Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Non-Proprietary)

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CRR-001 *(spent fuel pool criticality)*

With the movement to the Advanced First Core (AFC), there is a change from a 3-region design to a 5-region design. With respect to spent fuel pool criticality analyses as described in APP-GW-GLR-029, Revision 3, titled, "AP1000 Spent Fuel Storage Racks Criticality Analysis,"

a) How does the change to a 5-region core affect the previously identified limiting fuel assembly depletion characteristics? For example, the current analysis in APP-GW-GLR-029 identifies the limiting assembly insert combination during fuel depletion as having [

]a,c

b) Considering the AFC and future cycle core designs, how is the limiting assembly insert combination affected?

c) Also, with a change in the core design, it is likely that the axial burnup distributions have also changed. Demonstrate that the change in core design, including the effect of reload core designs, either does not affect the limiting axial burnup distributions as discussed in APP-GW-GLR-029 or update the safety analyses in APP-GW-GLR-029 to include the appropriate distributions and analysis impacts.

d) Demonstrate that the cumulative impact of the AFC and reload core designs still satisfy the appropriate criteria in 10 CFR 50.68.

Westinghouse Response to CRR-001

a) The spent fuel pool (SFP) criticality analysis described in APP-GW-GLR-029P Revision 3 considered the Advanced First Core (AFC) design 5-region core, as well as the original 3-region first core design and anticipated reload cycles, when determining the limiting fuel assembly types for depletion characteristics. This is evidenced by the fact that both WABA and Pyrex burnable absorber designs were considered in combination with the IFBA burnable absorber design in Table 5.14 of the subject analysis. The WABA/IFBA combination is used in the AFC design and the PYREX/IFBA combination is used in the original 3-region first core design.

   The SFP criticality analysis analyzed conservative combinations for the number of WABA and IFBA rods used in the same assembly. The limiting fuel assembly type identified in the Region 2 SFP criticality analysis used [ ]a,c The AFC design will actually employ fuel assemblies with maximum burnable absorber combinations of [ ]a,c By bounding the total amount of initial burnable absorber material used in the first cycle (i.e., as quantified by the total Boron-10 initially contained within both the IFBA and WABA rods), the reactivity of the fuel at the time of its final discharge into Region 2 of the SFP is increased, due to variations in the plutonium isotopic content of the discharged fuel.

In addition, both the silver-indium-cadmium and tungsten gray rod designs were considered in the depletion calculations performed to support the Region 2 SFP criticality analysis, as evidenced in Table 5.14 and Appendix F of the subject analysis.

Therefore the implementation of the AFC design and the tungsten gray rod design do not affect the selection of the most limiting fuel type identified in the Region 2 SFP criticality analysis.

b) As discussed above, the depletion calculations performed to support APP-GW-GLR-029P Revision 3 considered the AFC design and typical reload cycles subsequent to the AFC design. This included the potential placement combinations of the most limiting fuel type during its second and third cycle of operation. The selected limiting assembly insert combination has already taken into account the AFC design and subsequent reload cycles.

c) The axial burnup distributions considered in APP-GW-GLR-029P Revision 3 were developed based on both rodded and un-rodded operation of the AFC design and subsequent reload cycles, including both Cycle 2 and a representative equilibrium cycle reload design. The most limiting axial burnup distributions were determined by comparing the relative burnups near the top of the fuel assembly as described in Section 7.2.2 and Appendix E of APP-GW-GLR-029P Revision 3.

