April 8, 2003

MEMORANDUM TO: Marsha Gamberoni, Deputy Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

FROM: Joseph Colaccino, Senior Project Manager ***/RA/***
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

SUBJECT: FEBRUARY 21, 2003, TELEPHONE CONFERENCE CALL SUMMARY

On Friday, February 21, 2003, a telephone conference call was held with Westinghouse Electric Company (Westinghouse) representatives and Nuclear Regulatory Commission (NRC) staff to discuss several requests for additional information (RAIs). The following RAIs were discussed: 251.011, 251.012, 252.001, 252.002, and 281.001. Westinghouse submitted responses to these RAIs on November 8, 2002 (ADAMS Accession No. ML023170535), November 26, 2002 (ADAMS Accession No. ML023400058), and December 2, 2002 (ADAMS Accession No. ML023400058). A list of call participants is included in Attachment 1. Attachment 2 contains NRC staff comments regarding the subject RAIs that were sent to Mr. Michael Corletti of Westinghouse via electronic mail on February 19, 2003, and that were used to facilitate discussions during the telephone conference call.

Following is a brief summary of the discussions regarding the identified RAIs (see comments in Attachment 2):

RAI 251.011

Westinghouse stated that they would revise the Design Control Document (DCD) to include the response to this RAI.

RAI 251.012

Westinghouse stated that the standard to calculate the delta ferrite was ASTM A-800. The NRC staff stated Westinghouse needs to verify that this method produces an equivalent level of accuracy (plus minus 6 percent deviation between measured and calculated values).

RAI 252.001

This RAI requests a comparison of the AP1000 design to the current fleet of pressurized water reactors regarding the cracking of vessel head penetration (VHP) nozzles. With regard to the ability to access and inspect the reactor vessel head, Westinghouse stated that the methods and drawings to address the staff's inquiries have not been developed. Westinghouse stated that they would revise the RAI response to address these issues and discuss these changes

with the staff. The staff requested the DCD be revised to address unresolved issue items a, b, c, and d.
RAI 252.002

Westinghouse agreed to revise the RAI response to include the following information:

1) Clarification of which stainless steel pieces will be heat treated along with qualification testing or supporting information that stress corrosion cracking is not a concern.
2) Clarification if safe end is heat treated with the nozzle.

Westinghouse also stated that the DCD will be modified to include:

1) Clarification of Table 5.2-1 to reference safe end heat treatment (if applicable) and testing/support associated with heat treatment.
2) DCD Section page 5.2-11 discussing A-8 should be modified to make consistent with Table 5.2-1.

RAI 281.001

Westinghouse stated that they would revise the DCD to include the response to this RAI.

Docket No. 52-006

Attachment: As stated

RAI 252.002

Westinghouse agreed to revise the RAI response to include the following information:

1) Clarification of which stainless steel pieces will be heat treated along with qualification testing or supporting information that stress corrosion cracking is not a concern.
2) Clarification if safe end is heat treated with the nozzle.

Westinghouse also stated that the DCD will be modified to include:

1) Clarification of Table 5.2-1 to reference safe end heat treatment (if applicable) and testing/support associated with heat treatment.
2) DCD Section page 5.2-11 discussing A-8 should be modified to make consistent with Table 5.2-1.

RAI 281.001

Westinghouse stated that they would revise the DCD to include the response to this RAI.

Docket No. 52-006

Attachment: As stated

Docket No. 52-006

Attachment: As stated

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*See previous concurrence
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FEBRUARY 21, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
Joseph Colaccino	Mike Corletti
Edmund Sullivan	Charlie Kim
George Georgiev	Dale Wiseman
Carlyon Lauron	Ed Cummins
Andrea Keim	Rick Richal
Allen Hiser	Mr. Burke
Edward Andruszkiewicz	

Attachment 1

<u>NUCLEAR REGULATORY COMMISSION STAFF
COMMENTS THAT WERE SENT TO WESTINGHOUSE TO
FACILITATE DISCUSSIONS OF THE
REQUEST FOR ADDITIONAL INFORMATION (RAI) RESPONSES
FOR CALL HELD ON FEBRUARY 21, 2003</u>

251.011

The application does not address the impact of irradiation on the integrity of the reactor vessel (RV) internals. In particular, the peak neutron fluence for the RV internals at the end of the license period should be identified and its impact on irradiation assisted stress corrosion cracking (IASCC) and void swelling should be discussed. In addition, do the RV internals contain any cast austentic stainless steel (CASS) components? CASS RV internals components are subject to both thermal and irradiation embrittlement. Please discuss the impact of these aging effects on the integrity of the RV internals components. Since the American Society of Mechanical Engineers (ASME) Code inspections may not detect the impact of these aging effects on the RV internals, augmented inspection may be required. What augmented inspections will be performed by potential AP1000 licensees to detect these aging effects?

