Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

Direct tel: 412-374-6206
Direct fax: 412-374-5005
e-mail: sisk1rb@westinghouse.com

Your ref: Docket No. 52-006
Our ref: DCP/NRC2287

November 11, 2008

Subject: AP1000 Responses to Requests for Additional Information (SRP6.2.2)

Westinghouse is submitting responses to the NRC requests for additional information (RAIs) on SRP Section 6.2.2. 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.

Responses are provided for RAI-SRP6.2.2-CIB1-01,-02,-04,-07,-12,-13, and -16, RAI-SRP6.2.2-SPCV-09, and RAI-SRP6.2.2-SRSB-02,-03,-05,-08,-09,-10,-11,-13, and -14 as sent in an email from Billy Gleaves to Sam Adams dated August 14, 2008. Responses for RAI-SRP6.2.2-CIB1-03,-05,-06,-08 thru -11,-14, and -17 thru -19, RAI-SRP6.2.2-SPCV-06,-07,-08,-10, and -11, and RAI-SRP6.2.2-SRSB-01,-04,-06,-07,-12, and -15 were submitted under letter DCP/NRC2285 dated November 6, 2008. Responses for RAI-SRP6.2.2-SPCV-01,-02,-04, and -05 were submitted under letter DCP/NRC2209 dated July 18, 2008.

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
Regulatory Affairs and Standardization

/Enclosure

1. Response to Request for Additional Information on SRP Section 6.2.2

D063
NRO

cc:	D. Jaffe	-	U.S. NRC	1E
	E. McKenna	-	U.S. NRC	1E
	B. Gleaves	-	U.S. NRC	1E
	P. Ray	-	TVA	1E
	P. Hastings	-	Duke Power	1E
	R. Kitchen	-	Progress Energy	1E
	A. Monroe	-	SCANA	1E
	J. Wilkinson	-	Florida Power & Light	1E
	C. Pierce	-	Southern Company	1E
	E. Schmiech	-	Westinghouse	1E
	G. Zinke	-	NuStart/Entergy	1E
	R. Grumbir	-	NuStart	1E
	T. Schulz	-	Westinghouse	1E

ENCLOSURE 1

Response to Request for Additional Information on SRP Section 6.2.2

RAI Response Number: RAI-SRP6.2.2-CIB1-01
Revision: 0

Question:

The use of non-safety injection systems is not described in any of the applicant submittals. Provide an evaluation of the effects of using non-safety active systems for removing core and containment heat while mitigating a LOCA, including:

a. Address the effects of possible additional amounts of debris ingested as a result of use of the active systems.
b. What impact does debris transported by non-safety systems have on recirculation performance and debris accumulation on recirculation and IRWST Screens and the core? Explain how you come to these conclusions.
c. Address how ingested debris could affect the capability of these active systems for long-term cooling. When these systems are used early in the event and encounter blockage or wear as a result, their capability to function adequately for long-term cooling (post-72 hours) could be impaired.
d. Address how ingested debris could affect the pressure integrity, leakage, and containment isolation function of these active systems. When these systems are used early in the event with higher levels of debris laden water, some components, such as pump seals, may encounter excessive wear and fail. The resulting leakage outside containment would then require the isolation and loss of the active system function, and isolation valves will be required to close and not leak excessively under the higher debris laden conditions.
e. Address whether leakage through pump seals or other components could increase local dose rates so that credited operator actions, if any, would not be met.

Westinghouse Response:

a. The RNS system can provide containment recirculation flows as high as 2400 gpm. Note that this flow is the combination of flow that comes through the containment recirculation screens and what comes from the IRWST (steam condensate return via the IRWST gutter). For comparison purposes, the PXS passive flows rates can be as high as 1130 gpm.

   The operation of the RNS system would not affect the evaluations performed for downstream effects. The evaluation of the effect of debris on the operating components of the RNS was performed with the conservative assumption that all latent containment debris could be ingested. Similarly, the evaluation of chemical deposition on fuel cladding was also performed with the conservative assumption that all latent containment fiber was available for deposition on fuel cladding. Therefore, from a downstream effects consideration, both equipment and fuel, have already conservatively accounted for debris that might be ingested through the recirculation screens by the operation of the RNS system. These evaluations have demonstrated that the AP1000 will perform successfully for at least 30 days following a postulated large-break LOCA.

b. The operation of non-safety systems would have no impact on the recirculation performance and debris accumulation on the recirculation and IRWST screens. The nominal flow rates for the flume recirculation flow rate were 50 gpm. This is higher than the scaled flow rate of 17.9 gpm at the face of the recirculation screen, which is scaled from an approach velocity of 4.3 gpm/ft^2 (0.01 ft/sec) for the AP1000. The higher flow rate of the test bounds the flow rate that would be generated by the operation of the non-safety RNS system. The debris used in the test bounds the amount that is evaluated to be transported to the recirculation screen, regardless of the operation of PXS or RNS. Thus, the screen performance test data is applicable for operation of either the PXS or the RNS.

   As noted above, the evaluation of the effect of debris on the operating components of the RNS was performed with the conservative assumption that all latent containment debris could be ingested. Similarly, the evaluation of chemical deposition on fuel cladding was also performed with the conservative assumption that all latent containment fiber was available for deposition on fuel cladding. Therefore, from a downstream effects consideration, both equipment and fuel, have already conservatively accounted for debris that might be ingested through the recirculation screens by the operation of the RNS system. These evaluations have demonstrated that the AP1000 will perform successfully for at least 30 days following a postulated large-break LOCA.

c. The RNS has been evaluated for wear, abrasion and erosion using the evaluation methods of WCAP-16406-P-A, Revision 1. This evaluation was has determined that, if RNS operation were initiated at the time of recirculation from the sump, the system would successfully perform its function for a 30 day period. This evaluation is documented in a calculation note that can be made available for review by NRC personnel at the Westinghouse Twinbrook Office.

d. The RNS has been evaluated for wear, abrasion and erosion using the evaluation methods of WCAP-16406-P-A, Revision 1. This evaluation was has determined that, if RNS operation were initiated at the time of recirculation from the sump, the system, including pump seals, would successfully perform their function for a 30 day period. This evaluation included the actuation of the RNS containment isolation valves in case it was desired to terminate RNS operation. This evaluation is documented in a calculation note that can be made available for review by NRC personnel.

e. Note that the RNS is automatically isolated from the containment following a LOCA if there is a large source term (like the design basis core melt source term). Also, as noted previously, the RNS has been evaluated for wear, abrasion and erosion using the evaluation methods of WCAP-16406-P-A, Revision 1. This evaluation has determined that, if RNS operation were initiated at the time of recirculation from the sump, the system, including pump seals, would successfully perform their function for a 30 day period.

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-SRP6.2.2-CIB1-02
Revision: 0

Question:

Please clarify why transport of coating debris for the AP1000 was assumed to be limited to the resident debris. TR 26, Revision 3, page 11 states the only debris that can be transported to the AP1000 screens following a LOCA is resident debris. For operating reactors a Zone of Influence (ZOI) must be identified, and coatings both inside the ZOI are assumed to fail as particles. In addition, all unqualified coatings both inside and outside the ZOI are assumed to fail.

Westinghouse Response:

The NRC has reviewed the AP1000 use of high density coatings inside containment and the effectiveness of the recirculation screen protective plates in preventing coating debris from being transported to the recirculation screens in DCD Rev. 15. The staff stated in the FSER for the AP1000 in section 6.2.1.8.3 on page 6-51 that the AP1000 design with respect to coating debris is acceptable.

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-SRP6.2.2-CIB1-04
Revision: 0

Question:

Please provide the basis for not including degraded-qualified coatings in the debris source term as recommended by the NRC safety evaluation of NEI 04-07. It is not clear to the staff how the amount of coating debris assumed in the AP1000 analysis relates to the condition of coatings in containment (e.g., degraded coatings identified during routine coating inspections).

Westinghouse Response:

The basis for the AP1000 design is that high density coatings are used on containment internal structures and debris from failure of these coatings will not be transported to the recirculation screens. The approach is discussed in the DCD in section 6.1.2 and the staff has reviewed and accepted this approach as discussed in the response to RAI-SRP 6.2.2-CIB1-02.

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-SRP6.2.2-CIB1-07
Revision: 0

Question:

Provide the results of the evaluation performed for plugging and wear of the PXS piping system with the assumed latent and coatings debris for the necessary mission time. What is the amount and composition of debris in the downstream flow path, other than latent debris and the assumed addition of coatings debris? Provide an evaluation of its effect on possible plugging and wear in the downstream flow path components.

Westinghouse Response:

The results of this evaluation are presented in Technical Report 26 ,APP-GW-GLR-079, Revision 3, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA", March 2008. As discussed in the response to RAI-SRP 6.2.2-CIB1-06, coating debris are prevented from being transported to the AP1000 recirculation screens. The calculation notes that were performed to support this TR are available for NRC 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-SRP6.2.2-CIB1-12
Revision: 0

Question:

Since it is expected that during PXS cooling, in addition to steam, there will be some carryover of liquid downstream of the reactor vessel, address the effects of debris entrained in the carryover liquid and boric acid and other chemicals in the liquid which could settle or precipitate in the flow path downstream of the reactor vessel (i.e., flow path from the vessel back to the break location.)