The most limiting axial burnup shape used in the SFP criticality analysis is provided in Table 5.11 of APP-GW-GLR-029P Revision 3. This limiting axial burnup shape came from an end of Cycle 1 fuel assembly, taken from the AFC design, which was located directly under a partially inserted RCCA for the entire cycle. The average burnup of this assembly was relatively low at the end of Cycle 1, but the relative axial shape of the burnup distribution is the most limiting, due to the low exposure in the top of the fuel assembly relative to the assembly average. This limiting axial burnup shape was applied to all burnup credit calculations by scaling the relative burnup shape up to higher assembly average burnup values. This approach is very conservative because, in reality, the axial burnup distributions from fully discharged fuel operating over multiple cycles in different core locations would tend to have higher relative burnups near the top of the fuel than are simulated with the scaled axial burnup shape taken from the partially rodded location at the end of Cycle 1. This is demonstrated graphically by the plot of "Weighted Relative Burnup as a Function of Distance from the Top" for fuel from different reload cycles, which is shown on Page E-10 of APP-GW-GLR-029P Revision 3.

Since the limiting axial burnup shape was developed in a conservative manner with appropriate consideration given to both the AFC design and limiting rodded operation, the results of the Region 2 SFP analysis in APP-GW-GLR-029P Revision 3 remain unaffected.

d) As discussed in the above response to CRR-001, the SFP criticality analysis performed in APP-GW-GLR-029P Revision 3 considered all aspects the AFC design 5-region core, as well as the original 3-region first core design and anticipated reload cycles. Therefore the results presented in APP-GW-GLR-029P Revision 3 remain valid and still satisfy the regulatory criteria in 10CFR 50.68.

While the analysis documented in APP-GW-GLR-029P Rev. 3, envelopes both the core and fuel design associated with the Advanced First Core (AFC), a minor discrepancy in the analysis was identified and entered into the Westinghouse Corrective Action Process. Specifically, it was noted that the diameter of the GRCA absorber assumed in the analysis is not consistent with the final design of the GRCAs used in the new core design (i.e., the actual absorber diameter is slightly larger than assumed in the analysis). A preliminary evaluation indicates that the impact of this discrepancy is not significant and would have no impact on the final conclusions of the analysis.

The revised analysis will be available to support a formal resolution of this issue by January 31, 2013.

CRR-004 *(LBLOCA Analysis)*

In the large-break LOCA analysis (Calculation Note APP-SSAR-GSC-772, Rev. 0, "Evaluation of TCD for AP1000 Advanced First Core Application Program and DCD Rev. 19 Best-Estimate LBLOCA ASTRUM Analyses,") using the ASTRUM method to evaluate the impact of the fuel thermal conductivity degradation (TCD), the average fuel assembly burnups are limited to [ ]^{a,c}.

Describe the processes that the average assembly peaking factors and burnups are calculated so as not to underestimate the initial stored energy of the average fuel assemblies. Provide justifications of limiting the burnups of the average assemblies to [ ]^{a,c}.

Westinghouse Response to CRR-004

The average assembly radial peaking is selected to preserve the total core power given the peaking factors of the other rod groups. The average assembly burnup is calculated as follows:

[ ]^{a,c}

Where: Sampled Time-in-Cycle ranges from [ ]^{a,c} for the **AP1000**® plant design.

This results in an analyzed average rod burnup range of approximately [ ]^{a,c}. The expected burnup range of the average rods is [ ]^{a,c}. The difference in the end of cycle burnup results in a maximum average fuel temperature difference of about [ ]^{a,c} at the peak power elevation in the average rods (the difference is slightly less at lower power elevations).

Sensitivity studies were executed for two of the TCD analysis cases, run031 (the limiting ASTRUM case prior to the TCD analysis) and run069 (the limiting ASTRUM case from the TCD analysis), to determine the impact of increased fuel temperature in the average rods. The average rod burnup for these studies was set to [ ]^{a,c}, which results in a higher than expected average rod maximum average fuel temperature.

