September 24, 2002

Mr. Michael M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects
Westinghouse Electric Company
Post Office Box 355
Pittsburgh, Pennsylvania 15230-0355

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 9 - AP1000 DESIGN CERTIFICATION REVIEW (TAC NO. MB4683)

Dear Mr. Corletti:

By letter dated March 28, 2002, Westinghouse Electric Company (Westinghouse) submitted its application for final design approval and standard design certification for the AP1000.

The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of your design certification application to ensure that the information is sufficiently complete to enable the NRC staff to reach a final conclusion on all safety questions associated with the design before the certification is granted.

The NRC staff has determined that additional information is necessary to continue the review. The topics covered in these requests for additional information (RAIs) include the areas of inservice inspection, component integrity, materials application, and chemical technology. These RAIs were sent to you via electronic mail on September 12, 2002. You agreed that Westinghouse would submit a response to these RAIs by December 2, 2002. Receipt of the information by December 2, 2002, will support the schedule documented in our letter dated July 12, 2002.

Enclosure 2 contains a history of previously-issued RAI correspondence.

If you have any questions or comments concerning this matter, you may contact me at (301) 415-3053 or ljb@nrc.gov.

Sincerely,

/RA/

Lawrence J. Burkhart, AP1000 Project Manager
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

Docket No. 52-006

Enclosure: As stated

cc: See next page

If you have any questions or comments concerning this matter, you may contact me at (301) 415-3053 or ljb@nrc.gov.

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OFFICE	NRLPO/PM	EMEB/BC	NRLPO/DD
NAME	LBurkhart:cn	BBateman	MGamberoni
DATE	9/20/02	9/20/02	9/24/02

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Request for Additional Information (RAI)
AP1000 Standard Design Certification
Series 250 - Inservice Inspection
Series 251 - Component Integrity
Series 252 - Materials Applicaton
Series 281 - Chemical Technology

AP1000 Design Control Document (DCD)

3.5.1.3 Turbine Missiles

251.001

It was stated in Section 10.2.2 that "(t)he rotor design, manufacturing, and material specification and the inspections recommended for the AP1000 provide an acceptable very low probability of missile generation." Section 10.2.2 further explains that "(t)he probability of destructive overspeed condition and missile generation, assuming the recommended inspection and test frequencies, is less than 1 x 10^{-5} per year."

Provide the source of the assessment of this low probability value (e.g., from WCAP-15783, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines"). List all modifications that Westinghouse has made to the Monte-Carlo simulation methodology since 1984, the year when an early version of the methodology described in WSTG-3-P, "Analysis of the Probability of a Nuclear Turbine Reaching Destructive Overspeed," was approved by the Nuclear Regulatory Commission (NRC). (DCD Section 3.5.1.3)

251.002

Address questions 2.a to 2.f if the modifications mentioned in question 251.001 affect the subject of each of these six questions/comments. (DCD Section 3.5.1.3)

a. Provide the probability distributions for both undetected and reported indications for the probabilistic burst and missile analysis and the bases for the selection. How are they used in the Monte-Carlo simulations?

b. Explain how stress corrosion crack growth and fatigue crack growth were considered in your turbine missile analysis. Assess the impact of crack growth due to low cycle fatigue (associated with turbine unit startups and shutdowns) on your missile probability analysis. Also, please demonstrate that the vibratory stresses of the turbine disks due to various excitations are negligible.

c. Was the stress corrosion crack growth rate independent of the level of stress intensity factor? Plot data from tests and operating plants to support your conclusion. If any variables related to stress corrosion crack initiation were included in your current turbine missile probability analysis, please provide detailed information regarding the use of the stress corrosion crack initiation parameters in your analysis.

d. Provide all random variables, their distributions, and suggested number of standard deviations that were used in your Monte-Carlo simulations. Explain the need for distributions other than the normal distribution for any of the variables, and justify the use of the suggested number of standard deviations for all variables. Comment on the convergence of the calculated P_1 value for your Monte-Carlo simulation involving this many random variables. Further, how were the mean values for these variables determined, especially for the mass of bladed disk fragments and fragments from other rotating parts? How do they correlate with industry experience on turbine missile events? Are values for these variables dependent upon the specific design or model of a turbine? Please use a typical turbine model to be used in the AP1000 application as an example and illustrate how these values were established. How was the degree of blade crushing, blade bending, and deformation of stationary blades considered in your calculation of the probability of casing penetration?

e. Assess the contributions due to these modifications to the turbine missile probability reported in the submittal.

