February 26, 2004

MEMORANDUM TO: Laura A. Dudes, Section Chief
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Office of Nuclear Reactor Regulation

FROM: Amy Cubbage, Project Manager ***/RA/***
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SUBJECT: JANUARY 13, 2004, AP1000 TELEPHONE CONFERENCE CALL SUMMARY

On Tuesday, January 13, 2004, a telephone conference call was held with Westinghouse Electric Company (Westinghouse) representatives and Nuclear Regulatory Commission (NRC) staff to discuss issues related to reactor coolant pressure boundary materials. The NRC staff specifically discussed open item (OI) 5.2.3-3, which was raised by the NRC in a letter dated September 3, 2003 (ML032330275). Westinghouse responded to this open item by letter dated September 8, 2003 (ML032530326). By e-mail sent to Westinghouse on October 28, 2003, (Attachment 1), the NRC staff requested additional information. A list of call participants is included in Attachment 2.

The following is a brief summary of the discussion regarding the staff's request for additional information concerning OI 5.2.3-3:

a) Westinghouse stated that the hydrogen concentration is 25 to 50 cm^3 per kg of water at standard temperature and pressure. This was provided by Westinghouse in the AP1000 design control document (DCD) table 5.2-2.

b) The NRC staff stated that the question had been broadened because Westinghouse indicated that this was not a pressurized thermal shock (PTS) issue. Westinghouse stated that they would provide a response to this question.

c) The NRC staff indicated that the DCD was not clear with respect to the material composition of the weld. Westinghouse stated that they would review the issue further.

d) The NRC staff requested additional detail to quantify the effects. Westinghouse stated that they would revise the OI response and either revise the DCD or provide a justification for not revising the DCD.

Docket No. 52-006

Attachments: As stated

d) The NRC staff requested additional detail to quantify the effects. Westinghouse stated that they would revise the OI response and either revise the DCD or provide a justification for not revising the DCD.

Docket No. 52-006

Attachments: As stated

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Staff Comments on Open Item 5.2.3-3
E-Mailed to Westinghouse on October 28, 2003

The staff has reviewed your September 8, 2003, response on lowered fracture toughness of Alloy 690/52/152 materials after exposure to hydrogenated water. For the staff to evaluate this information, the staff needs the following additional information.

a. Provide the H_2 concentration of the reactor coolant at normal plant operating conditions.

b. For a simulated Pressurized Thermal Shock transient, what temperature levels would the most susceptible bi-metallic welds reach? What bi-metallic welds in the reactor coolant system would experience the most significant cooldown effect?

c. Provide a schematic drawing of welds at these locations. For example, describe whether these welds would consist of Alloy 52 material through the entire wall, or Alloy 52 in contact with the reactor coolant and Alloy 82/182 for the remainder of the wall thickness.

d. Assume a small ID surface breaking flaw in the bi-metallic weld identified in parts b. and c. above. Evaluate what conditions and effects the flaw would see as a result of a simulated Pressurized Thermal Shock transient (i.e., hydrogen concentration, final temperature, loading rate, and failure potential).

Attachment 1

JANUARY 13, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

Nuclear Regulatory Commission	Westinghouse
J. Colaccino	R. Gold
M. Mitchell	R. Vijuk
T. Sullivan	W. Banford

Attachment 2

cc:

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