The average fuel temperature for the average rods at steady-state conditions for both sensitivity studies are shown in Figures 1A and 1B. As expected, the average rod fuel temperatures increased in the sensitivity studies at the increased burnup. This leads to an increase in the average rod peak cladding temperature (PCT) during the Large Break LOCA (LBLOCA) transient as shown in Figures 2A and 2B. However, it is observed that the increased temperature in the average rods does not substantially impact the global response and had only a small impact on the calculated hot rod PCTs (Figures 3A and 3B). As such, it is concluded that the modeling approach for the average rod burnup is acceptable for the **AP1000** plant TCD analysis.

a,c

[ ]

Figure 1A: Average Fuel Temperature in the Average Rods for ASTRUM Case 031 Sensitivity Study

a,c

[ ]

Figure 1B: Average Fuel Temperature in the Average Rods for ASTRUM Case 069 Sensitivity Study

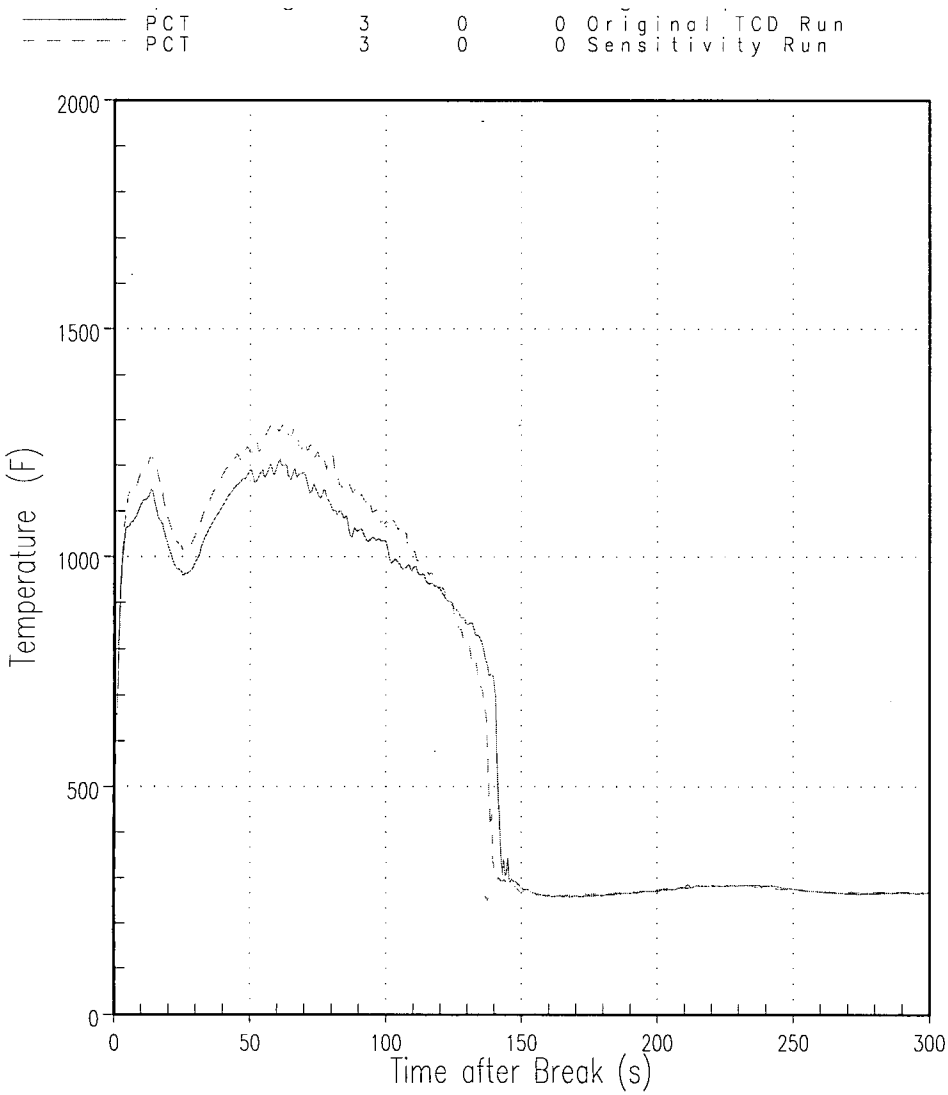

Figure 2A: Average Rod Peak Cladding Temperature for ASTRUM Case 031 Sensitivity Study