251.003

How was the probability (1 x 10^{-7} per year) determined for the high-trajectory missile to impact safety-related areas of the AP1000? (DCD Section 3.5.1.3)

3.6.3 Leak-Before-Break (LBB) Evaluation Procedures

251.004

Due to the primary water stress corrosion cracking (PWSCC) of the V. C. Summer primary loop welds, the staff finds that the information we have today is substantially different from the information that was available when we approved leak-before-break (LBB) applications for existing pressurized water reactor (PWR) systems which contain Inconel 82/182 materials. The following three questions are related to staff concerns regarding this recently discovered degradation mechanism as it applies to any LBB-candidate piping system proposed in the AP1000. (DCD Section 3.6.3)

a. Section 5.2.3 of the DCD indicates that the “use of nickel-chromium-iron alloy in the reactor coolant pressure boundary is limited to Alloy 690. Alloy 600 may be used in limited areas for welding or buttering. Where Alloy 600 is used, it is not in contact with the reactor coolant.” However, in addition to the reactor coolant system (RCS) piping, there is LBB-candidate piping, for example the passive core cooling system (PCCS), exposed to primary water under temperature and pressure conditions similar to those in the RCS. Discuss the susceptibility of these systems to PWSCC.

b. Provide test and plant operational data regarding the crack growth rate for Alloy 52/152 welds to be used in contact with reactor coolant in the proposed lines for which LBB will be applied and demonstrate that this material is not susceptible to PWSCC.

c. LBB is based, in part, upon the premise that LBB will only be applied to piping materials that are not susceptible to any known degradation mechanisms. Until sufficient information is acquired to ensure that Inconel 52/152 materials are essentially “PWSCC resistant” through the anticipated 60-year operational lifetime of an AP1000 facility, the staff believes that augmented inservice inspection of Inconel welds in LBB lines, including the use of inside-diameter (ID) eddy current on a periodic basis, is an essential element for approval of the AP1000 “design” to support application of LBB. To facilitate resolution of the PWSCC issue for the AP1000, please provide an inspection plan that the combined licensee would be required to perform. This inspection plan should address additional inspection techniques (e.g., eddy current testing) to supplement ultrasonic testing (UT) so that tight flaws in piping welds similar to those detected in the V. C. Summer primary loop weld could be detected.

251.005

Provide crack morphology parameters, e.g., surface roughness, number of 45 degree and 90 degree turns, etc., that were used in generating the bounding analysis curves for LBB. To address the staff’s concerns resulting from recent experience with stress corrosion cracking in Inconel and stainless steel materials in PWR environments, please provide a comparative study on the most biased line from the LBB candidates using the crack morphology parameters for transgranular stress corrosion cracking. Information regarding crack morphology parameters for various degradation mechanisms is available in NUREG/CR-6443, “Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations.” Report the reduced margin on flaw size from this comparative study of the most biased line when the original bounding analysis curve (BAC) for this line is maintained. (DCD Appendix 3B)

251.006

NUREG-1512, “Final Safety Evaluation Report Related to Certification of the AP600 Standard Design,” September 1998, documents Westinghouse’s actions in resolving open items with regard to the AP600 review. These actions included (1) performance of fatigue crack growth analyses for the Class 2 and 3 piping systems selected for LBB applications, and (2) consideration of thermal stratification loads in three piping systems (pressurizer surge line, PRHR return line, and another line not identified) that Westinghouse identified to be susceptible to thermal stratification. Are these actions to be taken for AP1000 also? If not, please provide justification. (DCD Section 3.6.3)

<u>3.6.4 Combined License (COL) Information</u>

251.007

Section 3.6.4.2 states that “Combined License applicants referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves (BAC) documented in Appendix 3B.” The staff has concerns with this approach. Since piping satisfying all American Society of Mechanical Engineers (ASME) Code requirements on stresses could have a stress state that is outside the BAC for LBB, you need to establish a process to give the LBB BAC the same status as the ASME Code requirements on stresses to ensure a successful path for the design and construction of all LBB candidates proposed in the submittal. Please provide additional information addressing this issue. (DCD Section 3.6.4)

