September 4, 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. 5 - AP1000 DESIGN CERTIFICATION REVIEW (TAC NO. MB4683)

Dear Mr. Cummins:

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. Enclosure 1 contains requests for additional information (RAIs) regarding mechanical engineering design issues for the AP1000 design including reactor vessel internals design basis and dynamic analysis and piping design acceptance criteria (DAC) implementation. These RAIs were sent to you via electronic mail on August 19 and August 20, 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.

The RAIs on the implementation of piping DAC for the AP1000 will be the subject of a meeting scheduled for Monday, September 9, 2002, through Wednesday, September 11, 2002, at Westinghouse Energy Center in Monroeville, PA.

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.

Sincerely,

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Request for Additional Information
AP1000 Standard Plant Design
Series 210 - Mechanical Engineering

Review Subject: Reactor Vessel Internals Design Basis and Dynamic Analysis

Reference: AP1000 Design Control Document (DCD), Revision 0, dated January 2002.

210.001

Reference, Volume 6, Section 3.9.2.3, Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions, Page 3.9-31, last paragraph (¶):

Westinghouse (W) proposes that the assessment of reactor pressure vessel (RPV) internals flow-induced vibrational response is done using a combination of analysis and testing as specified in Regulatory Guide (RG) 1.20. However, W also proposes that the entire vibration assessment program, including the predictive analysis portion, will be performed by the Combined License (COL) applicant. This proposal is repeated in DCD Section 3.9.8.1 (Volume 6, Page 3.9-93) citing consistency with RG 1.20 as a basis for deferral of the performance of the entire vibration assessment program to the COL applicant.

The NRC staff is not in complete agreement with this proposal for the following reasons: Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52.47(a)(2) requires that applications for standard design certification must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design before the certification is granted. Delaying the predictive analysis portion of the vibration assessment program to the COL applicant stage of plant construction does not provide the staff with a level of design information sufficient to reach a final conclusion regarding adequacy of the RPV internals design. Conformance with RG 1.20 alone does not necessarily fulfill the requirements of 10 CFR 52.47(a)(2) for certification of the adequacy of the standard design of the RPV internals, primarily because the RG 1.20 scheduling requirements for the submittal of analytical results to the staff occurs much too late to support the standard design certification process.

The staff's position on this issue is that the detailed predictive analysis portion of the RPV internals flow-induced vibration analysis program should be provided for staff review during the design certification process, and should not be deferred to the COL applicant stage of actual plant construction. It is recognized that the other phases of the comprehensive RG 1.20 vibration assessment program, i.e., vibration measurement and physical inspection, must be done later by the COL applicant to confirm the predictive analysis results. However, the staff considers the results of the predictive analysis phase of this program to be the kind of detailed information necessary for the staff to make a determination of adequacy of the AP1000 RPV internals design for purposes of final design certification.

Please provide technical documentation of the predictive analysis phase of the vibration assessment program. The technical details should be provided with descriptions of the analytical methods used including computer models, results of the analyses summarized in tabular format, and comparisons of calculated stresses to the American Society of Mechanical Engineers Code (ASME Code) allowables for the major components of the RPV internals design. An example of the presentation of the type of analytical data requested is W topical report WCAP-14761, “AP600 Reactor Internals Flow-Induced Vibration Assessment Program.” This type of topical report would also be appropriate for presentation of key details of the AP1000 prototype RPV internals design necessary for staff review at the standard design certification stage.

210.002

Reference, Volume 6, Section 3.9.2.3, Page 3.9-33, second ¶: The DCD refers to vibration test results of other reactor internals designs (Doel 3, Doel 4, and Paluel 1) that will be utilized to perform the AP1000 flow-induced vibration assessment program, proposing that this will all be done at some future time.

Please include these existing vibration test results in comparison to the results of the predictive analysis requested in RAI 210.001 above.

210.003

Reference, Volume 6, Section 3.9.2.3, Page 3.9-33: The discussion in the lower half of this page (beginning with the fourth ¶) highlights specific differences in the AP1000 RPV internals design compared to previous W reactor designs.

