February 26, 2007

Ms. Andrea Sterdis, Manager
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SUBJECT: WESTINGHOUSE AP1000 COMBINED LICENSE (COL) PRE-APPLICATION TECHNICAL REPORT 12 - REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MD2694)

Dear Ms. Sterdis:

By letter dated July 31, 2006 (DCP/NRC1767), you submitted AP1000 Technical Report 12, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant," which provided information related to the void swelling and IASCC in reactor vessel internals core support structure materials. The U.S. Nuclear Regulatory Commission staff has reviewed the report, and has determined that additional information is required. Our questions are provided in the enclosure. We discussed these issues with your staff on February 1, 2007. Your staff indicated that you would attempt to provide your response by March 16, 2007.

Please contact me at (301) 415-1313, if you have any other questions on these issues.

Sincerely,

/RA/

Steven D. Bloom, Senior Project Manager
AP1000 Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Project No. 740

Enclosure:
Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

WESTINGHOUSE AP1000 WCAP-16620-P, Rev 0

CONSISTENCY OF REACTOR VESSEL INTERNALS CORE SUPPORT STRUCTURE MATERIALS RELATIVE TO KNOWN ISSUES OF IRRADIATION-ASSISTED STRESS CORROSION CRACKING (IASCC) AND VOID SWELLING FOR THE AP1000 PLANT

PROJECT NUMBER 740

TR12-1 Section 3.1 of WCAP-16620-P establishes a threshold neutron fluence limit of 3 displacements per atom (dpa) for the initiation of IASCC in pressurized water reactor (PWR) reactor vessel internals (RVI) components. Section 3.1 of the report also established "PWR-specific screening criteria" based on stress state in the RVI component and fluence for fluence levels that exceed 3 dpa. For fluence levels exceeding 3 dpa, the PWR-specific screening criteria that were used in the IASCC evaluation are specified for three threshold stress levels and fluence ranges; exceeding the screening criteria (i.e., exceeding the threshold stress and fluence levels) does not imply that IASCC will absolutely occur, rather that IASCC should be considered.

a) Based on the statement from the report cited above, please indicate what is meant by "PWR-specific screening criteria." Specifically, are these screening criteria specific to the AP1000 PWR design or are they more generic criteria that may be applied to PWRs, regardless of the PWR design?

b) The WCAP 16620-P report provides a reference for the IASCC screening criteria: Electric Power Research Institute (EPRI) Letter on "Screening Criteria for PWR Internals Components Ranking and Categorization," dated November 1, 2005. Based on the staff's review, the IASCC screening criteria appear to correspond to those developed by EPRI in the MRP-175 report, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values." The WCAP-16620-P IASCC screening criteria specifically appear to fit the empirical IASCC screening curve/equation from Figure B-3 of Appendix B to the MRP-175 report: Stress = -158.4 ln(dpa) + 789.65. Please confirm whether the IASCC screening criteria from WCAP-16620-P were established based on the IASCC screening curve developed by EPRI in the MRP-175 report, Figure B-3.

c) If the WCAP-16620-P screening criteria were established based on the IASCC screening curve developed by EPRI in the MRP-175 report, please provide justification, based on environmental and material similarity, regarding how they are applicable to RVI core support structure components for the AP1000. Otherwise, please state the specific EPRI report from which the WCAP-16620-P screening criteria were taken.

Enclosure

d) Please state whether RVI components that do not exceed the IASCC screening criteria (i.e., not exceeding the threshold stress and fluence values) would ever be considered susceptible to IASCC.

TR12-2 Section 4 of WCAP-16620-P states that EPRI has established one set of criteria for evaluation of irradiation-induced void swelling in RVI components. Specifically, if the structural temperature in a component exceeds 608°F during normal operation and the fluence equals or exceeds 20 dpa then void swelling has a potential to occur. Please confirm whether these criteria were established based on the EPRI void swelling criteria in Section G.7 of the MRP-175 report. Please provide justification, based on environmental and material similarity, regarding how these criteria are applicable to RVI core support structure components for the AP1000.