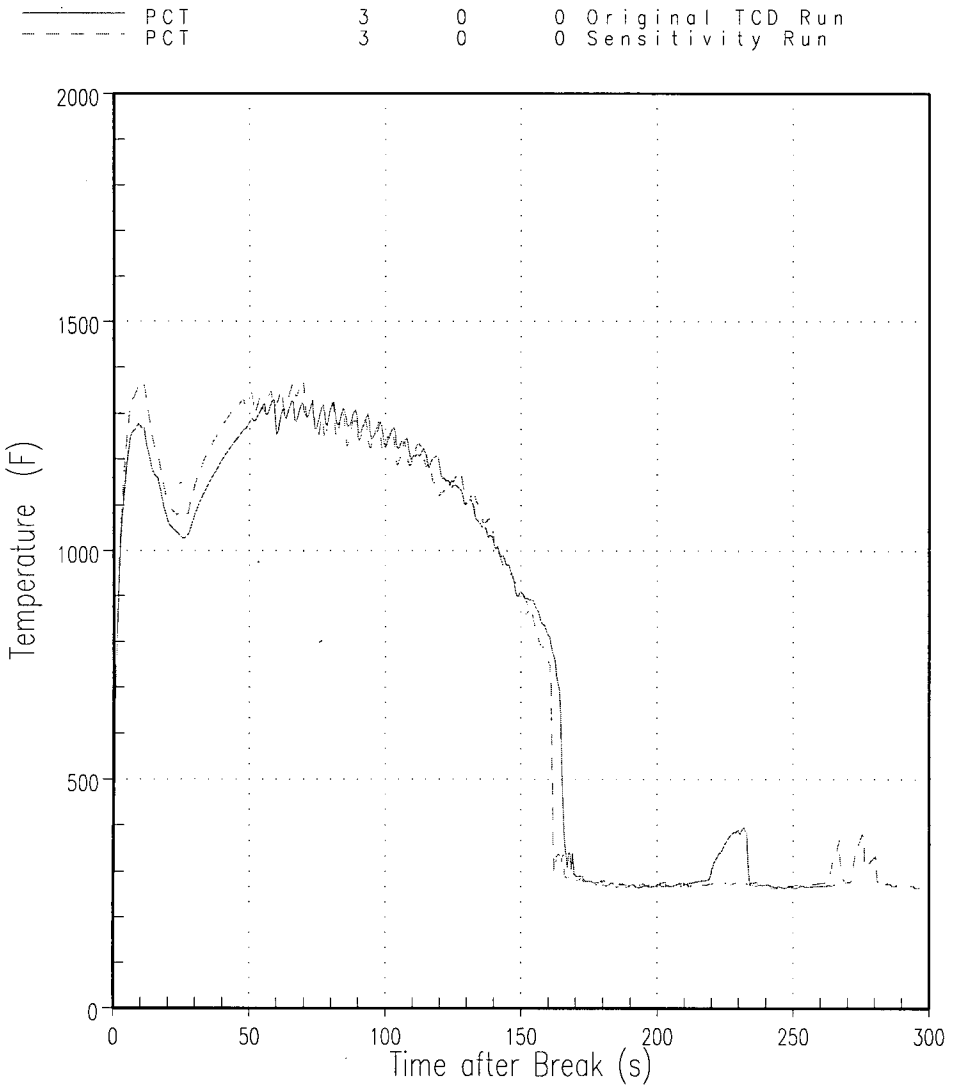

Figure 2B: Average Rod Peak Cladding Temperature for ASTRUM Case 069 Sensitivity Study