Please provide technical data which demonstrates that, in fact, these AP1000 design differences have been evaluated analytically and result in a design configuration for the RPV internals with acceptable vibration levels when compared to appropriate allowable values for component stress and deflection.

210.004

Reference, Volume 6, Section 3.9.2.3, Pages 3.9-34 and -35: These pages include discussions which are informative and useful for introductory purposes, but are highly speculative in terms of demonstrating adequacy of the RPV internals design. A technical safety evaluation for purposes of approving/certifying the AP1000 internals design cannot be based solely on these types of general expectations. This further emphasizes the need for a topical report similar to WCAP-14761 (see RAI 210.001 above).

Please provide additional discussion, including analytical results compared to appropriate allowable values, which can be used to develop conclusions of adequacy based on an analytical process.

210.005

Reference, Volume 6, Section 3.9.2.3, Page 3.9-34, fifth ¶: The third sentence of this paragraph includes the term “the key elevation.” Please further explain the meaning of this term in the context of the discussion of inlet nozzle coolant velocity, and the resulting effects on core barrel response to the fluid forcing function.

210.006

Reference, Volume 6, Section 3.9.2.3, Page 3.9-35, first ¶: This paragraph is an example of the kind of general expectations/conclusions that require analytical verification for purposes of design certification. A recognized specific difference in the AP1000 lower internals design produces a lower natural frequency of the internals assembly, resulting in higher estimated amplitudes of vibration. The stated expectation is that these higher displacements of the internals assembly will be acceptable, but no further technical justification for this expectation is provided. The discussion which follows suggests that the internals vibration frequencies and amplitudes will be accurately determined based on the instrumentation measurements during pre-operational testing of the first plant (but this obviously will not occur until the plant has been built).

Please provide results of a predictive analysis to conservatively quantify the frequencies and displacements of the internals assembly resulting from enveloping estimates of forcing functions due to operational flow transients. The resulting component stresses and deflections should be compared to applicable allowable values to justify any conclusions of adequacy of the AP1000 RPV internals standard design.

210.007

Reference, Volume 6, Section 3.9.2.3, Page 3.9-35, fifth ¶: This paragraph indicates that the AP1000 reactor coolant pumps (RCPs), although lower in rated horsepower, operate at a higher rotational speed than RCPs in previous plant designs. Please include the effects of the RCP flow-induced vibration in the predictive analysis requested in RAI 210.006 above.

210.008

Reference, Volume 6, Section 3.9.2.4, “Pre-operational Flow-Induced Vibration Testing of Reactor Internals,” Page 3.9-35, seventh ¶: The three aspects, or phases, of a RG 1.20 pre-operational vibration assessment test program are appropriately identified. However, the first phase, i.e., a prediction of the vibrations of the reactor internals, although specifically identified, is never really presented in detail anywhere in the following discussion of the overall program. Please provide additional description of the vibration analysis phase of the program, including results of the predictive analysis requested in RAI 210.001 above. Alternatively, provide a reference to the topical report (similar to WCAP-14761) which contains the RPV internals vibration analysis for the AP1000 prototype design.

210.009

Reference, Volume 6, Section 3.9.2.4, Page 3.9-36, last ¶: Please provide clarification of the following statement regarding the visual inspection of RPV internals before and after hot functional testing:

“This inspection is performed on AP1000 plants subsequent to the first.”

Does this imply that the first plant built is excluded from these inspections? If so, why would the prototype plant be excluded from inspection?

210.010

Reference, Volume 6, Section 3.9.2.4, Page 3.9-38, second ¶: This is a continuation of the DCD approach that verification of acceptability of expected RPV internals vibration levels can be deferred to the pre-operational/hot functional test phase of plant construction. While the final verification of acceptability may, in fact, be most specifically demonstrated at that time by the use of actual instrumented test data, the deferral of predictive analysis is not compatible with the need for this type of technical data for the design certification process. 'Expected' vibration levels have to be quantified at some point in the process. The discussion in DCD Section 3.9.2.5.1.2 indicates that the AP1000 RPV internals are represented in detailed analytical models which can be used to analyze the dynamic characteristics of the internals response to various hydraulic forcing functions. The staff considers the analytical results of this type of predictive analysis to be the kind of detailed information necessary for the staff to make a determination of adequacy of the AP1000 RPV internals design for purposes of final design certification.