Questions related to the BACs

Appendix 3B.3.1.3, 3B.3.1.4, and 3B.3.1.5

251.008

Using Figure 3B-12 as an example, provide flow stress and the ASME Code specified S_m value for the material. Flow stress can be defined as one-half of the ultimate strength and yield strength, or $3S_m$ of a material. Justify your choice if your selection gives a higher flow stress for the piping material. Provide the axial stress, bending stress, leakage flaw size, and critical flaw size for the normal stress state and the maximum stress state corresponding to the low normal stress case (Case 1). Provide similar information for the high normal stress case (Case 2). (DCD Appendix 3B)

251.009

The high normal stress case was determined using flow stress as the bending stress. In some figures, for instance Figure 3B-21, a normal stress of 30 ksi (thousand pounds-per-square inch) would correspond to more than two times the flow stress of the material. Even greater multiples of flow stress are expected for the maximum stress of 40 ksi. What is the meaning of the region to the right of Point “B (the point corresponding to Case 2)” for all BACs in terms of the piping design ASME Code criteria? For each BAC shown in Figures 3B-1 to 3B-21, construct a separate design curve based on the appropriate piping design ASME Code such that every point within the design curve would automatically satisfy all ASME Code requirements on piping stresses. If any of the design curves exceed its corresponding BAC by 25%, provide detailed piping stress and LBB analyses for that line to demonstrate that it is feasible to build a line according to a more restrictive piping design criteria considering the LBB BAC. This additional work needs to be performed for lines other than those lines that have been approved for LBB applications for operating plants with essentially the same analysis parameters (pipe diameter, wall thickness, material properties, and loading conditions) and for the five exemplary lines studied in the AP600. (DCD Appendix 3B)

251.010

Since it is unlikely that the relationship between the maximum stress and the normal stress shown in Figures 3B-1 to 3B-21 is linear, an intermediate point should be plotted on all of these curves. Please provide revised figures. (DCD Appendix 3B)

Section 4.5.1 Control Rod and Drive System Structural Materials

252.001

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

a. geometry of the VHP nozzle weld joint,

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

c. accessibility for inspection of the VHP nozzles and the 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)

Section 4.5.2 Reactor Internal and Core Support Materials

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 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)

Section 5.2.3 Reactor Coolant Pressure Boundary Materials

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)

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 ^{o}F. Please address this concern as part of your response. (Section 5.2.3)

252.003

Paragraph 5.2.3.3.2 on page 5.2-13 (2nd paragraph) discusses welding material control. Storage and handling of welding materials is also covered in NB-4400. Should this Code paragraph also be referenced? (Section 5.2.3)

252.004

Paragraph 5.2.3.4.6 on page 5.2-16 (last phrase of the 4th paragraph) refers to using welding material that is fully austenitic. This phrase could imply that if a material such as 308, 309, or 310 was not purchased with the minimum amount of ferrite required, it could still be used if it were fully austenitic. Please state the exact materials that you wish to exempt from the delta ferrite requirement. List the exact materials that are considered to be fully austenitic welding materials. (Section 5.2.3)

252.005

Regulatory Guide (RG) 1.71 pertains to welder qualification for areas of limited accessibility. Westinghouse's exception to this RG states: "The performance of required nondestructive evaluations helps to confirm weld quality. Limited accessibility qualification or requalification in excess of ASME Code, Section III or IX requirements is considered an unduly restrictive requirement for component fabrication, where the welders' physical position relative to the welds is controlled and does not present significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised."

With respect to welds for which a surface examination is the only examination method used, the positions in RG 1.71 are necessary to keep an unacceptable weld from getting into service. If a welder has limited accessability and/or visibility, the welder can produce a weld with many types of defects, such as lack of side wall fusion, lack of root penetration, excessive porosity, and slag inclusions; without a volumetric examination, these types of defects cannot be identified. With limited accessability and/or visibility, it is possible that an adequate final visual examination of the weld will not/cannot be achieved. If a welder is qualified to make a weld in an area of limited accessibility, the welder should be able to make similar limited accessibility welds in a shop or in the field. For welds that are not volumetrically examined, how does Westinghouse intend to ensure that welds made in areas of limited accessability and/or visibility will meet the fabrication requirements of ASME Code Section III? (DCD Appendix 1A)

Section 5.2.4 Inservice Inspection and Testing of Class 1 Components

250.001

The ASME Code Section XI, 1999 Addenda, eliminated the pressurizer and steam generator (SG) nozzle inside-radius inspections in Table IWB-2500-1, Examination Category B-D, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B). The staff disagrees with this code change, which has not yet been endorsed in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.55a. Verify that the AP1000 pressurizer and SGs will be designed to permit inside-radius inspection per the provisions of Table IWB-2500-1, Examination Category B-D, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition or to permit a visual examination, such as a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in the 1998 Edition, Table IWB-3512-1 in place of a UT examination. (Section 5.2.4)