TR12-3 Section 2 of WCAP-16620-P states that, with respect to IASCC, screening of RVI components is based upon the peak stress to which a component is subjected at full hot power. The staff requests further detail regarding how the peak stresses for the various RVI components were determined:

a) Peak stresses are said to be comprised of "membrane stress intensity," with additions due to "bending and stress concentrations," steady-state thermal stress additions, and high-cycle fatigue components. Please discuss how these components of the overall peak stress were calculated.

b) The report states that transients do not need to be considered in determining the peak stress. However, Table 2-1, which lists the structural stress level during normal operation, provides a footnote (Note 2) stating that the stress for the core barrel outlet nozzles includes thermal transient stresses because stresses under steady-state conditions are not presently available. Is this the only component for which the stress was calculated under thermal transient conditions? If so, please explain why transients were not considered for the remaining RVI components.

c) It is the staff's understanding that thermal stresses should be calculated assuming the greatest possible thermal gradient in the component under operating thermal transient conditions. Therefore, regarding the steady-state thermal stress additions, please clarify how thermal stresses arise under steady-state conditions and why such stresses were not calculated assuming some thermal gradient in the RVI component.

d) For RVI components that are exposed to high dpa levels at end-of-life (EOL) (e.g., core barrel outer and inner walls), do the peak stress levels take into consideration faulted conditions that are within the design basis? If not, please explain why.

TR12-4 Table 2-2 of WCAP-16620-P provides fluence values (in dpa) at EOL for the various RVI components. Please define EOL in terms of the total effective full power years (EFPY) of facility operation.

TR12-5 Please discuss how inservice inspection (ISI) will be conducted for the RVI components during the operating life of the AP1000 plant.

TR12-6 Please discuss how the EOL dpa levels were determined for the various RVI components listed in Table 2-2 of WCAP-16620-P.

TR12-7 Please discuss how the estimated structural temperatures were determined for the various RVI components listed in Table 2-3 of WCAP-16620-P. Please discuss whether any localized regions in these components could ever be exposed to temperatures in excess of those stated in Table 2-3.

TR12-8 Section 3.2 of WCAP-16620-P concludes that, based on the IASCC screening criteria established in Section 3.1 of the report, the maximum projected EOL fluence values, and peak stress levels that the RVI core support structure components are subject to at full-power conditions, none of the RVI core support structure components will meet the threshold for IASCC established by the IASCC screening criteria. It was therefore concluded that none of the core support structure components are considered susceptible to IASCC. The staff requests that Westinghouse document whether there are any localized areas in these components that could be exposed to neutron fluence levels that exceed the screening criteria established in Section 3.1 of WCAP-16620-P. If such localized areas occur, please discuss how this information would be assessed to determine susceptibility of the RVI component to IASCC.

TR12-9 Section 4.3 of WCAP-16620-P concludes that void swelling is not significant for RVI core support structure components. This conclusion is based on the fact that none of these components meet the criterion for void swelling: structural temperature in excess of 608 ^{o}F during normal operation and EOL fluence greater than or equal to 20 dpa. However, the staff found that the core barrel inner wall is listed as having an estimated structural temperature of 558 ^{o}F and an EOL fluence level of 24.8 dpa. WCAP-16620-P and MRP-175 do not address how the void swelling criterion could be applied for components with either temperature or fluence in excess of the above levels. Therefore, the staff requests that you address whether void swelling may be a significant degradation mechanism for the core barrel inner wall.