Please provide analytical results of the predictive analysis phase of the RG 1.20 vibration assessment program for staff review. See also RAI 210.001 and RAI 210.006.

210.011

Reference, Volume 6, Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions," Pages 3.9-38 and -39: The application of leak-before-break (LBB) criteria is discussed as a means of defining those pipe breaks which are used in the faulted condition analysis of the reactor internals, but the specific breaks postulated for the design-basis analysis are not specified.

Please identify those pipe breaks which, after application of LBB criteria, do not qualify for elimination of postulation and post-rupture dynamic analysis. Also, further identify those postulated pipe breaks analyzed to determine the maximum dynamic response of the AP1000 RPV internals.

210.012

Reference, Volume 6, Section 3.9.2.5, Page 3.9-41, last sentence of Section 3.9.2.5.2: This section describes the analytical methods used to calculate stresses and deflections in the RPV internals due to the combined loads from postulated pipe rupture and the safe shutdown earthquake (SSE). The last sentence in this section states the final conclusion that the reactor internals components are within acceptable stress and deflection limits. This significant conclusion is stated without providing, or referencing, any supporting stress and deflection data from the actual analyses (which presumably have been done in order to reach this conclusion).

Please provide a results summary of analytical data, including comparison to appropriate allowable values, which demonstrate that stress, deflection, and stability criteria for the RPV internals design have been met when subjected to the combined effects of the limiting postulated pipe break, and the SSE.

210.013

Reference, Volume 6, Section 3.9.2.6, “Correlation of Reactors Internals Vibration Tests with the Analytical Results,” Page 3.9-41, first ¶: The results of dynamic analysis of reactor internals vibration (used for comparison to test results) are generally mentioned, but it is not clearly defined which reactor design was used to generate the analytical results. Please provide additional information for the reactor design used to obtain the dynamic analysis results, the reference plants providing pre-operational vibration testing data, and an evaluation of the applicability of the data comparisons to validation of the AP1000 prototype design.

210.014

Reference, Volume 6, Section 3.9.2.6, Page 3.9-41, first ¶: A conclusion of adequacy regarding the analytical model is stated without providing, or referencing, any analytical results to demonstrate adequacy. Please provide a summary of results to justify conclusions which would establish the analytical model as a benchmark for future analyses.

210.015

Reference, Volume 6, Section 3.9.5.1.1, “Lower Core Support Assembly,” Page 3.9-77, second ¶: The discussion states that in this design the core barrel is modeled as a beam. Please provide additional description of the core barrel design analysis, addressing in particular, the support type specified as the design basis, i.e., plate-and-shell type, or linear type as defined in ASME Boiler & Pressure Vessel Code, Section III (ASME III), Subsection NF, and the stability criteria used in addition to allowable stress criteria. Additionally, for other RPV internal structures subjected to compressive loading, e.g., support columns, specify the stress and stability criteria used for the design analysis.

210.016

Reference, Volume 6, Section 3.9.5.1.1, Page 3.9-77, fourth ¶: The discussion mentions energy absorbing devices which limit dynamic forces imposed on the RPV, and also limit displacement of the core. Please provide additional description of the energy absorbing devices, and an explanation of how they can function to limit applied forces (presumably by allowing large displacement), while also limiting displacements of the reactor core to maintain alignment of the core to facilitate control rod insertion.

210.017

Reference, Volume 6, Section 3.9.5.2, “Design Loading Conditions,” Page 3.9-79: Please indicate how the potential effects of thermal stratification in the RPV and in attached piping are accounted for in the design of the RPV internals.

