to be hinged "on top", such that the incoming water is expected to force them open in an upward swinging motion.

   It is understood that the severe accident guidance has been revised to, in principle, ensure that the annulus between the RVIS and the Reactor Pressure Vessel (RPV) is flooded prior to the expected point of core relocation to the lower head. Nonetheless the surface of the RPV will still be "quite hot", and most likely well above the boiling point of water for postulated containment atmospheric conditions (particularly if the accident sequence doesn't result in that much generation of steam in the containment prior to initiating External Reactor Vessel Cooling (ERVC)).

   When the water that enters the annulus makes initial contact with the RPV, it is conceivable that there could be a significant amount of flashing to steam which could locally pressurize this region of the annulus. Because of the way the doors are hinged to "swing upward", it appears that the local pressurization could be felt "behind the doors", and possibly impose a closing force on them, which may or may not be greater than the buoyant force trying to open them. The net force calculations in APP-MN20-Z0R-001 R0 did not appear to consider this potential "dynamic backpressure".

   Please provide analytical and/or experimental evidence that the inlet doors, as currently configured will still open sufficiently to provide the required flow area cross-section as specified in APP-GW-GLR-060, given the above potential dynamics.

b) In addition, with the water initially entering some number of degrees of arc away from the very bottom, it is conceivable that in the course of this process, the motion of the flows could turn into eddy currents that could conceivably have an impact on the doors' ability to remain sufficiently open. Please provide analytical and/or experimental evidence that such phenomena will not occur to the point of impairing the ability of the doors' buoyancy to sufficiently open and remain sufficiently open during the ERVC process.

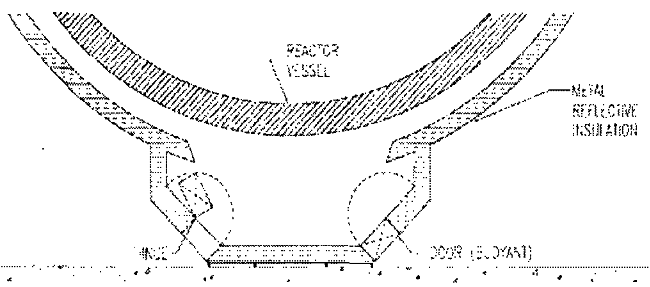
c) Please provide evidence (geometric and/or otherwise) that the thickness of the inlet doors will not encounter physical interference from the adjoining RVIS structure during the opening motion. Please take the impact of environmental temperatures (with the potential for differential thermal expansion) and radiation fields during prior normal operation into account in your explanation.

d) Please explain how surveillance testing during refueling outages will be able to simulate post-accident conditions sufficiently to provide assurance that doors will be free to open under such conditions.

Westinghouse Response:

a) As shown in Reference 1, the water inlet doors are located on the lower part of four sides of a hopper-shaped inlet assembly. This inlet assembly is located below an opening in the center bottom of the insulation on the bottom head of the reactor vessel. The bottom of each door is inches above the floor of the reactor cavity.

The hopper has flat sloping sides and each side has one door. The doors are spaced so that no door impedes the action of another. The doors are buoyant and therefore float open with water rising in the reactor cavity during an accident. Water pressure or momentum from inward flowing water is not required to open the doors. The following figure from Reference 1 provides a pictorial.



For clarity, this pictorial only shows two of the four doors. For reference purposes, the door on the right is shown closed and the door on the left is shown in its open position due to buoyancy. All doors would normally be in the same degree of openness.

The center of gravity of the open door is at the intersection of the dashed diagonal lines connecting opposite corners of the door. The center of buoyancy of the doors is in the same location as their center of gravity. In the open position due to buoyancy alone, the center of gravity of each door is over its hinge. The door reaches this position well before the water level reaches the reactor vessel bottom head.

The success criteria for in-vessel retention (IVR) requires that the reactor coolant system (RCS) be depressurized by a minimum of two ADS stage 4 valves in accident sequences that credit successful vessel integrity in a severe accident. Therefore, the maximum water temperature in the reactor vessel lower plenum prior to cavity flooding is the saturation temperature of water at the containment pressure. The vessel wall is cooled from the inside of the vessel prior to being externally cooled by the reactor cavity water. Also, due to the depressurization of the RCS, the accident sequence must result in a significant generation of steam to the containment. Therefore, the containment pressure will be elevated.