TR12-10 A previous Westinghouse report, WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," Revision 1-A established a threshold neutron fluence limit of 1 X 10^{21} n/cm^2 (E > 0.1 MeV) for the initiation of IASCC in PWR RVI components. This appears to differ from the 3 dpa IASCC susceptibility threshold fluence for the initiation of IASCC provided in Section 3.1 of the WCAP-16620-P report. Furthermore, the Babcock and Wilcox (B&W)'s BAW-2248 report, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," stipulates a threshold limit of 1-2 X 10^{21} n/cm^2 (E > 1.0 MeV). Since the threshold limits specified in reports WCAP-16620-P, WCAP-14577, and BAW-2248 appear to be different, please establish one threshold limit for the RVI components and document a basis for selecting a threshold fluence value of 3 dpa or 2 X 10^{21} n/cm^2 (E > 1.0 MeV) in light of the information in the other reports.

The following questions pertain to the technical information on IASCC and void swelling in RVI components from MRP-175. These questions specifically relate to how the WCAP-16620-P screening criteria and related conclusions regarding the susceptibility of AP1000 RVI components to IASCC and void swelling address the relevant technical information from MRP-175 concerning IASCC and void selling phenomena.

TR12-11 Section B.1.1 of the EPRI MRP-175 report states that at fluence levels greater than 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV) there is little difference in IASCC behavior between cold worked and solution annealed stainless steel materials. In light of this information, please indicate whether the 3 dpa threshold limit from WCAP 16620-P was determined taking into consideration the effect of cold work on RVI components or welds exposed to fluence levels less than or equal to 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV). If the onset of crack initiation due to IASCC is enhanced due to cold work on RVI components that are exposed to fluence levels less than or equal to 6.7 X 10^{21} n/cm^2 (E > 1.0 MeV), the 3 dpa threshold fluence limit specified in Section 3.1 of the WCAP-16620-P report for the cold worked RVI components should be reduced to a lower value.

TR12-12 Section B.2 of MRP-175 states that IASCC is affected by bulk composition of the RVI components. The threshold limits and screening criteria specified in Section B.3 of MRP-175 do not take into consideration the effect of bulk composition on IASCC mainly due to lack of experimental data. As the IASCC screening criteria from Section 3.1 of WCAP-16620-P correspond to those in MRP-175, please address whether the 3 dpa threshold fluence limit was developed taking into consideration the effect of variation in composition of the various RVI components, specifically elements like silicon, manganese and boron.

TR12-13 Section B.1.7 of MRP-175 states that the susceptibility of Alloy X-750 to IASCC increased when exposed to a neutron fluence greater than 1 X 10^{19} n/cm^2 (E > 1.0 MeV). Please discuss how the 3 dpa threshold neutron fluence value specified in Section 3.1 of WCAP-16620-P is applicable to Alloy X-750 RVI components.

TR12-14 It is widely understood that oxidizing ions are produced when the reactor coolant is exposed to neutron radiation. In RVI component crevices, these ions can play a major role in enhancing crevice corrosion. Please provide a discussion regarding the effect of crevices in RVI components on the IASCC screening criteria from Section 3.1 of WCAP-16620-P.

TR12-15 Transmutation products such as helium are known to play an important role in void swelling. In order to reduce overall interfacial energy, helium atoms will combine with vacancy clusters, thereby facilitating void nucleation and growth. Section G.1 of MRP-175 states that a potentially important aspect of void swelling in PWRs arises from transmutation of trace amounts boron, preexisting in most austenitic stainless steels, to produce lithium and helium. Section G.1 of MRP-175 indicates that at low neutron exposure (~10^{21} n/cm^2 thermal), almost all Boron-10 (20% of natural boron preexisting in trace quantities in most stainless steels) will be converted to lithium, producing helium in the process. Since the original concentration of boron in austenitic stainless steel RVI components is not generally reported in the certified material test reports, it is

difficult to assess the concentration of helium in the RVI components. The potential effects of helium on void swelling and IASCC should be taken into account in establishing neutron fluence threshold limits. Please address whether the void swelling and IASCC screening criteria from Section 3.1 of WCAP 16620-P account for the synergistic effects of helium on void swelling and IASCC in stainless steel RVI components.

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