Westinghouse Response:

The discharge for the reactor vessel will be through the ADS stage 4 valves. These valves are large 14" valves with a straight through layout that has an ID that is greater than 9". These valves are not susceptible to plugging. The ADS 4 lines discharge directly into the loop compartments above the water level. The flow path back to the recirculation screens is through the loop compartment or through the corridor that connects the two compartments. The limiting flow path is the corridor which is at least 7.5 feet wide and as a result is not subject to plugging by debris. Note that the maximum boron concentration in the core is 7400 ppm for the AP1000 (refer to DCD Section 15.6.5.4C.4). This concentration occurs after several hours operation such that the containment water will also be hot and mixing containment water with this water will not cause any precipitation.

The evaluation of the dimensions of the flow paths from the ADS stage 4 valves to the recirculation screens given above is also applicable to other post-accident chemicals. In addition to large lines, as noted previously, calculations for other post-accident chemicals up stream of the core demonstrate their concentration is low throughout the 30-day period considered. Considering the low concentration of post-accident chemical products, the solubility limits of the chemical products is not challenged. Thus, there is no mechanism that would cause their precipitation out of solution in the flow path from the vessel back to the break location.

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-SRP6.2.2-CIB1-13
Revision: 0

Question:

Provide an evaluation of the effects of the possible collection of non-condensable gases in high points in the PXS flow path, including gases which may evolve out of solution in the piping system, gaseous chemicals, and gases which may form as a result of chemical reactions. Gases in sufficient quantities which collect and are trapped at high points could cause unacceptable pressure losses and restriction of system cooling flow, especially in a gravity-driven system.

Westinghouse Response:

The CMT and PRHR HX circuits are not susceptible to gas accumulation during pre-accident standby conditions except in an engineered high point pipe stub. Both of these circuits have inlet lines that are carefully routed from the top of RCS main loop piping up to a high point. Along the highest pipe elevation there is a vertical pipe stub that is provided with redundant level sensors that are able to measure if non-condensable gas is present. Gas could be present because of a lack of venting during shutdown operations or due to accumulation during power operations. Note that gas accumulation is not expected to occur during power operation; although hydrogen is dissolved in the RCS the concentration of hydrogen is low and very little pressure is required to maintain it in solution. In addition, the CMTs and the PRHR HX are maintained at RCS pressure with a single isolation barrier. This design approach eliminates the possibility of an isolated portion of the circuit depressurizing and allowing hydrogen to come out of solution. This design approach also prevents water hammer occurrences when the system is actuated.

The lines associated with the accumulator are expected to contain nitrogen dissolved in the water. When the accumulators inject they have a substantial DP available for injection such that there is no issue with gas pockets impeding flow. Once the accumulators empty, nitrogen from the tank will be injected into the RCS. This nitrogen is readily vented from the RCS through open ADS valves. Note that the integral testing performed for AP1000 showed that the plant and system performance were not adversely affected by Nitrogen dissolved in accumulator water or injected into the RCS from the accumulator after the water in the tank is injected.

The lines from the IRWST to the RCS are isolated by squib valves. On the IRWST side the head of water from the IRWST maintains a positive pressure in the lines and prevents the ingress of air into the lines. There are also no dissolved gases in the IRWST water other than atmospheric air. Since these lines are routed such that they remain below the bottom of the IRWST there is sufficient pressure to keep the air dissolved in the water in solution.

The lines from the containment to the IRWST injection lines are isolated by squib valves. On the IRWST side of the squib valves the situation is the same as the IRWST injection lines. On the containment side of the squib valves the lines will normally be filled with air. Each recirculation line from the containment to the squib valves is a short, straight horizontal pipe. As a result, during an accident where the containment floods, these lines will fill with water during the 2 ½ hours between the initial LOCA and the time when the recirculation lines are opened.

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-SRP6.2.2-CIB1-16
Revision: 0

Question:

TR 26, Revision 3, in the section called "Post-Accident Chemical Effects," page 14 of 26, there is a statement that the AP1000 containment "has little concrete that can come in contact with the post accident water as a result of the use of structural steel module construction." Please provide a detailed explanation of how the amount of calcium phosphate listed in Table 4 of TR 26 was determined based on the amount of concrete that can contact the post-accident water.

Westinghouse Response:

This evaluation was performed using, among the other required inputs, the concrete surface area exposed to recirculation pool liquid, the recirculation pool liquid temperature and the recirculation pool pH as inputs to the chemical effects spreadsheet calculation based on WCAP-16530-NP. This analysis is documented in a calculation note that can be made available for review by NRC personnel.

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-SRP6.2.2-SPCV-09
Revision: 0

Question:

Provide a description of how permanent plant changes inside containment are programmatically controlled so as to not change the analytical assumptions and numerical inputs of the licensee analyses supporting the conclusion that the reactor plant remains in compliance with 10 CFR 50.46 and related regulatory requirements.

Westinghouse Response:

The programmatic controls for permanent plant changes are the responsibility of the licensee. The programmatic control of these changes is not within the scope of the DCD. The program controlling permanent plant changes made by the licensee must comply with Appendix D to Part 52-- Design Certification Rule for the AP1000 Design, Section VIII, "Processes for Changes and Departures."

The programmatic controls for changes to the certified design fall under 10 CFR 52.63, Finality of standard design certifications. Any change to the certified design requires NRC review and approval prior to implementation.

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-SRP6.2.2-SRSB-02
Revision: 0

Question:

In TR 26, Revision 3, on page 8, in the "Applicability to the AP1000 Design" subsection, Westinghouse states that:

> "The flow velocities have been reduced further by the increase in face area of the screens (approximately 55% larger for containment recirculation)."

Since Westinghouse credits the low flow velocities in minimizing the potential for a LOCA to generate debris challenging the recirculation flow path, the staff requests additional information to assess the velocities during every phase of the transient.

Identify the calculated flow velocities through the IRWST screens (leading to the intact and broken DVI lines, respectively), the containment recirculation screens, and the reactor vessel during all phases of the transient including the reverse flow through the recirculation screens when the IRWST is switched to recirculation. For example, when recirculation flow starts, identify the velocity of the initial reverse flow through the recirculation screens and the velocity variation across the screens once all water has filled the recirculation sump cavity.

Westinghouse Response:

The sensitivity study performed to evaluate effect of increased head loss caused by post accident debris was based on the DCD Chapter 15.6.5.4C analysis. This DCD case is a double ended DVI LOCA that results in flooding of PXS room B and reduces the final containment floodup level. The IRWST injection line associated with the faulted DVI line is assumed to open early in the event when the CMT connected to the faulted DVI line blows down to the ADS 4 actuation setpoint. This assumption is made to maximize the IRWST draindown flow rate and reduce the time when switch to recirculation occurs. When recirculation is actuated on a low IRWST level signal, it is assumed that the recirculation squib valves located in the flooded PXS room fail because they are not qualified to operate under water, this assumption minimizes the number of recirculation lines in operation. The recirculation squib valves located in the unflooded PXS room provide for recirculation; the limiting single failure is one ADS 4 squib valve. Note that the operating recirculation line draws water from both recirculation screens because the screens are cross connected. In the final long-term cooling configuration, water recirculates from the containment through one recirculation line and from PCS condensate that is collected and drained into the IRWST.

Each of the two IRWST injection lines are separated from each other and are connected to opposite sides of the tank. As described above. IRWST flow into the intact DVI line is zero until the RCS pressure is significantly reduced. The flow through the IRWST line associated with the faulted DVI line starts as soon as the IRWST squibs open.

In the injection phase, the volumetric flow in the IRWST injection line associated with the intact DVI line reaches a maximum of 1260 gpm and decreases to the long term recirculation flow shown below as the IRWST drains. At this flow, the IRWST screen face velocity is 0.070 ft/sec.

The IRWST spill flow (volumetric) through the broken DVI line reaches a maximum as soon as the IRWST injection valves open and decreases as the IRWST drains and the PXS valve room floods. The maximum initial flow in this line is 2700 gpm. This flow decreases quickly to 2220 gpm as the PXS valve room floods due to RCS blowdown and IRWST spill. The face velocity in the IRWST screen at these flow rates is 0.150 and 0.124 ft/sec.

When the IRWST level has dropped to about 112 ft, the recirculation squib valves automatically open. At this time the IRWST level is about 6 ft above the containment water level which results in a brief period of flow from the IRWST into the containment. This flow rate will be less than 650 gpm and will flow through one recirculation line (associated with the intact DVI line) and through both recirculation screens. At this flow rate, the velocity through the front face of the recirculation screens is 0.0132 ft/sec.

During the long-term cooling analysis recirculation phase, flow from the IRWST maintains the water level in the PXS room such that it serves as a reservoir for the broken DVI line injection into the vessel. The IRWST level is maintained by steam condensate return from the IRWST gutter and back flow from the intact PXS recirculation line. The containment recirculation line connected to the intact DVI line provides flow to the reactor vessel through the intact DVI line as well as some flow back through the IRWST to the faulted DVI line.

The following table shows the calculated flow velocities through the IRWST screens, the containment recirculation screens, and the reactor vessel during the containment recirculation phase. Note that because of the cross connection between the two containment recirculation screens, both of these screens share the flow which reduces the velocity. The IRWST screen flows and velocities listed below are for the screen associated with the faulted DVI line; the other IRWST screen sees a lower flow rate. The reactor vessel velocity is calculated at the fuel assembly inlet.

	IRWST screen	Recirculation screen	Reactor Vessel
Time [sec]	6800 to 10,000	6800 to 10,000	6800 to 10,000
Flow Rate [gpm]	540	755	1042
Velocity [ft/s]	0.030	0.008	0.0313

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-SRP6.2.2-SRSB-03
Revision: 0

Question:

In TR 26, Revision 3, on page 8 in "Applicability to the AP1000 Design" subsection, Westinghouse states that metal reflective insulation (MRI) is used on components that may be subjected to jet impingement loads; MRI is not transported to the AP1000 Containment Recirculation Screens with low flow rates; and as a result, there is no fibrous debris generated by the LOCA blowdown. On page 11 in "Break Selection Criteria" subsection, Westinghouse states that the density of the MRI material ensures that any debris generated by the damage of this insulation material to settle in the containment sump and not be transported onto the screens.

The staff notes there is a significant amount of MRI in the AP1000. In the SER for NEI 04-07, on page 7, the staff stated that MRI is assumed to degrade to 75 percent small fines and 25 percent large pieces.

a. Describe testing or evaluations that show that this type of insulation, once it has been damaged by the LOCA jet, will not become debris that will cause potential plugging of the screens.
b. Verify that the same degradation for the MRI as described in the NEI 04-07 SER exists in the AP1000 or identify what the degradation would be. Describe the impact of the degradation on the debris loading.
c. Was an evaluation performed showing that the MRI material under AP1000 break conditions will not migrate to the containment sump and screens? If so, provide the reference or detailed information in the reference.
d. Is there any chemical residual associated with the MRI that could impact the screen blockage or the downstream blockage in the core? If so, what is the impact to the screens and to the core blockage?
e. Are there any other objects in the zone of influence that can be damaged by jet impingement and contribute to the debris (e.g., cable insulation, instrumentation, hot/cold leg temperature instrumentation, nuclear instrumentation, signs, caulking...)?
f. Is there any fiber insulation encased in MRI that could contribute to the debris? If so, are the configurations qualified for jet impingement? Provide the qualification details.
g. How will lack of debris generating materials in the zone of influence be verified?

Westinghouse Response:

a. Information regarding the generation of debris from MRI is given in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance." Information regarding the transport of MRI is given in NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" and has been accepted in the associated NRC Safety Evaluation on NEI 04-07. The liquid velocities evaluated for the AP1000 recirculation flows are lower that the values listed for MRI in Section 4.0 of NEI 04-07.

b. NEI 04-07 and its associated NRC Safety Evaluation are applicable to current operating PWR's. The thermal hydraulic conditions associated with a postulated break for the AP1000 are the same as those for current operating PWR's. Therefore, the degradation or damage characteristics for MRI described in NEI 04-07 and its associated Safety Evaluation for current operating plants are also applicable to the AP1000.

c. MRI is constructed of stainless steel. Tests reported in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," have been performed that demonstrate that MRI damaged by LOCA tests will settle and require more velocity to transport than will occur in AP1000. These tests indicate that a velocity of 0.2 ft/s is required to move ½" x ½" crumpled foil MRI debris; larger velocities are required to move larger MRI debris. The AP1000 will have a liquid velocity less than 0.072 ft/s available to move MRI toward the containment recirculation screens.

d. MRI is constructed of stainless steel and contains no substances that would contribute to the post accident chemical precipitants.

e. Damage due to jet impingement is dependant upon the material of interest. Based on a review of the AP1000 design, there are no other materials that that would be affected by jet impingement loads associated with a high energy pipe break that would become debris.

f. As stated in TR 26, on page 8, MRI contains no fibrous material.

g. Tier I ITAAC Table 2.2.3-4, item ix, addresses this issue. The ITAAC requires the inspection of the "insulation used inside the containment on ASME Class 1 lines and on the reactor vessel, reactor coolant pumps, pressurizer and steam generators". The acceptance criteria are that "the type of insulation used on these lines and equipment is a metal reflective type or a suitable equivalent". Also note that the DCD (Section 6.3.2.2.7.1, item 3) also requires metal reflective insulation or a suitable equivalent be used where LOCA jet impingement damage insulation and generate debris.

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-SRP6.2.2-SRSB-05
Revision: 0

Question:

In TR 26, Revision 3, on page 10, Westinghouse states that debris samples removed from operating plants and visual observations during plant walk downs provide the basis for the debris composition as particulate material (85% by volume), coatings (5% by volume) and fiber (10% by volume).

a. Provide the references or sources of the operating plant walkdowns and debris samples. Confirm that the percentages of various debris types are based on total volume, as reported, and not total mass (NRC SE of NEI 04-07 recommends that fiber be 15% of total mass).
b. Explain why these references and data are representative of the AP1000.
c. Describe whether there could be other types of latent materials in the AP1000 that are different from the operating plants.
d. Explain how actual as-designed and operated AP1000 will be verified to be consistent with this data both prior to start-up and during the life of the plant? Propose appropriate surveillance testing and/or programmatic controls that will be necessary to ensure the actual plant is operated consistent with the analysis assumptions

Westinghouse Response:

a. A debris composition of 85% particulates, 14% coatings and 1% fiber by mass is used in the calculation note that documents the AP1000 resident debris loadings. This data is based on walk downs performed at three plants. The walk down data is proprietary; the calculation note is available for review by the NRC. It is noted that rounding the fiber debris upward to 1% fiber by mass added conservatism to the assumed amount of fibrous debris considered in the calculations.

b. The three plants whose latent containment debris walkdown data was used to assess latent debris loading for the AP1000 were evaluated to have containment cleanliness programs that the AP1000 might serve as a model for the AP1000 containment cleanliness program. The AP1000 DCD section 6.3.8.1 requires that the AP1000 cleanliness program be consistent with cleanliness programs used in the evaluation of debris loadings for the AP1000. Given this requirement, the walkdown data from the three plants is applicable to the AP1000 design. It is the responsibility of the AP1000 licensee to define and implement the COL containment cleanliness program to assure that the actual latent containment debris loads are less than or equal to those determined by evaluation and used in testing the recirculation screens.

c. The total amount of latent debris is limited by the COL cleanliness program. So the only question is whether the types of debris that might be found in an AP1000 containment could be different such that the resulting head loss across AP1000 screens would be increased.

Some of the resident debris found in the walk downs was transported into the containment by people that enter the containment during shutdown activities. This source of debris is expected to be reduced because the simplifications in the AP1000 and the reduced maintenance resulting from the use of canned motor reactor coolant pumps will result in fewer people entering the containment during shutdowns.

The materials used in the AP1000 containment are similar or less likely to create resident debris than those used in the plants where the walk downs were performed. An example of an improved material is the use of MRI instead of fiberglass insulation; removing and re-installing fiberglass insulation can create small amounts of latent debris.

One area where AP1000 might have an increase in a specific type of resident debris is in the area of coatings. The AP1000 uses qualified coatings inside containment, however they are not applied, inspected and maintained as safety coatings. As a result, the latent debris found in an AP1000 could have a greater percentage of coatings debris. Since these coatings are required to be high density they could not be transported to the screens.

In summary, the AP1000 latent debris is expected to be made up of similar materials and if anything a mixture that would result in less pressure loss (less fiber and more (high density) coatings).

d. The AP1000 DCD contains a COL item (6.3.8.1) that requires the COLA to provide a cleanliness program that is consistent with the assumptions used in determining the AP1000 latent debris amounts. In addition, sensitivity studies have been performed that demonstrate that the AP1000 long-term cooling operation can tolerate much higher head loss than was shown to occur in the debris testing performed for the AP1000 screens and core.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

RAI Response Number: RAI-SRP6.2.2-SRSB-08
Revision: 0

Question:

In TR 26, Revision 3, in the "Post-Accident Chemical Effects" subsection on page 13, Westinghouse states:

> "An analysis was performed to determine the type and quantity of chemical precipitants which may form in the post-LOCA recirculation fluid for the AP1000 design. The analysis evaluated these post-LOCA chemical effects using the methodology developed in WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191" (Reference 7)."

The staff notes that the chemical composition of the AP1000 containment is significantly different from the PWR study (WCAP-16530-NP).

Provide the details of the calculation, including assumptions, which resulted in the predicted chemical precipitate formation in Table 4 of TR-26, Revision 3.

Westinghouse Response:

The details of the calculated values for chemical precipitate listed in Table 4 of TR 26 were calculated using the spreadsheet tool generated as part of WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," and are documented in a calculation note. If additional details are required the NRC may review this calculation.

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-SRP6.2.2-SRSB-09
Revision: 0

Question:

In the AP1000 DCD Revision 16, Section 6.3.2.2.7.2, Westinghouse states that:

> "During a LOCA, steam vented from the reactor coolant system condenses on the containment shell, drains down the shell to the operating deck elevation and is collected in a gutter. It is very unlikely that debris generated by a LOCA can reach the gutter because of its location. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4 inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is one potential source of debris that may enter the gutter and the IRWST. As described in subsection 6.1.2.1.5, failure of this coating produces a heavy powder which if it enters the IRWST through the gutter will settle out on the bottom of the IRWST because of its high specific gravity. Settling is enhanced in the IRWST by low velocities in the tank and long tank drain down times."