The water entering the insulation will be sub cooled by approximately 50°C (90°F) since it is draining from the cold IRWST which has a maximum initial temperature of 49°C (120°F). The maximum boiling rate at the reactor vessel surface is limited to the critical heat flux for pool boiling conditions. Low water level tests performed in the ULPU Configuration IV tests show that the critical heat flux correlation developed for the AP600 is applicable for pool boiling conditions. See page 9 of the ULPU Configuration IV Test Report (Reference 2).

As there is no molten debris in the lower plenum at the time of flooding, there is no heat load from the vessel wall other than the remaining sensible heat in the vessel metal. There will be no "flashing." The ULPU tests demonstrate that the flow of steam generated at the critical heat flux will be vented along the surface of the vessel lower head to the annulus between the vessel and the insulation. Therefore, there is no "dynamic backpressure" to close the inlet doors.

b) It is unclear how eddys in the inlet assembly would be large enough to close the doors or to partially close them for any length of time. As shown in Reference 1, the flow rate through all four doors is 15,000 gallons per minute and the door buoyancy force is greater than 50 lbs. The momentum of this water and the door buoyancy combine to keep the doors open.

Slight to moderate variations in door position have no impact on the doors being able to meet the design requirement of 6 ft^2 minimum flow area. Four open doors provide 8 ft^2 or 33-1/3% more area than this minimum design requirement. One door could close completely and the minimum flow area would still be provided. All four doors could be permanently held closed 46^0 and the minimum flow area would still be provided.

Eddys might cause minor and temporary fluctuations in door position, but would not close the doors far enough and long enough to have a negative impact on IVR performance.

c) Clearance was provided during the design process for the Reactor Vessel Insulation System (RVIS). The design process allowed for thermal expansion and two dimensional drawings were prepared to develop the concepts. A three dimensional model of the RVIS was created and all clearances and motions were checked.

A test is planned at the RVIS vendor to confirm proper motion of the inlet assembly doors at cold, severe accident, and operating temperatures of the reactor vessel to confirm the design. During hot functional testing of the AP1000 the free motion of the installed doors will be confirmed at these temperatures to confirm proper installation.

The RVIS is constructed of Type 304 stainless steel and the inlet assembly doors contain molded perlite insulation. Neither of these materials is expected to show any radiation effects at design exposure levels that would inhibit proper functioning of the RVIS. Perlite is an inorganic rock and such materials typically have a high radiation resistance. The radiation resistance of Type 304 stainless steel is well documented.

d) Testing is planned at the RVIS vendor to confirm proper motion of the inlet assembly doors at cold, severe accident, and operating temperatures of the reactor vessel. During hot functional testing of the AP1000 the free motion of the installed doors will be confirmed at these temperatures to show that in the installed condition the doors operate properly at all temperatures. Thereafter, testing can be limited to confirming hinge free motion and that the doors and surrounding structure are visibly undamaged.

Reference:

1. APP-MN20-Z0R-001, Reactor Vessel Insulation System Design Report
2. CRSS 02/05-1, Quantification of Limit to Coolability in ULPU-2000 Configuration IV

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-TR24-SPLA-08
Revision: 0

Question:

In amplification of RAI TR24-SPLA-3b, please address the following potential concern:

> In your proposal of surveillance requirements, please give consideration to the environmental conditions (expected radiation fields during refueling outages, etc) that personnel may have to deal with in order to exercise the doors.

Westinghouse Response:

RAI-TR24-SPLA-03 identifies the testing requirements for the steam vent and inlet assembly doors. The pre-start up testing, material selections, design simplicity, design basis accident scenarios, and the fact that this is not a safety system reduce the number, frequency and type of tests to be performed periodically over the life of the plant. Periodic testing is performed during refueling, concurrently with other operations in these areas, such as testing of the loose parts sensors or replacement of the ex-core detectors. In addition, the design and accessibility of the steam vents and inlet assembly in their respective locations minimize the amount of time required to perform the tests.

Individual doors and frames are designed to be removed as a unit from the inlet assembly. While not expected to be required, if a door needs to be replaced, this design allows the replacement to be done quickly and eliminates door-frame fit up in the reactor cavity environment. The steam vent doors are similarly replaceable in the nozzle gallery.

Design Control Document (DCD) Revision:
None

PRA Revision:
None

Technical Report (TR) Revision:
None