210.018

Reference, Volume 6, Section 3.9.5.2.3, “Level D Service Conditions,” Page 3.9-79: Please define the specific pipe breaks which cannot be excluded from dynamic analysis as a result of application of mechanistic pipe break LBB criteria.

210.019

Reference, Volume 6, Section 3.9.5.2.3, Page 3.9-79: Please identify the limiting, worst case pipe break used for the faulted condition design basis analysis of the AP1000 RPV internals prototype design. Also provide a summary of analysis results which demonstrate that the appropriate faulted condition allowable stress criteria have been satisfied. (See RAI 210.011)

210.020

Reference, Volume 6, Section 3.9.5.3, “Design Bases,” Page 3.9-80: Please define which components of the RPV internals assembly are designated as ASME III Class CS core support structures.

210.021

Reference, Volume 6, Section 3.9.5.3, Page 3.9-80: For those components designated ASME III Class CS core support structures, please provide a specific commitment that they are designed, fabricated, and examined in accordance with the requirements of ASME III, Subsection NG, Core Support Structures. Also for Class CS core support structures, provide a statement that the design documentation for these components includes a certified Design Specification and a certified Design Report conforming to the requirements of ASME III, Subsection NCA.

For those RPV internals components not designated ASME III Class CS core support structures, please identify the design basis used for design, construction, and examination.

210.022

Reference, Volume 6, Section 3.9.5.3.1, Mechanical Design Bases, Page 3.9-80: The fourth bullet states that, “The core internals are designed to withstand mechanical loads arising from the SSE and to meet the requirements of the following item.” The term “following item” is not clearly defined any further. Please provide additional explanation of what is meant by, “requirements of the following item.”

210.023

Reference, Volume 6, Table 3.9-14: The component terms “upper barrel” and “upper package” are listed in this table for maximum allowable deflections. The physical locations and functions of these components are not clear from the arrangement drawings and discussions provided elsewhere in the DCD. Please provide additional arrangement details, e.g., in Figure 3.9-8, “Reactor Internals Interface Arrangement,” for clarification.

210.024

Reference, Volume 6, Table 3.9-14: The maximum allowable deflection for the secondary core supports is not specified in Table 3.9-14 (see discussion of energy-absorbing devices in Section 3.9.5.1.1, “Lower Core Support Assembly”).

210.025

Reference, Volume 6, Section 3.9.5.3.2, “Allowable Deflections,” Page 3.9-81: The last ¶ of this section presents a brief description of the function of the secondary core supports assuming failure of the primary core supports. The stated assumption of complete and instantaneous failure of the primary core support structure appears to be conservative in terms of maximizing the energy imparted to the secondary core supports due to a vertical drop of the core. This assumption would appear to result in a complete vertical drop with no off-axis rotation of the core producing uniform strain in all four energy absorbing devices simultaneously. How would the secondary core supports function in the event of a partial, non-uniform failure of the primary core supports resulting in potentially vertical plus rotational displacement of the core? Could rotational plus vertical displacement of the core prevent complete insertion of the control rods?

210.026

Reference, Volume 6, Section 3.9.5.3.2, Page 3.9-81: The functional description of the secondary core supports indicates that the kinetic energy resulting from the postulated core drop accident is absorbed by tensile deformation. Please provide additional secondary core support arrangement details sufficient to illustrate how the energy absorbing material is loaded in tension.

210.027

Reference, Volume 6, Section 3.9.5.2.1, “Level A and B Service Conditions,” Page 3.9-79: The last bullet specifies “Earthquake” as a service condition for both Level A and Level B service limits. How is this earthquake defined? Is it included in <u>both</u> Level A and Level B load combinations? Earthquake loading does not appear to be included in <u>any</u> of the Level A or Level B loading combinations listed in Table 3.9-5.

Please provide clarification regarding earthquake loadings, other than SSE, and loading combinations considered in the design of the RPV core support structures.