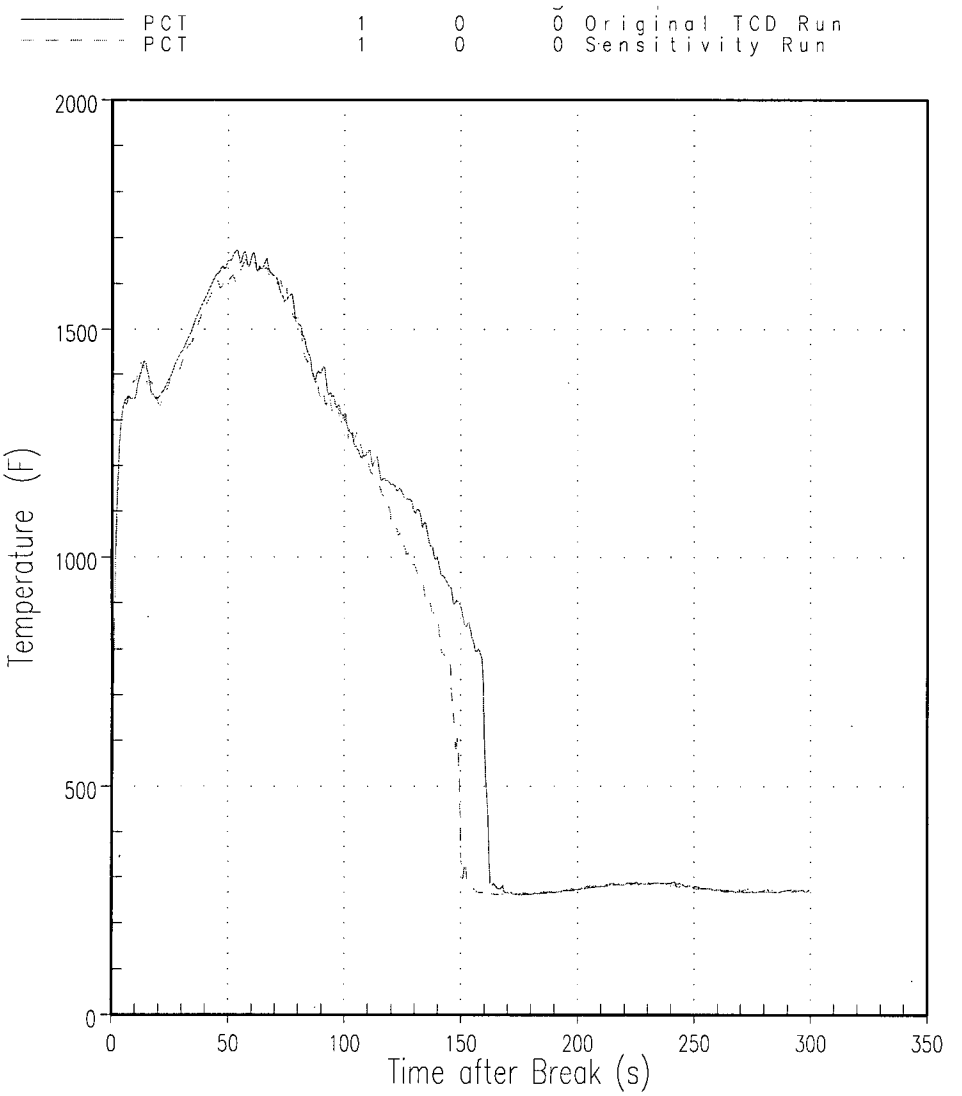

Figure 3A: Hot Rod Peak Cladding Temperature for ASTRUM Case 031 Sensitivity Study