The staff notes that this collection gutter appears to be potential collection point for debris because runoff from the upper containment enters this gutter system and then flows into the top of the IRWST. The staff also notes that there are no screens in this system, only a trash rack, and a debris flow path could easily be formed from the entry point at the top of the IRWST to the bottom of the IRWST where the flow exits through the IRWST screens into the flow path to the reactor vessel. Westinghouse stated in DCD Section 6.3.2.2.7.2 that much of the debris that enters through the gutter system settles to the bottom of the IRWST. In Table 2 of TR 26, Revision 3, Westinghouse indicated that some debris was caught in the IRWST screens, 2.47 kg (5.44 lbm).

Justify the calculated IRWST screen debris mass and volumes for Table 2 in TR26, and discuss in the justification how the significant runoff from the containment into the IRWST during a LOCA was specifically factored into the calculation. Also discuss in your justification how this relates to the total residue mass in the containment that you assumed to be (i.e., 227 kg (500 lb)) as documented in NUREG-1793 Section 6.2.1.8.2.

Westinghouse Response:

As discussed in TR 26 and in the response to RAI-SRP-SRSB-06, latent debris loadings were developed from operating plant walk down data. Loadings were developed for different surfaces (floors, walls, equipment, ...) in terms of lb/ft^2. These loadings were then applied to the actual AP1000 surface areas. Two levels of loadings were developed; one was a "best estimate" and the other one was a "bounding" case. As explained in TR 26, the best estimate case was based on the average loadings from plant walk down data for similar surfaces and included a 25% margin to address uncertainties. The bounding case was based on the highest loading from plant walk down data for similar surfaces. Also note that the percentage of fiber assumed in the loadings was rounded up to 1%; this assumption increased the amount of fiber assumed in the AP1000 by a factor of four.

As discussed in TR 26 and in the response to RAI-SRP-SRSB-06, all of the latent debris that was assumed to exist on AP1000 surfaces is assumed to be transported if the surfaces were covered with water that would flow to one of the screens or to the core following a LOCA.

To demonstrate that there is not a "cliff" near the bounding case, a sensitivity case was defined that assumed that there was 200 pounds of latent debris in the containment. As discussed in TR 26, the 200 pounds was based on the NRC recommended value for latent debris in their safety evaluation of NEI 04-07. This sensitivity case resulted in the latent debris transported to AP1000 screens increasing by a about factor of 2. A value of 500 pounds (227kg) of latent debris was not assumed in the AP1000 long-term cooling evaluations.

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-SRP6.2.2-SRSB-10
Revision: 0

Question:

In TR 26, Revision 3, in the "Break Selection Criteria" subsection on page 13, Westinghouse states:

> "Note that the debris reaching the core is based on a DVI LOCA in the loop compartment. For this event the containment water level rises above the break so that some water can enter the reactor coolant system (RCS) directly and thereby bypass the Containment Recirculation Screens. It is calculated for such an event that no more than 60% of the total recirculation flow will bypass the screens. As a result, the core debris is set at 60% of the Containment Recirculation Screen amount."

The staff notes that in Table 3 the best estimate total mass bypass to the core is 14.35 lbm, and requests the following information.

a. Clarify, since the amount of bypass debris is significant for determining the effect on the core, the basis for the 60% number.
b. Describe how this relates to the total residue mass in the containment of 227 kg (500 lb) that you assumed, as documented in NUREG-1793, Section 6.2.1.8.2.
c. Clarify whether the total mass number to the core includes bypass debris from the recirculation and IRWST screens.

Westinghouse Response:

Page 13 of TR 26 discusses the basis for 60%. The basis is that for the limiting break location (a DVI break), the long-term cooling analysis shows that less than 60% of the integrated RCS flow will enter through the break. The other 40% will pass through the recirculation screens. As a result, the recirculation screens will collect 40% of the latent debris and the remaining 60% will enter the core through the break.

a. The AP1000 did not assume 500 lb of debris were inside the containment. Refer to the response to RAI 6.2.2-SRSB-07 for an explanation of how the AP1000 latent debris was calculated. The 60% value is the percentage of the recirculation screen debris that could be transported to into the core.
b. The AP1000 did not assume 500 lb of debris were inside the containment. Refer to the response to RAI SRP 6.2.2-SRSB-07 for an explanation of how the AP1000 latent debris was calculated.
c. The amount of bypass debris that might pass through the AP1000 screens will be very small considering the low debris loading of these screens and is bounded by the 60% value.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

RAI Response Number: RAI-SRP6.2.2-SRSB-11
Revision: 0

Question:

Provide responses to the following questions related to APP-PXS-GLR-001, Revision 0, "Impact on AP1000 Post-LCOA Long Term Cooling of Postulated Containment Sump Debris," issued April 28, 2008:

a. In the DEDVI break cases, it is noted that the containment water level exceeds the elevation of the break so that water can flow directly into the reactor pressure vessel bypassing the sump screens. For each of the cases analyzed, including the two sensitivity cases, provide the debris and chemical loading for the water bypassing the sump screens and that taken downstream of the sump screens.

b. Provide the hydraulic head of the IRWST, and the hydraulic head (i.e., water elevation in the containment) over the DVI break location and the recirculation screens with respect to time, the losses in the broken DVI line, and the core inlet resistance for each case analyzed, including the two sensitivity cases.

c. Provide plots of the integrated core boiloff rate and integrated core inlet flow rate for each of the cases analyzed, including the two sensitivity cases.

d. Figure 2-2 indicates the core collapsed level is decreasing. Explain why the level with the unblocked core inlet would decrease while those for the sensitivity cases, Figures 3.1-2 and 3.2-2, decrease for approximately 1500 seconds, then level off for the remainder of the transient. Also explain why the core collapsed liquid levels in the two sensitivity cases are generally higher than the base case.

e. Considering the differences in the core inlet flow rates between the base case and the sensitivity cases as shown by the intact and broken DVI line mixture flow rates (Figures 2-13, 2-14, 3.1-13, 3.1-14, 3.2-13, and 3.2-14), explain why the upper plenum collapsed liquid levels remain almost the same between the base and sensitivity cases (Figures 2-8, 3.1-8 and 3.2-8).

f. Discuss the local heatup effects due to capture of the debris and potential precipitates on fuel rods within the spacer grids and between the spacer grids. The discussion should also consider maximum pre-existing cladding oxide and crud. Justify the amount of oxide and crud assumed for the analysis.

g. On page 1 of APP-PXS-GLR-001, the staff notes that credit is taken for cooling the core from the bypass flow through the broken DVI line from the containment to the downcomer. In DCD Section 15.6.5.4B.3.1, on page 15.6-39, Westinghouse stated that a venturi was inline to limit the flow out the break is located in the DVI line. The bypass flow that flows to the core carries debris through this venturi.

Confirm that the plugging of this venturi has been factored into the cooling flow for the core. If not considered or factored in, please provide an evaluation.

h. In Section 1, "Introduction," Westinghouse provides five reasons or considerations for selecting the DCD long-term cooling case [DVI line break] as the base for the sensitivity study. The first bullet describes the amount of debris bypassing the containment recirculation screens and being transported to the core for cold leg and hot leg breaks. The second bullet describes a DEDVI break in a PXS room would make available only a small portion of the debris that would be available for a loop break. Explain how these two bullets justify the DEDVI break being the limiting break for long-term cooling sensitivity study. Explain why the DEDVI break chosen is the limiting case from a head-loss standpoint for the IRWST screens, recirculation screens and the core. Also explain when the analyses were begun and why debris would not be present prior to the analysis.

Westinghouse Response:

a. Technical Report 26 ,APP-GW-GLR-079, Revision 3, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA", March 2008 provides this information in Table 5 for the latent debris.

b. The hydraulic head of the IRWST, expressed as its liquid level elevation inside the containment during the IRWST injection phase, and the liquid level in containment during the recirculation phase are as follows in the DCD Revision 17 Chapter 15.6.5.4C DEDVI break analysis. The same values apply to the two containment debris sensitivity cases analyzed using the WCOBRA/TRAC AP1000 long-term cooling methodology.

IRWST hydraulic head, then containment hydraulic head		DEDVI LOCA, long-term cooling		
Transient Time (time after break occurs) (sec)	Analysis Time (WC/T time) (sec)	IRWST level: then Sump Level during Recirculation (ft)	Level relative to IRWST injection line location, @ 97.0 ft (ft)	
3000.00	0.00	125.96	28.96	
3000.00	500.00	125.96	28.96	
5232.53	2732.53	117.81	20.81	
6486.28	3986.28	113.79	16.79	
7390.81	4890.81	111.16	14.16	
7820.00	5320.00	110.00	13.00	
9098.65	6598.65	110.00	13.00	
9300.00	6800.00	110.00	13.00	Sump injection switchover time
9400.24	6900.24	106.86	9.86	For restart run
9450.47	6950.47	106.93	9.93	
9701.15	7201.15	107.80	10.80	
10654.80	8154.80	107.80	10.80	
11257.40	8757.40	107.80	10.80	
12666.80	10166.80	107.80	10.80	
14377.50	11877.50	107.80	10.80	

At the initiation of switchover, a reduced value of the level is assumed for recirculation to accommodate any dynamic effects from the draining the IRWST into the sump that might slightly affect the static head available for flow into the reactor vessel. The equilibrium containment floodup level of 107.80 ft. is established over the recirculation screens that feed the intact DVI line once 400 seconds have elapsed in the WCOBRA/TRAC restart problem, and this value is maintained thereafter. The 400 second time frame is the period between 'Sump injection switchover time (6800.00 sec)' and '7201.15 sec' in the analysis time (WC/T time column).

The hydraulic head of water in the PXS room with the broken DVI pipe, expressed as the liquid level elevation inside the containment, is identical to the above table values from 6900.24 seconds onward in WCOBRA/TRAC. During the IRWST injection phase of the DEDVI transient, the value is 107.1 ft. from WCOBRA/TRAC analysis time zero until 6598.65 seconds. A value of 106.61 ft. @ 6800 seconds is the sole intermediate input value between the 6598.65 and 6900.24 second points in the DCD Revision 16 Chapter 15.6.5.4C analysis and also in the sensitivity cases.

Consistent with the 107.8 ft. containment floodup level value in the above table being specified for flow from the IRWST, the design value of hydraulic resistance for the broken DVI line input into WCOBRA/TRAC is increased to include an additional loss coefficient (K-factor) of 1.5 to conservatively represent the exit loss for flow from the severed pipe into the PXS room and the subsequent entrance loss from the room into the pipe segment connected to the DVI nozzle.

In sensitivity case 1, the resistance at the core entrance due to postulated blockage equals 2.4*10-6 ft/gpm2; this value is approximately five orders of magnitude greater than the (unblocked) AP1000 core entrance resistance value used in the DCD long-term cooling case. In sensitivity case 2, the resistance at the core entrance due to postulated blockage is double that of sensitivity case 1.

c. The plots of integrated core boiloff rate and integrated core inlet flow rate are provided for the DCD long-term cooling analysis presented in Chapter 15.6.5.4C for both the IRWST injection and the containment recirculation segments of the DEDVI break transient, and for the two sensitivity cases, during the containment recirculation phase.

Each containment recirculation phase case is a window mode computation that begins at 6500 seconds WCOBRA/TRAC problem time and ultimately reflects the quasi-steady-state containment floodup level. The core inlet flow rate integrals show less liquid enters the core in Sensitivity Case 1 than in the DCD analysis, and that less liquid enters the core in Sensitivity Case 2 (the higher resistance sensitivity case) than Sensitivity Case 1. Thus, lower core inlet flow results from the reduced DVI flow rates that are predicted for containment recirculation as a consequence of the postulated sump screen blockages.

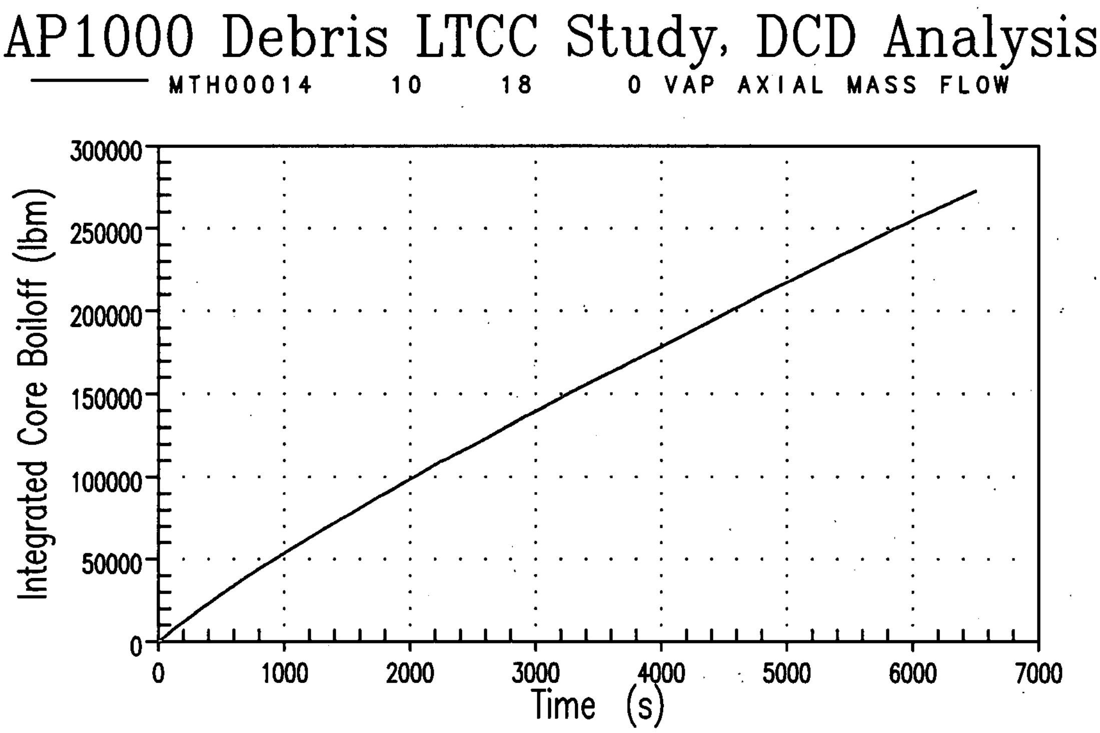

Figure RAI-SRP 6.2.2-SRSB-11c-1: DCD Chapter 15.6.5.4C Analysis Integrated Core Boiloff Rate, IRWST Injection Phase

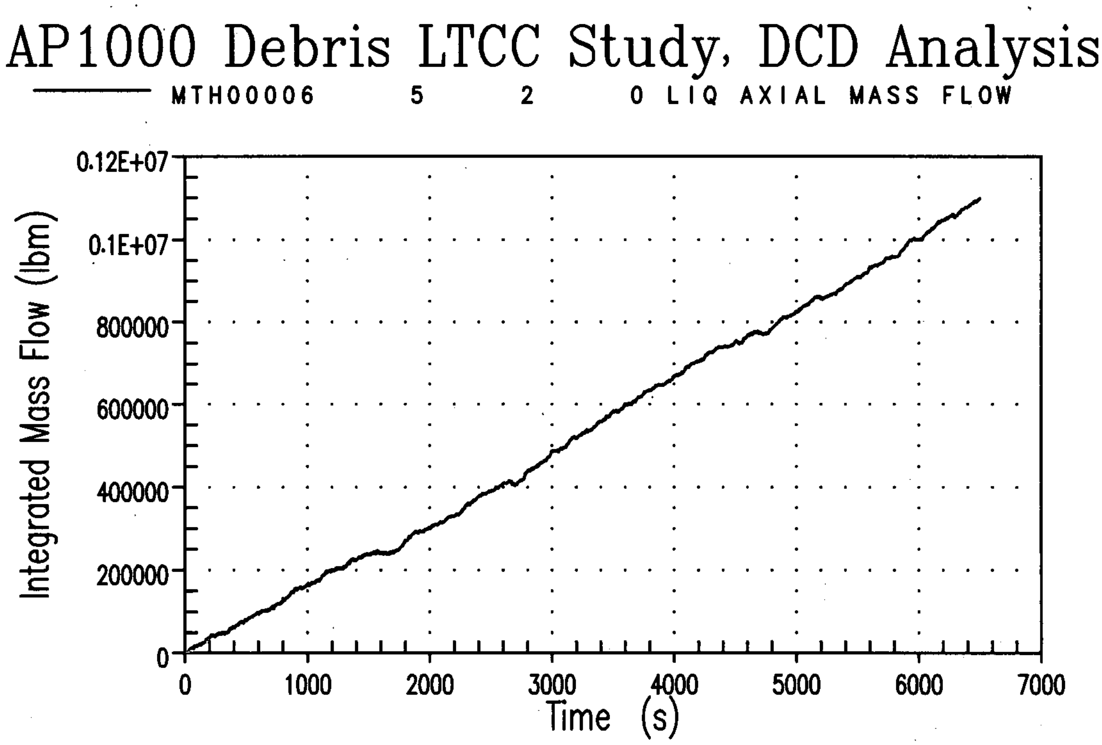

Figure RAI-SRP 6.2.2-SRSB-11c-2: DCD Chapter 15.6.5.4C Analysis Integrated Core Inlet Mass Flow Rate, IRWST Injection Phase

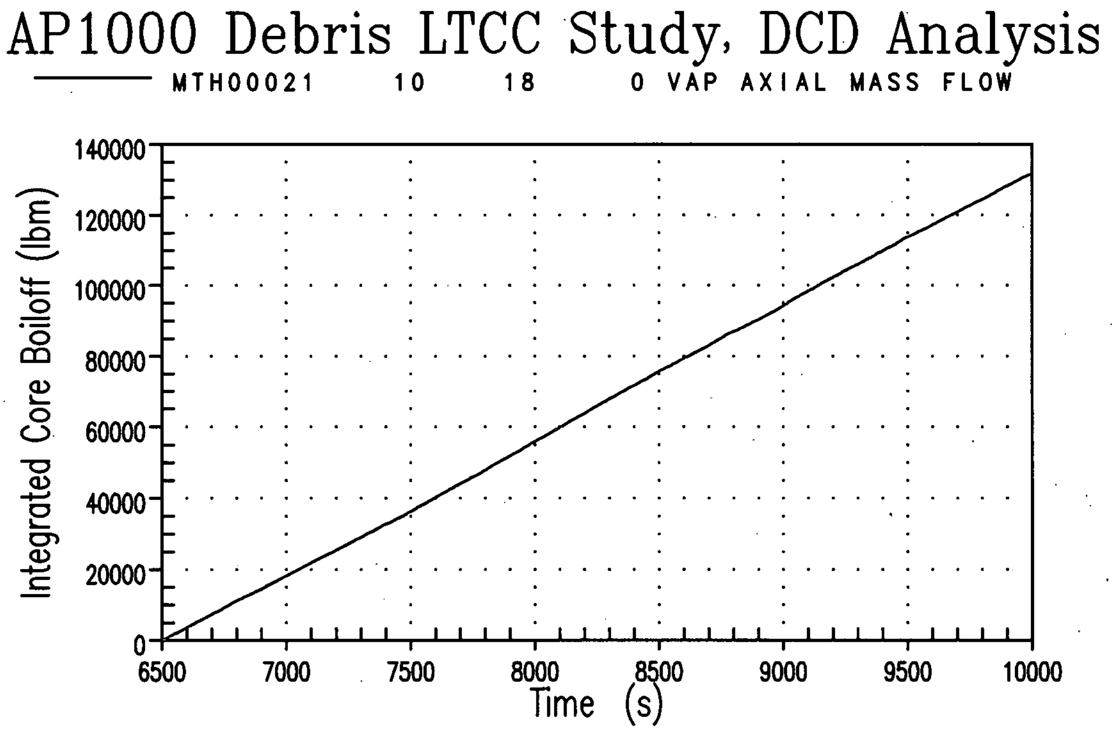

Figure RAI-SRP 6.2.2-SRSB-11c-3: DCD Chapter 15.6.5.4C Analysis Integrated Core Boiloff Rate, Containment Recirculation Phase

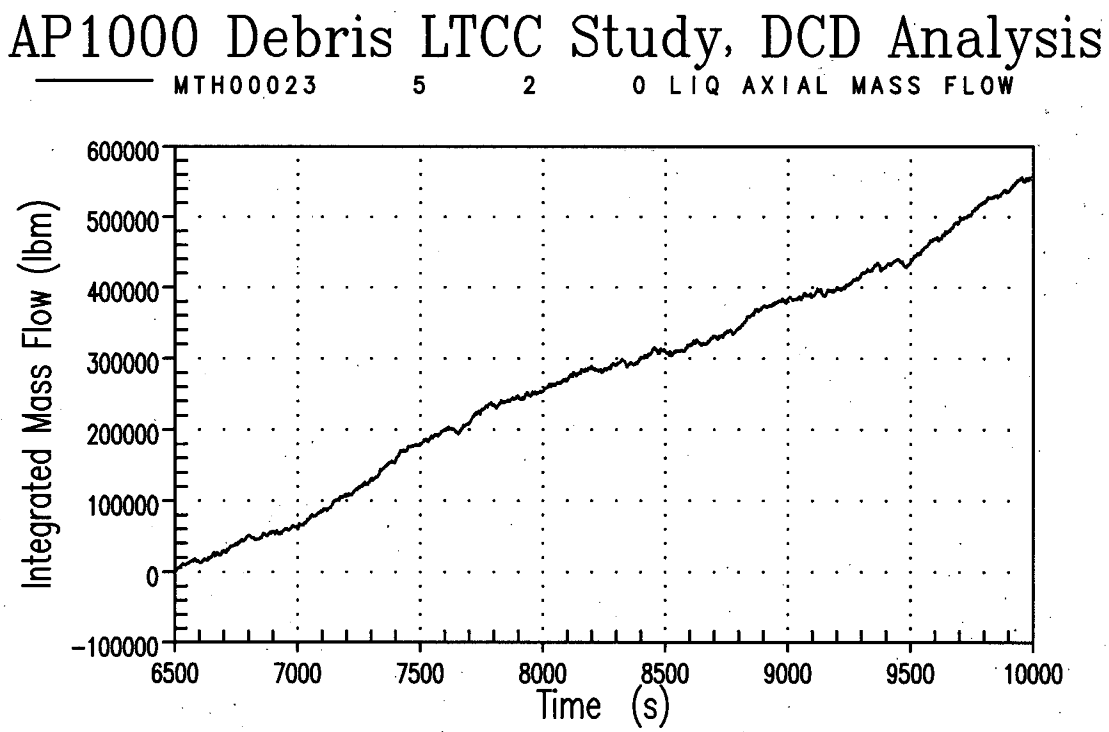

Figure RAI-SRP 6.2.2-SRSB-11c-4: DCD Chapter 15.6.5.4C Analysis Integrated Core Inlet Mass Flow Rate, Containment Recirculation Phase

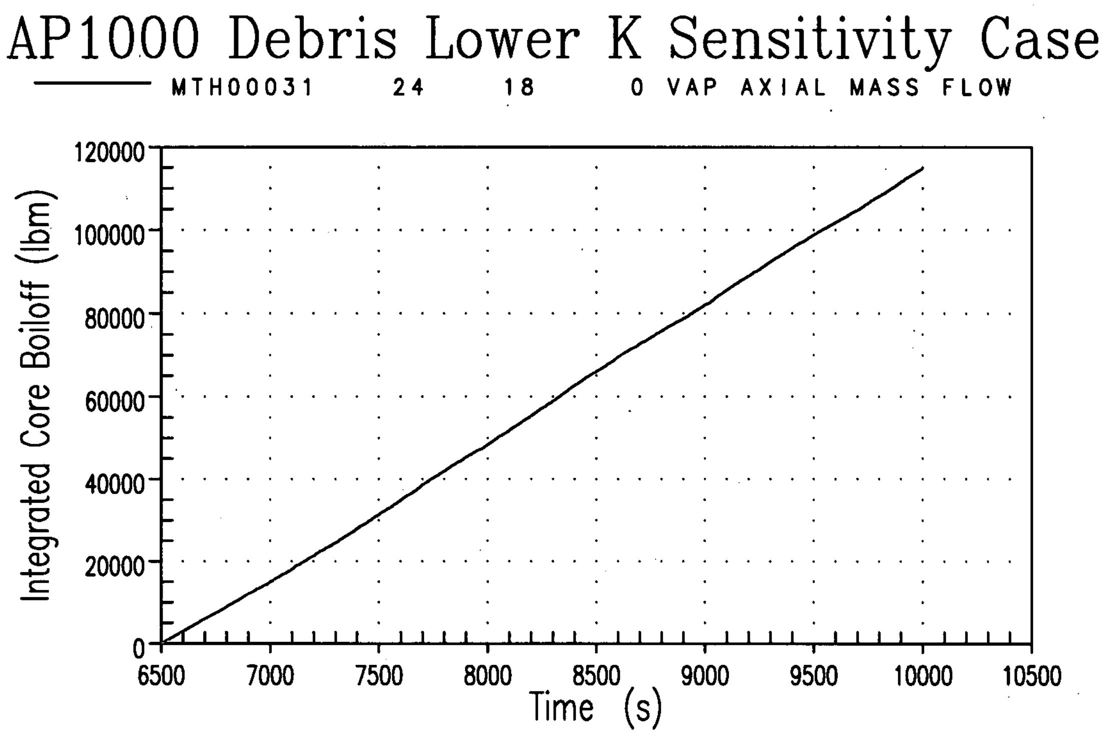

Figure RAI-SRP 6.2.2-SRSB-11c-5: Containment Recirculation Sensitivity Case 1 Integrated Core Boiloff Rate, Containment Recirculation Phase

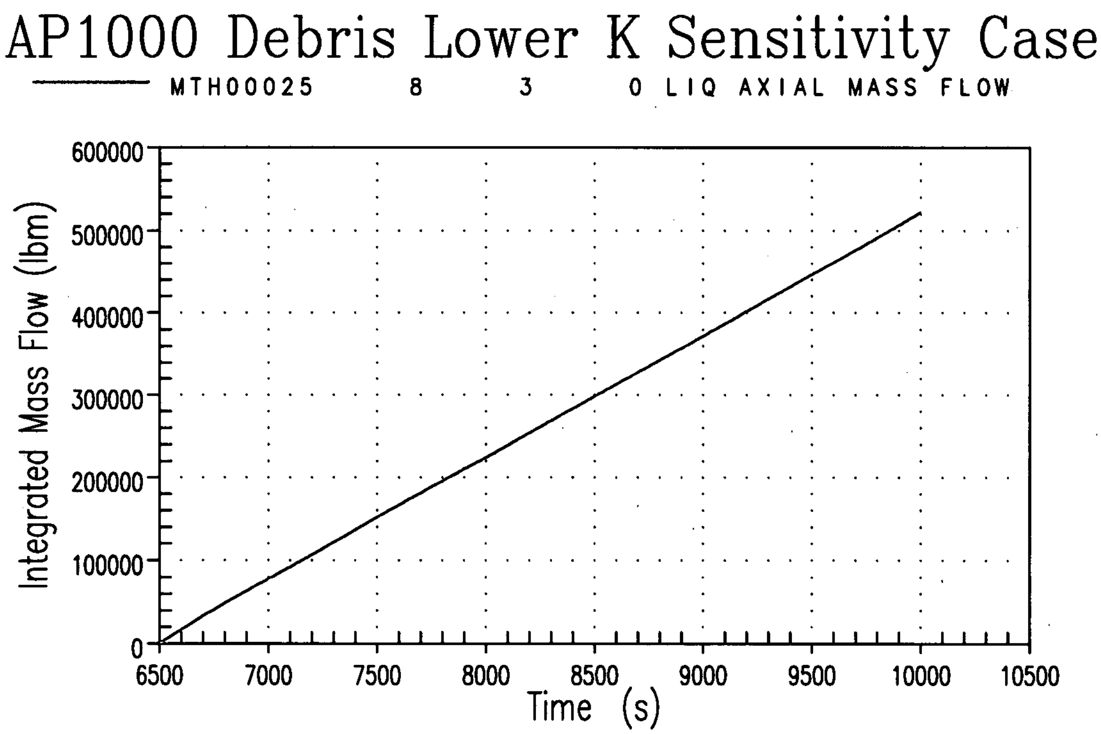

Figure RAI-SRP 6.2.2-SRSB-11c-6: Containment Recirculation Sensitivity Case 1 Integrated Core Inlet Flow Rate, Containment Recirculation Phase

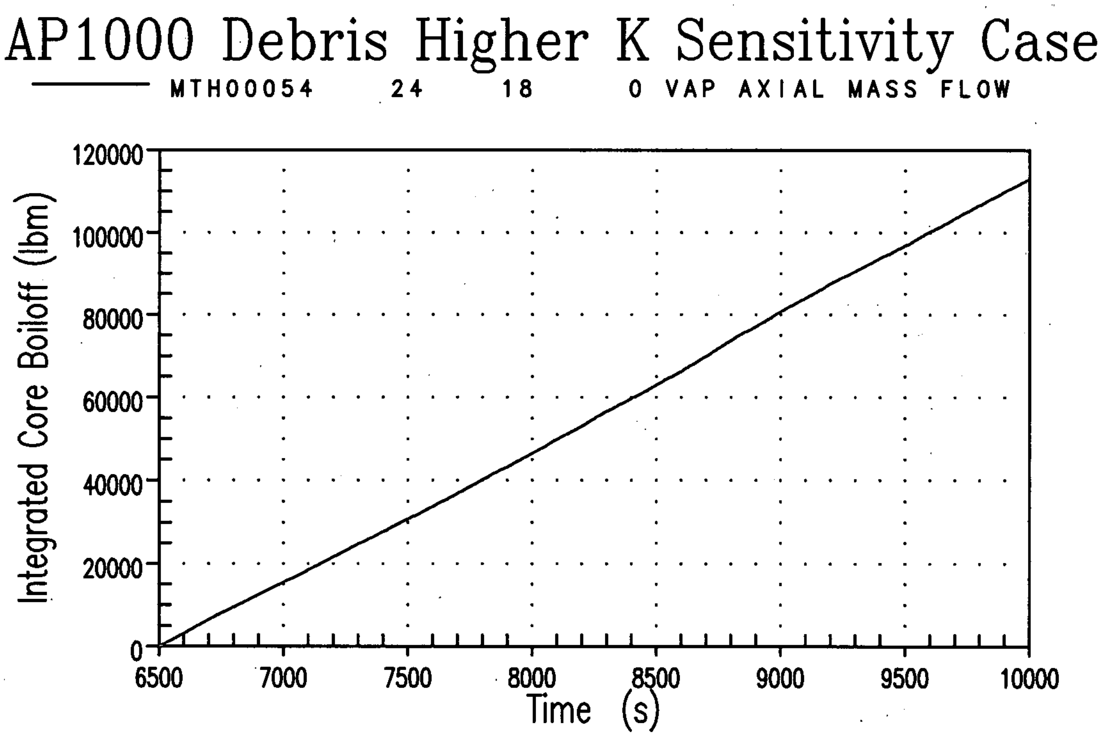

Figure RAI-SRP 6.2.2-SRSB-11c-7: Containment Recirculation Sensitivity Case 2 Integrated Core Boiloff Rate, Containment Recirculation Phase

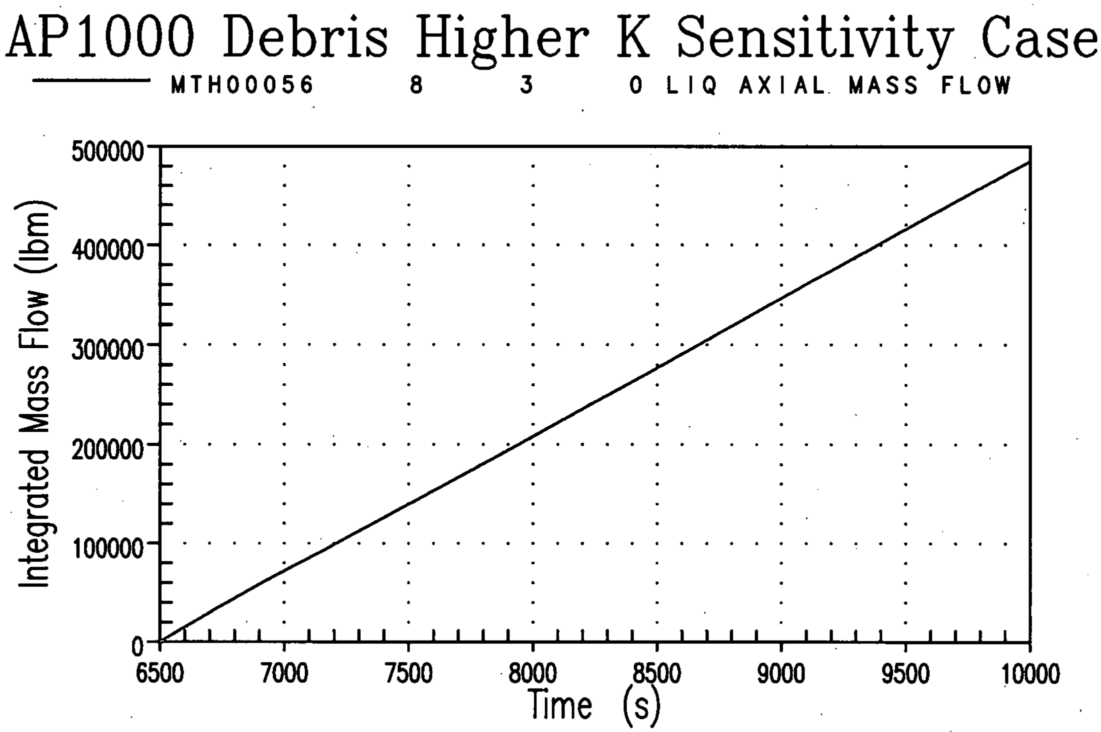

Figure RAI-SRP 6.2.2-SRSB-11c-8: Containment Recirculation Sensitivity Case 2 Integrated Core Boiloff Rate, Containment Recirculation Phase

d. The DCD analysis collapsed liquid level in Figure 2-2 is actually essentially constant over the final 1800 seconds of the calculation, as are the corresponding levels in the two sensitivity cases. In both the DCD analysis and the sensitivity cases, the core level decreases at the start of containment recirculation in response to the containment water level boundary condition presented in the part (b) response. Once the containment water level boundary condition becomes constant, the core collapsed liquid level reaches an equilibrium with it.

The long-term cooling behavior within the reactor vessel is a manometric phenomenon in which liquid in the vessel downcomer proceeds into the core on a net flow basis, as shown in the part (c) response. However, on a microscopic time scale, the flow at the core inlet can fluctuate back and forth, reversing direction when the downcomer and core collapsed liquid levels ebb and flow back and forth due to boiling heat transfer effects in the core. In the design basis (DCD) case, with a small resistance at the core inlet, the manometric flow direction can readily change as core boiling continues. However, in the sensitivity cases the core entrance resistance is orders of magnitude higher, making it much more difficult for reversals in flow to occur due to the manometric effects. The core entrance flow is much more stable in the sensitivity cases because the manometer fluctuations between core and downcomer are damped while the core inlet flooding rate decreases only minimally. The downcomer level in these cases increases to provide the liquid driving head necessary to overcome the increased resistance postulated postulated at the core inlet with little reduction in flow rate.

The continuous manometric perturbations predicted at the core inlet in the DCD case cause fluctuations in flow within the core which impact its predicted collapsed liquid level. The void fraction in the core is higher in the DCD case due to the impact of the inlet flow fluctuations, so the predicted collapsed liquid level is lower in the DCD case with equivalent decay heat removal.

e. During AP1000 long-term cooling, abundant liquid is continuously available in the reactor vessel during the containment recirculation quasi-steady-state process. With abundant liquid present, the predicted upper plenum collapsed liquid level is not directly related to the core inlet flow rates.

The upper plenum two-phase mixture level is a level swell phenomenon that is a function of the core boiloff flow rate and the interfacial drag prediction. Since the containment recirculation phase decay heat values are the same in all cases, the core boiloff rates are almost the same; therefore, the predicted interfacial drag in the upper plenum is about the same in every case, and the predicted upper plenum void fractions are about the same. The presence of the hot legs establishes the same maximum height of two-phase mixture in the upper plenum for every case. Because the same two-phase mixture level and approximately the same void fraction are present in the upper plenum in each case, it follows directly that the collapsed liquid level values are also approximately the same.

f. Technical Report 26 ,APP-GW-GLR-079, Revision 3, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA", March 2008 provides this information on pages 22-26. Additional information on head loss across the core due to debris (latent and chemical) is provided in the report (APP-FA01-T2R-001, Revision 0, "Evaluation of Debris Loading Head Loss Tests for AP1000 Simulated Fuel Assembly During Post-Accident Recirculation", August 2008.) that provided the results of testing performed for an AP1000 simulated fuel assembly.

g. The venturi line has a 4 inch inside diameter. It has been included in the plugging effects analysis, and was shown to have no impact on flow.

h. A DEDVI LOCA located in a PXS room results in the limiting long-term cooling thermal hydraulic conditions as noted in the five bulleted discussion points. It is also why this case was selected for the limiting case analyzed in the DCD for long-term core cooling. The comment in the second bullet about there being less debris available for injection in this case just points out the conservatism of the sensitivity study performed. The head losses assumed in the sensitivity studies are not based on specific debris loadings for the AP1000.