210.028

Reference, Volume 6, Table 3.9-5, Page 3.9-102: The Level B Service loading combinations do not appear to include earthquake loading (see NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 3.9.3, Appendix A, C.1.3.2). The Level C Service loading combinations do not appear to include design basis pipe break loading (see SRP 3.9.3, Appendix A, C.1.3.3). Please clarify.

210.029

Reference, Volume 6, Table 3.9-9, Page 3.9-107: Under the column heading labeled “Core Supports,” there appears to be typographical errors in some of the references to the ASME Code Sections for Service Levels C and D. Please confirm that the reference listed as NG-3324 should be NG-3234 for Service Level C, and the reference listed as NG-3335 should be NG-3235 for Service Level D.

Piping Design Acceptance Criteria

Reference: AP1000 Design Control Document (DCD), Revision 2, April 2002.

General

210.030

ASME Code Cases used in the AP1000 design and analysis are listed in Table 5.2-3 of the DCD. Please identify the specific Code cases that are applicable that will be used in the design and analysis of piping systems, including piping components and associated supports.

210.031

Section 3.9.3 of the DCD states that the design specifications and design reports will be completed by the COL applicant or his agent. It also states that design specifications for ASME Class 1, 2, and 3 components and piping are prepared utilizing procedures that meet the ASME Code. Please provide these procedures and discuss differences between the AP600 design and the AP1000 design for staff review.

210.032

Section 3.9.3 does not provide any information on the completion status for the large bore piping design and analysis. The DCD does not clearly describe where the DAC will be defined and used. Please identify the specific piping systems that will be designed and analyzed as part of the COL applicant's scope.

210.033

Actual piping and pipe support design/analysis have not yet been performed and may not be completed as part of the design certification. However, the staff learned in a public meeting with W on July 17, 2002, that preliminary piping layouts were completed and drawings were available. Please provide layout drawings and address the feasibility of including these piping layouts in the DCD.

Section 3.6.2

210.034

The third paragraph of Section 3.6.2.3.3 referenced an Electric Power Research Institute (EPRI) report (Reference 8) to confirm that for piping systems without closing check valves, there is insufficient energy in the high-frequency depressurization loadings to cause a collapse of the piping system. The tests were performed to simulate seismic and system loading until piping failure occurs. It is not clear how W could use the results of the test report to justify not considering loadings generated from the internal system depressurization in verifying piping system integrity and operability. Furthermore, Reference 8 is Volume I, "Project Summary," of a November 1989 draft of an EPRI report. The final report was published in October 1994. The final report should be referenced. In addition, Volume 3, "System Tests," may be more applicable than Volume I for referencing. Please modify the text and the reference accordingly. The modified version should quote specific portion(s) of the report for the purpose of the AP1000 DCD since the reference contains a number of tests and evaluations that are not applicable to the case being discussed in this section.

210.035

The completion status of the pipe rupture analysis for the AP1000 needs to be clarified. Subsection 3.6.2.5, under “Verification of the Pipe Break Hazard Analysis,” states that to support design certification, the pipe rupture hazard analysis is complete except for the final piping stress analysis, pipe whip restraint design, and the as-built reconciliation. However, the staff notes that: (1) pipe stress analysis is needed to identify intermediate break locations, and (2) pipe whip restraints protect essential components from unacceptable potential consequences resulting from pipe breaks.

As described in Subsection 3.6.2.5, the as-built reconciliation includes a number of activities normally associated with design including ASME Code fatigue analysis, evaluation of pipe break dynamic loads, reconciliation to the design floor response spectra, and confirmation of the reactor coolant loop (RCL) time history seismic analysis.

Table 3.6-2, “Subcompartments and Postulated Pipe Ruptures,” and Table 3.6-3, “NI Rooms with Postulated High Energy Line Breaks/Essential Targets/Pipe Whip Restraints and Related Hazard Source,” are essentially identical to the corresponding tables in the AP600 DCD with the exception that larger pipe sizes are listed for certain systems.

Based on the above comments: (1) the pipe rupture hazard analysis does not appear to be complete, and (2) if pipe rupture protection is based on the AP600 design, the differences between the AP1000 design and the AP600 design may not have been adequately considered. It appears that the bulk of the pipe rupture hazard analysis design effort will be the responsibility of the COL applicant. Furthermore, considering the higher power rating and larger pipe sizes of the AP1000 as compared to the AP600, it is reasonable to expect that the higher energy pipe rupture interactions will be potentially more damaging. A design based on pipe rupture protection for the AP600 may not be adequate. Additional information describing the W design effort in this area and addressing these concerns should be provided.

Section 3.6.4.2

210.036

The COL applicant needs **to confirm** that the results of the as-designed piping stress analysis fall under the bounding analysis curve for LBB as documented in Appendix 3B. The AP1000 DCD quoted “LBB criteria” and “LBB evaluation report” in Table 2.2.3-4 under “Design Commitment and Acceptance Criteria.” Please define clearly the term “LBB criteria” and “LBB evaluation report” and discuss how bounding curves, as described in Appendix 3B, will be considered by piping analysts in the design stage without completing the piping analyses and the LBB demonstration evaluation to assure the compliance to LBB criteria during the final as-built reconciliation phase.

Section 3.7.2

210.037

Section 3.7.2.5: Is there any enveloping involved in generating the floor response spectra as stated? It appears that there is only one analysis performed for a hard rock site. This should be clarified or corrected, if necessary.

Section 3.7.3

210.038

Section 3.7.3.6 states that for time history analysis, "When the responses from the three components of earthquake motion are calculated simultaneously, each component is statistically independent of the other two. For this case, the components are combined by algebraic sum." This is incorrect. The staff position in SRP 3.7.2 II.6 states that the responses from each of the three components of earthquake motion may be combined algebraically at each time step. When this method is used, the components of earthquake motions specified in the three different directions should be statistically independent (i.e., the input motions not response motions, must be statistically independent). This should be clarified.

210.039

Sections 3.7.3.8.2.2 and 3.9.3 indicate that the small bore piping design and analysis will be completed by the COL applicant as part of the as-built reconciliation. Please provide the small bore piping design and analysis procedures and criteria for staff review.

210.040

Section 3.7.3.15: W should verify that all limitations specified in RG 1.84 for Code Case N-411 apply to the use of 5 percent damping.

210.041

Section 3.7.3.15 of the DCD provides information on damping and references Section 3.7.1.3 for additional information. The staff found the information in Section 3.7.3.15 acceptable for piping systems. However, the following two inconsistencies were noted in Section 3.7.1.3 and need to be corrected:

Section 3.7.3.15 states that for time history analysis and independent support motion analysis of piping systems, damping values of 4 percent, 3 percent, and 2 percent are used as described in Table 3.7.1-1. Section 3.7.1 3 only specifies 5 percent damping for piping and 4 percent for the primary coolant loop.

Section 3.7.3.15 states that for subsystems composed of different material types, the composite modal damping approach with the weighted stiffness method is used to determine the composite modal damping value. Section 3.7.1.3 indicates that the composite modal damping is calculated using the strain energy method.

210.042

Section 3.7.3.17 discusses time history broadening which generally involves performing three analyses to include normal, as well as contracted and expanded time scales to account for uncertainties. References to time scale variations of "+ or - 15 percent" and to stiffness variations of "+ or - 30 percent" should be corrected to indicate "+ and - 15 percent" and "+ and - 30 percent," respectively, since both variations must be analyzed. This subsection also states that when the results are shown to be acceptable based on comparison with test data, only one analysis may be performed using normal time. For what types of loadings and under what conditions would this option be used? Provide justification.

210.043

Section 3.7.3.17 states that either direct integration or modal superposition methods may be used in performing time history analysis. However, there is no description of the direct integration analysis methodology. Please provide additional information to describe the significant aspects of this analysis methodology including computer programs, criteria for selection of time steps, specification of damping parameters, treatment of high frequency modes, and consideration of uncertainties.

210.044

The first paragraph in Section 3.7.3.17 indicates that WECAN is not used for linear time history analysis or response spectra analysis of piping systems. The last paragraph in this section discusses the use of WECAN in modal time history analysis (which is generally a linear analysis). Please clarify the application of WECAN in piping analysis and correct the apparent inconsistency in this section.

Section 3.9.1

210.045

Computer programs used in the AP1000 analysis are discussed in DCD Section 3.9.1.2 and are listed in Table 3.9-15. For piping design certification, Section 3.9.1.2 indicates that W will use PIPESTRESS, GAPPIPE, WECAN, AND ANSYS.

a. Provide a summary listing to identify the program or programs that will be used to analyze each specific piping system as well as the type of analysis that will be performed for each system.

b. WECAN and ANSYS are general purpose finite element analysis programs that are not normally used for piping analysis and does not have the built-in capability for performing ASME Code evaluations. Provide detailed information on their specific application in the AP1000 piping analysis and on their verification and validation for this type of application.

c. GAPPIPE is a special purpose computer program for the evaluation of piping systems that use gapped supports in place of snubbers. Identify the specific AP1000 piping systems that utilize these supports. Provide detailed information on the applicability and limitations of the GAPPIPE program and on its verification and validation for use in the AP1000.

d. The PIPESTRESS program is not listed in Table 3.9-15. Instead, the table lists PS+CAEPIPE and CAEPIPE as piping analysis programs. This appears to be an oversight that should be corrected.

e. It appears that PIPESTRESS will be the primary program for piping analysis and design. Provide a summary description of this program including its capabilities, limitations, verification and validation.

f. W reported in the July 17, 2002, public meeting that the PIPESTRESS program to be used for the AP1000 piping design and analysis is the same computer program PS+CAEPIPE used for the AP600 design. Please clarify and highlight the control and any changes in the process of converting PS+CAEPIPE to PIPESTRESS.

g. W needs to provide three sample piping analysis problems for NRC's consideration of independent benchmarking. Specific piping systems, computer programs, types of analyses, and types of loading will be identified and agreed upon at a later date.

h. The staff learned that the three piping benchmarking problems used in the AP600 were selected as verification problems for the PIPESTRESS computer code. The sample problems were re-analyzed and re-verified for every new revision of the PIPESTRESS code and, therefore, re-run of the sample problems by NRC is not required. While this is an acceptable and more effective alternative to g. above, it needs to be reflected in the AP1000 DCD for formal documentation.

210.046

Thermal hydraulic loads are applicable to several piping systems experiencing valve opening and discharge loads. Please provide the name and the description of the computer programs, along with the verification and validation, to be used for the thermal hydraulic analyses of the AP1000 design. The program used to derive the forcing functions should also be described.

Section 3.9.3

210.047

In defining the seismic events, cycles and magnitudes, Section 3.7.3.2 states as follows: ".......two safe shutdown earthquake events with 10 high-stress cycles per event. .... For ASME Class 1 piping, the **fatigue evaluation is performed based on five seismic events** with an amplitude equal to one-third of the safe shutdown earthquake response. Each event has 63 high-stress cycles".

Section 3.9.3.1.1 states "In addition, systems and components sensitive to fatigue are evaluated .... by including 20 full cycles of the maximum safe shutdown earthquake stress range **or five seismic events** each resulting in 63 full stress cycles with a magnitude equal to one-third of the calculated safe shutdown earthquake response......".

Please correct the apparent inconsistency between the two sections and if Section 3.9.3.1.1 is correct, W needs to say "whichever results in higher (or lower) cumulative usage factor."

210.048

As indicated in Section 3.9.3.1.2 under required actions in response to Bulletin 88-11, Request 2(c), W states that monitoring of the AP1000 surge line is not required. However, the last paragraph in this section states that a monitoring program will be implemented by the COL holder at the first AP1000. This inconsistency should be clarified.