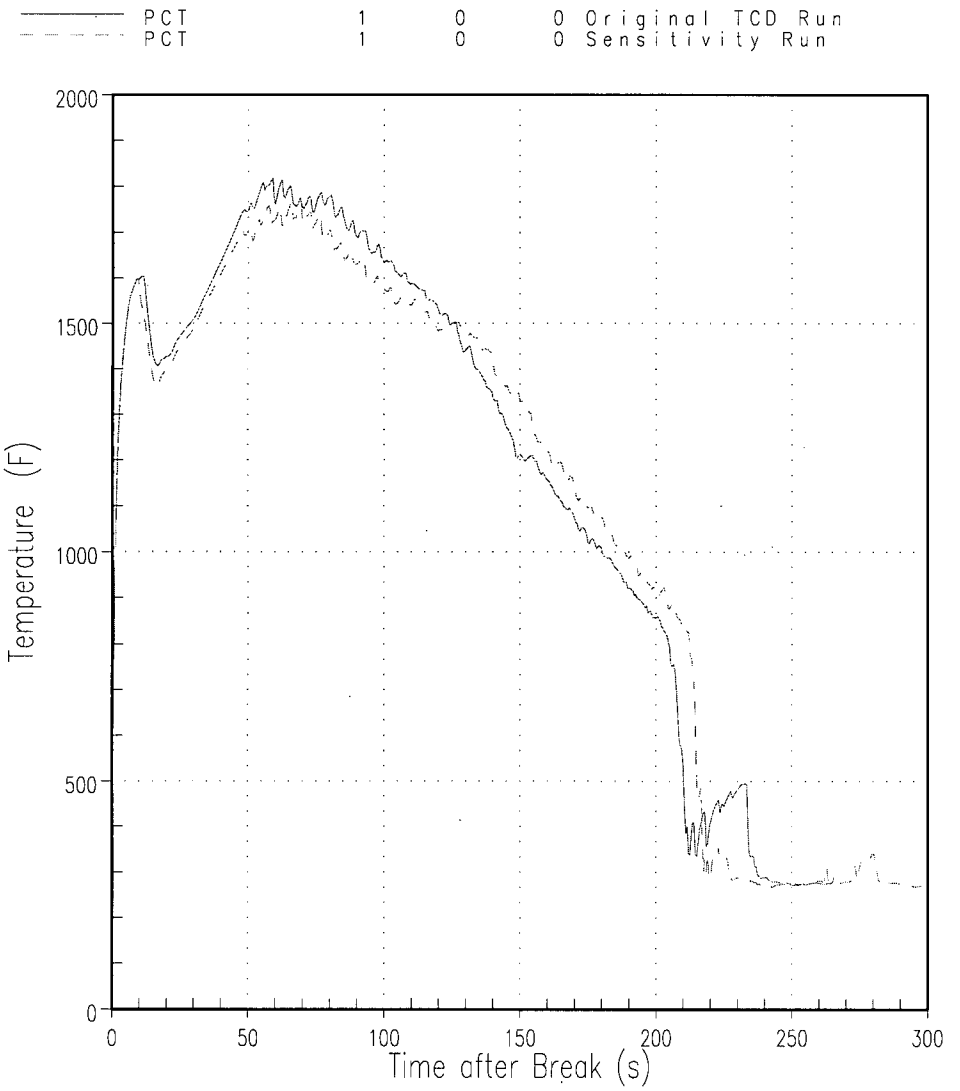

Figure 3B: Hot Rod Peak Cladding Temperature for ASTRUM Case 069 Sensitivity Study

CRR-016 *(Calc Notes-Open Items)*

During the audit held Aug 14-16, 2012, it was noted that some of the calc notes used to support WCAP-17524-P contain open items. The submitted topical report must be based on final analyses following approved quality assurance programs. Therefore, close all related open items and update WCAP-17524-P if necessary.

Westinghouse Response to CRR-016

Westinghouse has identified all open items in the calculation notes that support the Core Reference Report and has classified them into the following three categories:

1) Administrative/Editorial Corrections- These particular open items are administrative items that were identified in the calc notes that were reviewed during the Core Reference Report (CRR). These particular items will be resolved and the open items will be closed.

2) Parameter verification open items will be updated in the configured database. This will enable approximately 32 open items to be verified and closed.

3) Validate inputs that have been used in the updated safety analysis calculations are still the appropriate values and are still valid. If parameters were revised for any reason, an evaluation will be performed to demonstrate that the conclusions for updated safety analysis are still valid.

Several open items are longer lead type subjects (i.e. 4-loop mixing data) that will take several months to complete all the work.