The Materials Reliability Program (MRP) has initiated a program to evaluate the impact of these aging effects on RV internals. How will potential AP1000 licensees use the results from the MRP RV internals program to ensure the integrity of the RV internals? (Section 4.5.2)

> *The response to this question is acceptable but the response needs to be reflected in the DCD.*

251.012

The application indicates that the reactor coolant pump (RCP) pressure housing will be made from SA 351 or SA 352 CF3A material and that RCP pressure boundary valve bodies may be castings of SA 351 CF3A. The application also indicates that CASS will not exceed a ferrite content of 30 FN (Ferrite Number). CASS RCP pressure boundary components are subject to thermal embrittlement. Please provide additional information discussing the impact of this aging effect on the integrity of these components along with a discussion of how this thermal embrittlement mechanism has been considered in the design and material selection for these components. Also, please discuss the need for potential licensees of the AP1000 plants to perform inspections to detect this aging effect. (Section 5.2.3)

> *The Westinghouse response to this question is acceptable subject to clarification of the method used to calculate the δ–ferrite. The calculated δ–ferrite should use Hull's equivalent factors or a method producing an equivalent level of accuracy (plus minus 6 percent deviation between measured and calculated values). These clarifications need to be included in the DCD.*

252.001

Recent NRC generic communications, including NRC Bulletins 2001-01, 2002-01, and 2002-02, have addressed issues related to the cracking of the vessel head penetration (VHP) nozzles and the differences in the AP1000 design compared to the current fleet of power water reactors (PWRs), including the following specific items:

a. geometry of the vessel head penetration (VHP) nozzle weld joint,

b. processes used for fabrication of the nozzle base material,

c. accessibility for inspection of the VHP nozzles and the reactor pressure vessel (RPV) head - describe any impediments or limitations in the AP1000 design,

d. materials used for both the nozzle base material and the welds, and

e. operating conditions, including the operating temperature of the RPV head, provisions for bypass flow to cool the head, etc. (Section 4.5.1)

The response stated, in part, that "(t)he AP1000 design has an integrated head package permanently attached to the reactor vessel head. This acts to reduce access to the top of the vessel head for inspection as compared to the current fleet of PWRs. However, the integrated head package has doors just above the vessel head that allow inspection access. Vessel head insulation configuration and access ports through this insulation allow for the implementation of visual inspection approaches across the vessel head."

a. Please clarify what is meant by "visual inspection approaches across the vessel head." Does the access allow for bare metal visual examination of the vessel head penetration to RPV junction at the top of the RPV as discussed in NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs."

b. Explain how the design permits for a visual examination of 360° around each reactor vessel head penetration.

c. Provide drawing/diagrams of the integrated head package showing access, shroud, insulation and penetrations. Discuss how the insulation is fixed and the extent to which it is removable. What is the minimum offset of the insulation from the surface of the RPV head?

d. In response to a prior RAI, you stated that the geometry of the AP1000 VHP nozzle weld joint is the same as in current Westinghouse PWRs. Would the changes Westinghouse described to the fabrication and installation process of the penetration nozzles reduce residual stresses or the effects of work hardening? Have there been any changes to the volume of weld metal, surface conditioning, etc. that could serve to reduce the residual stresses in the welds? Have any calculations been performed to compare the residual stresses in these welds to the current Westinghouse PWRs, and if so, how do the stresses compare for the AP1000 and current PWRs on the nozzle ID and OD (magnitudes and directions)?

e. *Relative to the inspections of present heads, interpretation of inspection findings have at times been complicated by a need to determine if an indication is service-induced cracking or an artifact from fabrication. What pre-service examinations will be applied to the VHP nozzles, i.e., will the VHP nozzles be subjected to a volumetric examination? Will the welds be examined using either surface or volumetric techniques, or both?*

f. *How was the head operating temperature of 560 °F determined, and has this been reviewed separately by the NRC?*

252.002

Paragraph 5.2.3.2.2 on page 5.2-11 in the 2nd paragraph discusses safe ends. What is the purpose of these safe ends? If the purpose of the safe ends is to protect the austenitic stainless steel from sensitization, then an A-8 weld, which is austenitic stainless steel, will become sensitized when the component postweld heat is treated at 1100 °F. Please address this concern as part of your response. (Section 5.2.3)

> *The response is not acceptable since A-8 welds include austenitic stainless steels such as 304 and 316 type materials. Westinghouse needs to address the concern of this RAI.*

281.001

Regulatory Guide (RG) 1.54, Revision 1, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," July 2000, defines the protective coatings-based service levels and the effect of coating failures on equipment during normal and post-accident conditions as delineated in the referenced American Society for Testing and Materials (ASTM) standards. The use of the terms "safety-related" and "non-safety-related" are not used in this revision to RG 1.54 to classify coatings. Please clarify which of the coatings listed in Table 6.1-2 meet the definitions of Service Levels I, II, and III. (Section 6.1)

> In Section 6.1.2.1.6 of the markup of DCD Section 6.1 attached to the response to this question, the title of RG 1.54, Revision 1, is listed as "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." The correct title of RG 1.54, Revision 1, is "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants." This change needs to be made before the actual DCD changes are made.

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