210.049

Section 3.9.3.1.2: The discussion on the identification and evaluation of unisolable lines susceptible to thermal cycling (Bulletin 88-08) is identical to the AP600. Did W consider the differences between the AP1000 design and the AP600 design with regard to parameters that may affect the thermal cycling and stratification loadings (fluid temperatures, pressures, flow rates)? Detailed calculations should be provided for staff review.

210.050

Section 3.9.3.1.2: The discussion on the identification and evaluation of the pressurizer surge line susceptible to thermal stratification (Bulletin 88-11) is identical to the AP600. Did W consider the differences between the AP1000 and the AP600 with regard to the potential for stratification between the pressurizer and the hot leg? Specifically, it is not well known that the pressurizer could be stratified and **the heat-up and cool-down rate** could exceed the defined limit with large surge flow rate. Please describe in the DCD the control of the heatup and cooldown procedure such that the ΔT between the pressurizer and the reactor coolant system (RCS) hot leg will be less than acceptable value(s) and pressurizer stratification will not be a concern from the stress and fatigue points of view.

210.051

Section 3.9.3.1.5, Section 3.9.3.1.3, Section 3.9.3.1.7: W needs to clarify whether Sections 3.9.3.1.3 and 3.9.3.1.7 and the tables they reference apply to piping only or to other ASME Class 1, 2 and 3 components. They also need to clarify that Tables 3.9-5, 3.9-9, and 3.9-10 apply to piping (these tables are not referenced in Section 3.9.3.1.5 which discusses piping).

210.052

The last paragraph in Section 3.9.3.1.5 indicates that a monitoring program for the feedwater line at the first AP1000 is identified in Subsection 3.9.3.1.2. The staff did not find any information on a feedwater line monitoring program within this subsection. Please explain and make the necessary correction.

210.053

Section 3.9.3.1.7 indicates that there are no special stress limits required to provide functional capability. This is inconsistent with Section 3.9.3.1.5 and Table 3.9-11 which discuss and provide functional capability requirements. Please clarify and make the necessary correction.

210.054

It was discussed in the public meeting on July 17, 2002, that the AP1000 is designed for a 60-year life. However, when licensed, it will only be for a 40-year life. Please clarify in the DCD the fatigue life considered in the design and how the environmental effects on fatigue will be addressed.

210.055

Current test data indicates that the ASME Code, Section III design fatigue curves may not be conservative for nuclear power plant primary system environments. The Section III Subgroup on Design (SGD) has formed a task group to provide recommendations to the SGD regarding the effect of the environment on the Section III design fatigue curves. The NRC staff has been addressing the environmental fatigue issue in its review of license renewal applications. The NRC staff-referenced evaluations of the current test data are provided in NUREG/CR-6583, "Effects of LWR [Light-Water Reactor] Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," in its license renewal reviews. Describe the method that will be used to account for the effect of the environment on the fatigue design of reactor coolant pressure boundary components in the AP-1000 plant.

210.056

Section 3.9.3.3.1: The discussion on the design of the pressurizer safety and relief valve (PSARV) module is identical to the AP600. The differences, based on the staff's understanding, between the AP600 design and the AP1000 design are focused on whether the analysis of the piping system and supports have been completed at the design certification stage. Please justify, in the DCD, that the AP1000 plant specific PSARV piping configuration can be designed to withstand the combined action of transient thrust forces and the thermal gradients caused by the valve opening without performing the structural dynamic and thermal fatigue analysis.

Appendix 3E

210.057

This appendix contains a discussion on the hot water heating system (VYS), which includes a limited amount of high-energy piping in the auxiliary building. No breaks are postulated in these 3-inch lines in the nuclear island because there are no anchors or fittings on these lines in the nuclear island. "No anchor or fittings" is not an adequate basis for not postulating breaks on a high energy line. Please provide more specific justification, following the criteria contained in the SRP, for the conclusion that no breaks need to be postulated.

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		
3	8/27/2002	ML022390103	420.001 - 420.046, 435.001 - 435.015		
4	9/3/02	ML022460356	620.001 - 620.043		

Enclosure 2

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