Westinghouse intends to disposition a large portion of those open items that were identified during the NRC audit with the remainder to be dispositioned as expeditiously as possible to support the issuance of the Core Reference Report SER for June 2013.

Formal closure of all open items will occur in conjunction with revised calculations per our current **AP1000** plant Quality Plan.

CRR-018 *(non-LOCA accidents)*

As part of this topical report, Westinghouse developed a new procedure to mitigate the vessel head vent opening event by discharging high temperature and high pressure primary coolant into the in-containment RWST. Similar approaches have been developed for loss of feedwater heater event, inadvertent opening of CMT valves and CVCS malfunction. It was indicated by Westinghouse that the discharge of primary coolant into the containment will cause high containment air space temperature. Demonstrate that the in-containment equipment qualification program has already taken into account these containment heat up events. And the maximum allowable containment air space temperature for all the equipments required for operation is higher than the maximum containment air space temperature during these AOO events.

Westinghouse Response to CRR-018

The equipment qualification program methodology is described in DCD appendix 3D. The qualification of a safety-related component is based on an aging program followed by qualification for seismic, high temperature, high pressure, high radiation, and post accident chemistry conditions. The aging program includes the effect of temperature for normal plant operation plus 72 hours of accumulated Abnormal Group 1 conditions and 24 hours of accumulated Abnormal Group 2 conditions. The Abnormal Group 1 qualification temperature is 150°F. The Abnormal Group 2 qualification temperature is 250°F. Events such as a core makeup tank inadvertent injection and a CVS malfunction are considered to be Group 1. Long term PRHR operation or a small loss of coolant accident is considered to be Group 2. If these less severe accidents occur, the safety-related components are still within their aging basis, and the components are qualified for the more severe Condition IV LOCA or main steam line break because the qualification program includes aging for the components that undergo full Condition IV accident qualification.

There are no specific containment temperature/pressure analyses for all the Abnormal Group 1 and 2 design basis events. It is acknowledged that long term operation of the passive residual heat removal and mass input to the IRWST from the reactor vessel vent pipe will eventually cause an increase in containment temperature, containment pressure, and potentially some IRWST overflow. Most of the reason for the increase is the refueling water boiloff due to decay heat and not the head vent mass flow. For long term passive heat removal from the PRHR heat exchanger to the refueling water, a containment temperature of 230°F is calculated. This is less than the bounding 250°F basis for and Abnormal Group 2 event. The small LOCA containment temperature is less than 230°F. Therefore, a safety-related component in containment would be still qualified for additional Abnormal Group 1, Abnormal Group 2, plus the Condition IV limiting design basis event conditions described in the Appendix 3D figures and tables, given the way that the EQ methodology is applied.

Therefore, it is demonstrated that a calculation exists for the accidents that create the Abnormal Group 2 environment, and it has been demonstrated that the qualification methodology accounts for the Abnormal Group 2 events. Opening of core makeup tank valves or a CVS malfunction do not cause a high containment temperature. These events are considered to be Abnormal Group 1. A 150°F temperature is a conservative limit for events that do not release significant additional energy to containment.

As long as these PRHR and/or head vent-actuation events do not exceed a containment temperature of 250°F for an accumulated duration of 24 hours, then the affected components are bounded by component pre-accident aging design basis conditions. On top of the pre-aging conditions, the complete equipment qualification program includes Condition IV (e.g., main steam break or LOCA) seismic, radiation, chemistry, pressure and temperature conditions as defined in the equipment qualification methodology. If the accumulated Abnormal Group 1 and 2 accidents exceed the aforementioned aging bases, Westinghouse would no longer be able to claim that safety-related equipment, including the head vent valves, are qualified, and equipment replacement, additional testing, or further evaluation would be required.

CRR-023 *(LOCA)*

Westinghouse has used the previously approved containment mass and energy methodology to perform containment peak pressure calculation during LOCA. Does the corrected thermal conductivity degradation model affect the peak containment pressure and temperature calculation?