250.002

The ASME Code Section XI, 2000 Addenda, Table IWB-2500-1, Examination Category B-K, Item B10.10, pressure vessel welded attachments, permits the performance of single-side surface examination in place of earlier Section XI requirements for a surface examination from both sides of the weld or permits the performance of a single-side volumetric examination of the weld in place of surface examination of the inaccessible surface if surface examination from both sides of the weld is not performed. Because little useful information will result from single-side surface examination of pressure vessel welded attachments, the staff disagrees with these reduced requirements in the 2000 Addenda. Verify that the AP1000 vessel welded attachments will permit either a surface examination from both sides of the weld or a single-side volumetric examination of the weld in place of surface examination of the inaccessible surface if surface examination from both sides of the weld is not performed. (Section 5.2.4)

5.3.2 Reactor Vessel Materials

251.013

Because the temperature at which a ferritic material is irradiated affects the material's response to irradiation (i.e, the "shift" in charpy 30 ft-lb transition temperature), please provide information regarding the reactor pressure vessel (RPV) wall temperature during 100 percent power operation. The staff has, in previous applications, assumed that the RPV wall temperature is the same as the cold leg temperature. If a plant will operate at a cold leg temperature below 274°C (525°F), discuss the effects of temperature on embrittlement. A similar question was asked for the AP600 review (AP600 RAI 252.84). The staff is requesting that a similar response be provided and incorporated in the AP1000 DCD. (Section 5.3.2)

Note: AP600 RAI 252.84 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 8, 1993 (NUDOCS Accession No. 9301130165).

251.014

Describe the lead factors for surveillance capsules. This question was asked for the AP600 review (AP600 RAI 252.96). The staff is requesting that a similar commitment be made in the AP1000 DCD regarding an analysis that will be performed for the COL application with the capsule/holder modeled in order to more accurately define the surveillance capsule lead factors and azimuthal locations. (Section 5.3.2)

Note: AP600 RAI 252.96 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

251.015

Verify or discuss the design considerations for the AP1000 that facilitate in-place reactor vessel thermal annealing treatment should it become necessary. This question was asked for the AP600 review (AP600 RAI 252.102). The staff is requesting that a similar response be provided and incorporated into the AP1000 DCD. (Section 5.3.2)

Note: AP600 RAI 252.102 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

5.3.3 Pressure-Temperature Limits

251.016

Section 5.3.3.1 of the DCD indicates that the results of the material surveillance program will be used for the development of heatup and cooldown curves. Verify that the material surveillance program data that will be used for recalculating these curves is the plant specific data obtained by each COL. (Section 5.3.3)

251.017

Provide the details for the pressure-temperature limit calculations, including assumptions and margins. Identify any deviations from the recommended calculational procedures in Section 5.3.2 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis reports for Nuclear Power Plants.” This question was asked for the AP600 review (AP600 RAI 252.105). The staff is requesting that a similar response be provided and incorporated into the AP1000 DCD (Section 5.3.3)

Note: AP600 RAI 252.105 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

251.018

Demonstrate that the pressure-temperature limits are in accordance with Appendix G to 10 CFR Part 50. For example, verify that the limit for the closure flange is satisfied. This question was asked for the AP600 review (AP600 RAI 252.106). The staff is requesting that a similar response be provided and incorporated into the AP1000 DCD. (Section 5.3.3)

Note: AP600 RAI 252.106 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI on January 14, 1993 (NUDOCS Accession No. 9301250260).

5.3.4 Reactor Vessel Integrity

251.019

Provide the fluence value that was used in calculating the RT_{PTS} for end-of-life. (Section 5.3.4)

5.4.1.3.6.3 Reactor Coolant Pump Flywheel Integrity

251.020

Provide a basis for not providing inspections, test, analyses, and acceptance criteria (ITAAC) related to the RCP flywheel fatigue analysis in Table 2.1.2-4. (Section 5.4.1)

251.021

In the AP600 review, RAIs 251.2 through 251.23 pertain to RCP flywheel integrity. In addition, WCAPs-13734 and 13735, "Structural Analysis Summary for the AP600 Reactor Coolant Pump Flywheel," were submitted as supplemental information for the revised response to question 251.11. Confirm that these responses and the WCAPs are applicable to the AP1000 application as it pertains to RCP flywheel integrity. Should aspects of these responses or reports not be applicable, provide updated information to address the AP600 RAIs as applicable to AP1000 RCP flywheel integrity. (Section 5.4.1)

Note: AP600 RAIs 251.2 through 251.23 were issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its responses to these RAIs in letters dated January 14, May 24, and May 28, 1993 (NUDOCS Accession Nos. 9301250260, 9306020387, and 9306020220, respectively).

5.4.2.4 Steam Generator Materials

252.006

Section 5.4.2.4.2 indicates that tubes can be supported by either an open lattice design called eggcrates, or by a support plate design. The seventh paragraph of section 5.4.2.3.3 discusses tube supports only in terms of broached hole support plate design. Please clarify. (Section 5.4.2)

251.022

For the AP600 design, the response to RAI 252.110 indicated that the results of prototype tests and calculations were not yet completed with respect to the subject of flow-induced vibrations of the SGs with special emphasis on fluid elastic vibration (AP600 Tier 2 DCD/Standard Safety Analysis Report (SSAR) Section 5.4.2.3.3). Please provide the results from the AP600 tests and calculations, if these are applicable to the AP1000 design. If the AP600 results are not applicable to the AP1000 design, please provide the results of the AP1000 prototype tests and calculations related to flow-induced vibrations of the tubes in different locations of the bundle. In addition, please discuss in more detail than in section 5.4.2.3.3, the criteria for establishing the instability threshold for ensuring that the fluid-elastic behavior does not contribute unacceptably to flow-induced vibration or alternating stresses. (Section 5.4.2)

Note: AP600 RAI 252.110 was issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its response to this RAI in a letter dated January 14, 1993 (NUDOCS Accession No. 9301250260).

252.007

Section 5.4.2.4.1 of the AP600 SSAR (DCD Tier 2) was revised in response to a staff question on archival material to indicate that a minimum of seven feet of tubing in the final heat treat condition is supplied. This information was deleted from the AP1000 DCD. Please address the standard or criteria that will be used to specify minimal tube archive requirements. (Section 5.4.2)

5.4.2.5 Steam Generator Inservice Inspection

250.003

The requirements for the SG Tube Surveillance Program are contained in technical specification (TS) 5.5.5. This TS specifies that "SG tube sample size selection, sample size expansion, inspection results classification criteria, tube inspection frequency, plugging and repair limits, and specific definitions and limits be in accordance with [RG 1.83, Revision [ ], date]." (Square bracketed information is to be defined when TSs are determined for the COL applicant.) The most recent revision of RG 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" is Revision 1, dated July 1975. Specifying TS inspection requirements to be in accordance with this RG is inappropriate since the RG contains guidance and not requirements, i.e., recommended surveillances are written in terms of actions that should be taken. The guidance in this RG was superceded by the SG TSs in various documents, the most recent being NUREG-0452, Revision 4, "Standard Technical Specifications (STS) for Westinghouse Pressurized Water Reactors." The TS surveillance requirements for all domestic SGs are very similar, if not identical, to those in NUREG-0452, Revision 4, and under the Improved Standard Technical Specification (ISTS) program these surveillance requirements are unchanged. These requirements contain some essential surveillance requirements missing from RG 1.83, Revision 1, such as definitive sample expansion criteria. In addition, TS Section 5.6.8, "SG Tube Inspection Report," refers to condition C-3 for submitting certain reports; C-3 is not defined in RG 1.83 or elsewhere in the TSs although it is defined in the STS. Please revise the SG Tube Surveillance Program TSs to be consistent with the surveillance requirements contained in STS in NUREG-0452, Revision 4. (Section 5.4.2)

6.1 Engineered Safety Features Materials

252.008

Section 6.1.1.1 of the DCD states that “(t)he use of nickel-chromium-iron alloy in the engineered safety features is limited to Alloy 690.” It then goes on to state that “Alloy 600 may be used for welding or buttering.” Under what situations would Alloy 600 be used for welding or buttering? Is Alloy 600 ever in contact with primary water, for example in the passive containment cooling system (PCCS) which may also be exposed to pressure and temperature conditions similar to those in the RCS? (Section 6.1)

6.1.2.1 Protective Coatings

281.001

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)

6.2.7 Fracture Prevention of Containment Pressure Boundary

252.009

Section 3.8.2.2 indicates that the containment pressure vessel shell material is SA738, Grade B. This material has been approved by ASME Code Case N-655 which approves SA-738, Grade B, material for the construction of containment vessels. The staff finds this application of SA-738, Grade B, material acceptable subject to the following two conditions:

a. Westinghouse needs to specify in its purchase specifications that SA-738 Supplementary Requirement S17, Vacuum Carbon-Deoxidized Steel, applies to this material, and

b. SA-738 Supplementary Requirement S20, Maximum Carbon Equivalent for Weldability, also applies to the material.

These two requirements are needed to ensure adequate materials properties and weldability of the containment vessel material. SA-738, Grade B, material is exempt from postweld stress relief heat treatment up to 1.75 inches of thickness. The AP1000 containment vessel is 1.75 inches thick. That means that the welds will not be stress relieved and, therefore, higher residual stresses will be present in the welds. Also, the material will most likely be procured in the quenched and tempered condition. Welding will reduce the impact properties of the material in the heat affected zone. Requiring vacuum-degassed steel will ensure adequate material properties. Requiring a carbon equivalent weldability check will ensure that the steel is readily weldable because the residual elements of the steel will be more tightly controlled.

Westinghouse will need to include these two conditions in an updated revision to the DCD. (Section 6.2)

9.2.3 Demineralized Water Treatment System (DTS)

281.002

High concentrations of halogens and sulfates present in the system can accelerate the corrosion of components in the DTS. Please provide the maximum allowable concentrations of halogens and sulfates present in the system. (Section 9.2.3)

10.2.3 Turbine Rotor Integrity

10.2.3.1 Materials Selection

251.023

Provide a definition of fracture appearance transition temperature (FATT) and discuss the relationship between FATT and nil-ductility transition (NDT) temperature and reference nil-ductility temperature (RT_{NDT}). (Section 10.2.3)

10.2.3.2 Fracture Toughness

251.024

The second paragraph in Section 10.2.3.2 contains the statement, "(t)he ratio of material fracture toughness, K_{IC} (as derived from material tests on each rotor) to the maximum tangential stress for rotors at speeds from normal to design overspeed, will be at least 200 ksi x $\sqrt{in}$ (or at least 2) at minimum operating temperature." This sentence is not clear and should be revised. Confirm that you are trying to suggest that fracture toughness will be at least 200 ksi x $\sqrt{in}$ and the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed will be at least 2. (Section 10.2.3)

251.025

It was mentioned in the third paragraph of Section 10.2.3.2 that conservative factors of safety are included for the size uncertainty of potential or reported ultrasonic indications, rate of flaw growth, and the duty cycle stresses and number. Provide these factors of safety, and comment on how they are determined. (Section 10.2.3)

10.2.3.2.1 Brittle Fracture Analysis

251.026

It was mentioned in the first paragraph of Section 10.2.3.2.1 that the maximum rotor stress is determined from rotation, steady-state thermal loads, and transient thermal loads from startup and load change. Provide the operating speed and the first and second critical speeds for the rotor. If any of the rotor critical speeds are below the operating speed, explain why you do not need to consider rotor vibratory stresses when passing through critical speeds during startups and shutdowns. (Section 10.2.3)

251.027

Provide the K_{IC} value and the factor of safety that was used to generate the allowable initial flaw area from an initial flaw area. Discuss the appropriateness of the assumption that a crack would originate from the centerline for rotors without bores. (Section 10.2.3)

251.028

It was stated in the last paragraph of Section 10.2.3.2.1 that there is not a separate material toughness (K_{IC}) requirement for AP1000 rotors. Not having a K_{IC} requirement for the deterministic brittle fracture mechanics analysis is not appropriate. In the AP600 review, the staff accepted the use of the Rolfe-Novak-Barsom correlation of upper shelf Charpy values with K_{IC} in the turbine missile probability analysis. That was because for a missile probability analysis involving more than twenty random variables, the impact of the variability of K_{IC} on the final results is small. It was never the staff's intention to accept the Rolfe-Novak-Barsom correlation for a deterministic brittle fracture mechanics analysis on any components without sufficient safety margin (say 30%) to account for the uncertainty in using this empirical formula. Provide a K_{IC} requirement for AP1000 rotors. (Section 10.2.3)

<u>10.2.3.6 Maintenance and Inspection Program Plan</u>

251.029

It was mentioned in Section 10.2.3.6 that the maintenance and inspection program plan for the turbine assembly and valves is based on turbine missile probability calculations reported in WCAP-15783, operating experience of similar equipment, and inspection results. Provide the calculated turbine missile probability results that were used for this purpose and explain how they were used to determine the inspection intervals of 10 years for low-pressure (LP) turbines and 8 years for high-pressure (HP) turbines, the inspection intervals of 3 years for a variety of valves, and the quarterly testing frequency for valves. (Section 10.2.3)

<u>10.4.8 Steam Generator Blowdown System</u>

281.003

Please provide the bases for the values governing the blowdown flowrates, i.e., the minimum and maximum percentage values of the maximum steaming rate. (Section 10.4.8)

HISTORY OF PREVIOUSLY-ISSUED
REQUESTS FOR ADDITIONAL INFORMATION

Letter No.	Date issued	ADAMS Accession No.	RAI Nos.	Date of response	ADAMS Accession No.
1	6/26/2002	ML021780568	440.001 - 440.008	7/24/2002	ML022110430
2	8/16/2002	ML022280379	720.001	9/10/2002	
3	8/27/2002	ML022390103	420.001 - 420.046, 435.001 - 435.015		
4	9/3/2002	ML022460356	620.001 - 620.043		
5	9/4/2002	ML022470255	210.001 - 210.057		
6	9/5/02 Reissued 9/18/02	ML022480440, Reissued RAI ML022610042	440.009 - 440.148, 720.002 - 720.026		
7	9/19/02	ML022620026	260.001 - 260.003, 261.001 - 261.010, 471.001, 471.010, 472.001, 472.003		
8	9/19/02	ML022620319	220.001 - 220.019, 230.001 - 230.019, 240.001 - 240.004, 241.001 - 241.003		

Enclosure 2

cc:

Mr. W. Edward Cummins
Advanced Plant Safety & Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

Mr. Michael Corletti
Advanced Plant Safety & Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

Mr. H. A. Sepp
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230

Lynn Connor
Doc-Search Associates
2211 sw 1ST Ave - #1502
Portland, OR 97201

Barton Z. Cowan, Esq.
Eckert Seamans Cherin & Mellott, LLC
600 Grant Street 44th Floor
Pittsburgh, PA 15219

Mr. Ed Rodwell, Manager
Advanced Nuclear Plants' Systems
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94304-1395

Charles Brinkman, Director
Washington Operations
Westinghouse Electric Company
12300 Twinbrook Parkway, Suite 330
Rockville, MD 20852

Mr. R. Simard
Nuclear Energy Institute
1776 I Street NW
Suite 400
Washington, DC 20006

Mr. Thomas P. Miller
U.S. Department of Energy
Headquarters - Germantown
19901 Germantown Road
Germantown, MD 20874-1290

Mr. David Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1707 H Street NW, Suite 600
Washington, DC 20006-3919

Mr. Paul Gunter
Nuclear Information & Resource Service
1424 16th Street, NW., Suite 404
Washington, DC 20036

Ms. Wenonah Hauter
Public Citizen's Critical Mass Energy
  Project
215 Pennsylvania Avenue, SE
Washington, DC 20003

Mr. Tom Clements
6703 Guide Avenue
Takoma Park, MD 20912

Mr. James Riccio
Greenpeace
702 H Street, NW, Suite 300
Washington, DC 20001

Hugh Jackson, Policy Analyst
Public Citizen's Critical Mass Energy
  and Environment Program
1724 Duarte Drive
Henderson, NV 89014

Mr. James F. Mallay, Director
Regulatory Affairs
FRAMATOME, ANP
3315 Old Forest Road
Lynchburg, VA 24501

Mr. Ed Wallace, General Manager
Project Management
Lake Buena Vista Bldg., 3rd Floor
1267 Gordon Hood Avenue
Centurion 0046
Republic of South Africa
PO Box 9396 Centurion 0046

Mr. Vince Langman
Licensing Manager
AECL Technologies, Inc.
901 15th Street, NW., Suite 440
Washington, DC 20005-2301

Mr. Glenn R. George
PA Consulting Group
Chrysler Building, 34^{th} Floor
405 Lexington Avenue
New York, NY 10174

Mr. Gary Wright, Manager
Office of Nuclear Facility Safety
Illinois Department of Nuclear Safety
1035 Outer Park Drive
Springfield, IL 62704