Rather they demonstrate the capability of the AP1000 to operate with significant head losses even with the most limiting LOCA break location. If the LOCA was not located in a PXS room then the thermal hydraulic conditions would be more favorable and the plant could tolerate higher head losses.

Per the methodology documented in WCAP-14857, Figure 4-2, the initial segment of the AP1000 DCD DEDVI break transient analysis is performed with the NOTRUMP code. This LOCA break is analyzed with NOTRUMP from inception until continuous injection from the IRWST into the reactor vessel has been established. The transient is then continued into long-term cooling with the WCOBRA-TRAC code, which is initialized consistent with the final NOTRUMP-predicted system condition; the WCOBRA/TRAC long-term cooling results are presented in DCD Section 15.6.5.4C. As indicated in the response to part (b), the initial 500 seconds of the WCOBRA/TRAC problem are used to allow the code to equilibrate to the end-of-NOTRUMP condition.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

RAI Response Number: RAI-SRP6.2.2-SRSB-13
Revision: 0

Question:

In TR 26, Revision 3, in Section VII, "Ex-Vessel Downstream Effects Evaluation of AP1000 Recirculation Flow Paths," on page 22, Westinghouse states:

> "In summary, the evaluation performed using the applicable methods and models in WCAP-16406-P (Reference 11) consistent with the applicable amendments, limits and conditions of the associated NRC SE on the WCAP (Reference 12) demonstrates that the AP1000 PXS equipment utilized in post-LOCA recirculation is acceptable for the expected debris loading in the recirculating fluid resulting from a postulated LOCA."

a. Describe the details of this evaluation, including the specific methods and models used from WCAP-16406-P.
b. Identify in detail how each of the differences between the PWR flow paths and the AP1000 flow paths were evaluated.
c. Identify whether the use of applicable PWR methods and models in WCAP-16406 has been validated with respect to the AP1000 design.

Westinghouse Response:

a. The PWR Owner's Group (PWROG) sponsored the development of a methodology (WCAP-16406-P-A, Rev. 1), and data collection effort to evaluate the effects of debris ingested into the ECCS and CSS during post-accident operation for Pressurized Water Reactors (PWRs) in the current operating fleet. The NRC has reviewed and accepted this evaluation methodology, with certain limits and conditions, as described in their Safety Evaluation (SE) of WCAP-16406-P-A, Rev.1.

b. The flow paths of the debris laden water used in post-LOCA recirculation through the PXS are described below. Valves in the flow path of the PXS for post-LOCA recirculation are evaluated. Recirculation begins at Containment Recirculation Screen A, MY Y02A, (train A) and Containment Recirculation Screen B, MY Y02B, (train B). In train A, flow splits after passing through screen MY Y02A. The first line, L131A, passes through gate valve 117A and then squib valve 118A. The second line, L113A, passes through check valve 119A and then squib valve 120A. These two lines then combine to form line L132A. In train B, flow passes through screen MY Y02B and then splits after passing through L100. The flow through the first line, L131B, passes through gate valve 117B and then squib valve 118B. The flow through the second line, L101, passes through check valve 119B and then squib valve 120B. These two lines then combine to form line L132B. Because trains A and B are identical from this point forward, train A will be used to describe the flow path in this section. According to APP-PXS-M6-002 Rev. 1, following the junction of these two lines, recirculation flow from the IRWST Screen, MY Y01A, line L112A, is added to line L116A.

The combined flow moves through gate valve 121A and into line L117A, and then splits into two lines. The flow from the first line, L118A, passes through check valve 122A and into line L123A. The flow then moves through squib valve 123A. The flow from the second line, L124A, passes through check valve 124A into line L125A. The flow then moves through squib valve 125A. These two lines then combine to form line L127A. Flow from line, L127A, enters the direct vessel injection line (L021A). For train A, temperature elements TE 003 and TE 005 are strap on non-intrusive RTDs. For train B, temperature elements TE 004 and TE 006 are strap on non-intrusive RTDs. The flow from both trains A and B then enters into the reactor vessel through the Direct Vessel Injection (DVI) nozzles. It should also be noted that there is a venturi inserted into the nozzle of the DVI lines.

c. The data and methods used to evaluate ex-vessel downstream effects are outlined in WCAP-16406-P, Revision 1 "Evaluation of Downstream Sump Debris Effects In Support of GSI-191," May 2006. The evaluation methods identified in WCAP-16406-P Revision 1 that are applicable to long-term core cooling recirculation flow paths associated with the AP1000 design include:

- The fuel blockage evaluation as described in Section 5. This particular downstream effects evaluation method addresses the core evaluation from the NRC comment. Note that APP-FA01-T2R-001, Revision 0, "Evaluation of Debris Loading Head Loss Tests for AP1000 Simulated Fuel Assembly During Post-Accident Recirculation", August 2008, provides test results of a simulated AP1000 fuel assembly that demonstrates that with debris loadings applicable to the AP1000, head loss in the fuel assemblies will not be excessive.
- Valve evaluations for plugging and erosive wear are described in Sections 7 and 8 and Appendix F. The screening criteria for valves that are identified in Revision 1 to WCAP-16406-P are applicable to valves in the long-term core cooling recirculation flow path of PWR's. Only the explosively actuated (squib) valves in the post-LOCA flow path are not covered by the screening criteria. Once the squib valves are open they exhibit, very closely, the characteristics of a standard gate valve.

There are design features of the AP1000 that eliminate the need for downstream effects evaluations of components that are included in Revision 1 of WCAP-16406-P. Evaluations excluded by the AP1000 design include:

- Pump evaluations, including hydraulic performance, disaster bushing performance, and vibration analysis. There are no safety related pumps in the AP1000 passive core cooling flow paths to evaluate.
- Heat exchanger evaluations for both plugging and erosive wear. There are no heat exchangers in the post-LOCA recirculation flow paths of the AP1000 design.
- Orifice evaluations for plugging and erosive wear as described in Sections 7 and 8 and Appendix F. There are no orifices in the post-LOCA recirculation flow path of the AP1000 design.

- Settling of debris in instrumentation lines as described in Section 8. There are no instrumentation lines used in the AP1000 post-LOCA containment recirculation flow path design that are required to support a safety related function.
- Containment Spray System (CSS). The AP1000 does not have a conventional CSS. The non-safety containment spray function is not permitted to be used during a DBA. Therefore, this system is excluded from consideration of the AP1000 design.

Thus, where applicable design features exist in the AP1000, the data and methods identified in Revision 1 of WCAP-16406-P are applied to evaluate ex-vessel downstream effects for the AP1000 design.

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-SRP6.2.2-SRSB-14
Revision: 0

Question:

In TR 26, Revision 3, in Section VII, "In-Vessel (Core) Downstream Effects Evaluation Method," on page 23, Westinghouse states:

> "The calculation method of the LOCADM spreadsheet is described in WCAP-16793 (Reference 13). The evaluation makes some simplifications to the required inputs that are conservative for this evaluation. These data and methods are applicable to the AP1000 for the following reasons ..."

Provide a more detailed description of those simplifications and justify why those simplifications are conservative for the AP1000. Confirm that the simplifications presented on pages 23 and 24 of TR-26 are the only simplifications made to the original PWR analyses.

Westinghouse Response:

The simplifications to the LOCADM spreadsheet that were incorporated for AP1000 are listed and described in pages 23 and 24 of TR 26, Revision 3. Of the bulleted list presented, only two items are truly simplifications in how the LOCADM spreadsheet is set up:

1. This evaluation effectively increases the aluminum surface area to conservatively account for the zinc release from galvanized steel. It is conservative to increase the aluminum amounts because the aluminum release rate is greater than any other material used in this evaluation. Although, the properties (i.e. rate) of core deposition for both aluminum and zinc are different, a bounding thermal conductivity for the chemical deposition on the fuel cladding is evaluated regardless of the material deposited on the core.
2. This evaluation uses what is called "The Pre-Filled Reactor and Sump Option". Use of this option assumes that the entire sump volume is present in the sump at time t = 0, precluding the need to specify individual break flow rates. This is also conservative, as modeling the sump as full at the start of the transient allows the chemical reactions to begin at time t = 0 and provides for the calculation of a greater amount of precipitate deposition on the fuel.

The LOCADM spreadsheet does not predict releases from galvanized steel therefore the aluminum surface area was increased as described in TR 26, Revision 3 to account for chemical precipitate releases from galvanized steel.

The "Pre-Filled Reactor and Sump Option" is a simplification of the method in which to enter the data into the LOCADM spreadsheet and does not alter the basis of the calculations.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None