Westinghouse Response to CRR-023

The approved methodology for calculating mass and energy releases from a large break loss-of-coolant accident (LOCA) uses a value for the initial core stored energy (CSE) as an input. The resulting LOCA mass and energy releases are used in the approved methodology for calculating the long-term containment pressure and temperature response. The method for calculating the CSE is given in the request for additional information (RAI) CRR-026. Based on the description of the calculation for CSE in RAI CRR-026, the current value that is used as an input for CSE for generating large break LOCA mass and energy releases remains conservative when the effects of thermal conductivity degradation (TCD) are considered. Therefore, there is no change to peak calculated containment pressure or temperature resulting from a large break LOCA.

CRR-027 *(Fuel Seismic)*

What is the hydraulic damping coefficient that Westinghouse plans to credit for different flow conditions in the fuel assembly seismic response analysis? How would a coast down be handled?

Westinghouse Response to CRR-027

Figure 1 summarizes Westinghouse fuel assembly damping test data in still and flowing water for water temperatures between 100°F and 300°F. These tests are described in Reference 1. An analysis of the data leads to the following observations of the trends associated with credit for damping:

1) Damping in flowing water increases slightly with vibration amplitude.

2) Damping in flowing water decreases slightly with increasing temperature.

3) Flow velocity has a strong effect on damping and results in a significant increase in damping with increasing flow velocity.

These observations are consistent with other test results that are available publicly (References 2 and 3). The considerations for application of hydraulic damping coefficients due to flowing water are discussed further below.

Fuel assembly damping force in flowing water is actually the summation of fuel structural damping in air (due to material and friction damping), viscous damping in still water and hydraulic damping in flowing water as shown in Equation (1). All three damping coefficients are non-linear.

$$F_d = c_s\dot{x} + c_v\dot{x} + c_h\dot{x} \qquad (1)$$

c_s – Structural damping coefficient in air, mainly increasing with amplitude

c_v – Viscous damping in still water, mainly increasing with vibration velocity

c_h – Hydraulic damping in flowing water, mainly increasing with axial flow velocity

<u>Vibration Amplitude and Flow Rate Dependence</u>

Both structural damping and viscous damping terms are vibration-amplitude dependent and increase with vibration amplitude. The hydraulic damping term in flowing water is not vibration-amplitude dependent and it dominates the total damping. Typical fuel assembly displacements during a seismic event with grid impact occurrences are much greater than [ ]^{a,c}. Therefore, the damping data from the flowing water tests with vibration displacements greater than [ ]^{a,c} are used to obtain the damping versus flow velocity curve shown in Figure 2.

Temperature Dependence of Flowing Water Damping

As shown in Figure 1, damping is not very sensitive to temperature. For examples [

]^{a,c} This is consistent with the data from Reference 3 where it was concluded that in the range between 70° to 600°F "damping is minimally affected by temperature in water." It is estimated, based on data from Reference 3, that the damping coefficient from 204°C to 316°C (400°F to 600°F) at 5700 liters/min (~12 ft/s) and at 3800 liters/min (~8 ft/s) is reduced by approximately 3 percentage points. [

]^{a,c}

***AP1000* Plant Pump Coast-Down**

[

]^{a,c}

a,b,c

[ ]

Figure 1: Fuel Assembly Damping Factors in Still and Flowing Water – Westinghouse Test Data

a,b,c

[ ]

Figure 2: Fuel Assembly Damping in Flowing Water

a,c

[ ]

Figure 3: RCS Pump Coast-Down

References

1. R.Y. Lu and D.D. Seel, Westinghouse USA, "PWR Fuel Assembly Damping Characteristics," Proceedings of ICONE 14, 14th International Conference on Nuclear Engineering, July 17-20, 2006, Miami, Florida, USA.

2. S. Pisapia, et al. "Modal Testing and Identification of a PWR Fuel Assembly," Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Paper #C01-4, Prague, Czech Republic, August 17-22, 2003.

3. F. E. Stokes and R. A. King, "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES).