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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: R. W. BORCHARDT

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of March 16, 1994, April 29, 1994, May 5, 1994, May 11, 1994, May 16, 1994 and May 23, 1994. A number of the responses for requests for additional information from the March 16th letter have been discussed with the NRC staff. These responses will be transmitted when the staff comments are resolved.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

These responses are also provided as electronic files in WordPerfect 5.1 format with Mr. Kenyon's copy.

If you have any questions on this material, please contact Mr. Brian A. McIntyre at 412-374-4334.

Bri A McIntyre/for
Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory And Licensing Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
T. Kenyon - NRR

NTD-NRC-94-4171
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JUNE 16,1994

RAI No.	Issue
210.035	Addition of QA requirements to SSAR Table 3.2-3
210.037	Consistency of SSAR Table 3.2-3 & Sect 5.4.7.1.2
210.061	Low pressure side design critreia
210.071	Sheet 53 of SSAT Table 3.2-3
210.074	Exception to position C.7.b in RG 1.124
210.075	Exception to position C.6.b in RG 1.130
210.091	SRP section 3.9.3
220.078	Modular construction design information
410.126	CCS potential for water hammer
410.127	Analysis of CCS piping cracks
410.146	Main steam supply quality group classification
410.216	Protection of MCR/RSW from missiles
410.221	Missile prevention in shaft seal pump
440.125	Credit taken for CVCS
440.164	Loose parts monitoring system
480.051	Component cooling system isolation signals

Question 210.35

Quality assurance programs should be identified in Table 3.2-3 of the SSAR. Revise Table 3.2-3 to add a column that identifies the quality assurance requirements for each line item in the table.

Response:

The quality assurance requirements for systems and component are determined by the equipment classification. As noted in Subsection 3.2.2 and Table 3.2-1, AP600 equipment Classes A, B, C, and D correspond to Regulatory Guide 1.26 Quality Groups A, B, C, and D respectively. The quality requirements for AP600 equipment classes A, B, C, and D are discussed in Subsection 3.2.2.3 through 3.2.2.6. The identification of a component as Class A, B, or C uniquely identifies the quality assurance program requirements. Additional discussion on quality requirements for equipment Class D is provided in the response to RAI 260.12.

AP600 equipment classes E, F, L, P, R, and W are nonsafety-related classifications that apply to equipment not covered by the other classes. These classes are discussed in Subsection 3.2.2.7. Equipment in Classes F, L, P, R, and W are designed to comply with specific industrial standards such as the National Fire Protection Association Code, the National Plumbing Code, etc.; otherwise, there are no special quality assurance for equipment classes E, F, L, P, R, and W that need to be added to Table 3.2-3.

SSAR Revision: NONE

Question 210.37

Table 3.2-3 and Figure 5.4-7 of the SSAR, "P&ID for the Normal Residual Heat Removal System (RNS)," identifies all components within the RNS as seismic Category I and either AP600 Class A, B, or C (ASME Class 1, 2, or 3), with the principal construction code as ASME NB, NC, or ND, respectively. The staff's interpretation of these commitments is that RNS components will be constructed in accordance with ASME Subsections NB, NC, and ND, as applicable, where construction is as defined in NB-1110(a) or NC/ND-1100(a). In addition, applicable ASME Section XI rules will apply. However, Section 5.4.7.1.2 of the SSAR states that the RNS piping and components are Safety Class C, seismic Category I for pressure retention purposes only. This statement does not appear to be consistent with the staff's interpretation of Table 3.2-3 and the P&ID. Revise Section 5.4.7.1.2 to be consistent with Table 3.2-3 and Figure 5.4-7.

Response:

The normal residual heat removal system (RNS) components outside containment (between the containment isolation valves) are specified with an AP600 Class C, Seismic Category I pressure boundary. This classification recognizes the importance of pressure boundary integrity even though these components have no safety-related functions.

The RNS components will be constructed in accordance with ASME Subsections NB, NC, and ND, as identified in SSAR Table 3.2-3, where construction is as defined in NB-1110(a) or NC/ND-1100(a). Inservice testing of the safety-related active valves will be in accordance with ASME Section XI. Inservice testing of the RNS pumps' operability in accordance with ASME Section XI is not required because they do not serve an active safety function.

SSAR Revision:

Add a second paragraph to Subsection 5.4.7.1.1 as follows. Section 1.9.5.1 and 5.4.7.2.2 will be revised as shown in the response to RAI 210.61.

The normal residual heat removal system component outside containment (between the containment isolation valves) are specified with an AP600 Class C, Seismic Category I pressure boundary. This classification recognizes the importance of pressure boundary integrity even though these components have no safety-related functions.

Revise the first paragraph of Subsection 5.4.7.1.2 as follows:

Subsection 5.4.7 outlines the principal functions of the normal residual heat removal system. The normal residual heat removal system is designed to be ~~very~~ reliable. This reliability is achieved by using ~~highly reliable and~~ redundant equipment and with a simplified system design. The normal residual heat removal system is not a safety-related system. It is not required to operate to mitigate design basis events. ~~The system piping and components are Safety Class C, Seismic Category I for pressure retention purposes only.~~

The normal residual heat removal system is designed for a single nuclear power unit and is not shared between units. The normal residual heat removal system is designed to be ~~fully operable~~ operated from the control room.

Question 210.61

Section 1.9.5.1 of the SSAR, "SECY-90-016 Issues," under Intersystem LOCA, states, in part, that the design pressure of the normal residual heat removal system (RNS) piping downstream of the RNS isolation valves (including RNS pump casings and heat exchanger tubes) is designed to an ultimate rupture strength equal to full reactor coolant system (RCS) operating pressure. Provide a more detailed description of the low pressure side design criteria in order to reduce the likelihood of an intersystem LOCA. Revise Section 1.9.5.1 and any other applicable sections of the SSAR to provide the following for all AP600 systems where overpressurization of low-pressure piping systems due to RCS boundary isolation failure could result in rupture of the low-pressure piping outside containment:

a. Pipe and pipe fittings - Provide the pipe schedule number for all applicable diameters and materials.

b. Valves and flanges - Provide the American National Standard Class.

c. Pumps and heat exchangers - Provide the ratio of design pressure to RCS pressure.

Response:

The response to RAI 440.132 identifies the low pressure systems that carry reactor coolant outside containment that could fail due to overpressurization. Of those identified, system overpressurization that could result in the rupture of low pressure piping due to RCS pressure boundary failure can only occur in the normal residual heat removal system. Provided below is the requested information for the low pressure portion of the normal residual heat removal system:

a. The pipe schedule for the normal residual heat removal system AP600 Equipment Class C piping outside containment is 80S.

b. The American National Standard Class for the valves, flanges, and fittings in the AP600 Equipment Class C portions of the normal residual heat removal system outside containment has been specified to be greater than or equal to Class 900.

c. The ratio the of normal residual heat removal system and component design pressure to reactor coolant system normal operating pressure is 900 psig to 2235 psig, or 40 percent.

Further clarification is provided in the response to RAI 210.37.

SSAR Revision:

Revise the AP600 response to Intersystem LOCA in Subsection 1.9.5.1 as follows:

One of the potential sources for an intersystem LOCA is the residual heat removal suction line. This line connects the reactor coolant system hot leg to the residual heat removal system pump suction and therefore penetrates the containment boundary. The low-pressure portions of this line are normally isolated from the reactor coolant system by normally closed, series motor-operated valves.

The AP600 normal residual heat removal system ~~design~~ includes the following features ~~specifically aimed at~~ to reduce the likelihood of an intersystem LOCA:

- The portions of the normal residual heat removal system located outside containment (that serve no active safety functions) are classified as AP600 Equipment Class C so that the design, manufacture, installation, and inspection of this pressure boundary is controlled by the following industry and regulatory quality assurance requirements: 10CFR21; 10CFR50, Appendix B; Regulatory Guide 1.26 Quality Group C; and ASME Boiler and Pressure Vessel Code, Section III, Class 3. In addition, this pressure boundary is classified as Seismic Category I so that it is protected from failure following a safe shutdown earthquake.

- The portions of the normal residual heat removal system from the reactor coolant system to the containment isolation valves outside containment are designed to the operating pressure of the reactor coolant system. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve (including normal residual heat removal system pump casings and heat exchanger tubes) are designed so that its ultimate rupture strength is not less than the operating pressure of the reactor coolant system--specifically, the piping is designed as schedule 80S, and the flanges, valves, and fittings are designed to greater than or equal to ANS class 900.

- The common normal residual heat removal system suction line outside containment includes a normally closed, motor-operated isolation valve. This valve is in addition to the two normally closed, series motor-operated isolation valves in each parallel suction line inside containment.

- The normal residual heat removal system suction valves inside containment are interlocked to prevent them from opening when reactor coolant system pressure is above the normal residual heat removal system operating pressure. In addition, these four valves and the series valve outside containment are normally maintained in a condition with control power locked out to prevent inadvertent actuation. The suction piping upstream of the containment penetration piping includes a relief valve inside containment which protects the low-pressure piping from overpressurization and reduces the potential for rupture of system piping outside containment.

- The normal residual heat removal system pump seal is not designed for the higher normal residual heat removal system piping design pressure and could fail if overpressurized. However, the ~~design of~~ normal residual heat removal system limits the leakage to ~~within~~ the capacity of the chemical and volume control system. ~~, to allow the plant to be placed in safe shutdown.~~

See Subsection 5.4.7 for additional information on the normal residual heat removal system.

Add a paragraph in Subsection 5.4.7.2.2 between the existing first and second paragraph as follows:

Quality Assurance/Seismic Protection - The portions of the normal residual heat removal system located outside containment (that serve no active safety functions) are classified as AP600 Equipment Class C so that the design, manufacture, installation, and inspection of this pressure boundary is controlled by the following industry and regulatory safety-related quality assurance requirements: 10CFR21; 10CFR50, Appendix B; Regulatory Guide 1.26 Quality Group C; and ASME Boiler and Pressure Vessel Code, Section III, Class 3. In addition, this pressure boundary is classified as Seismic Category I so that it is protected from failure following a safe shutdown earthquake.

Revise the existing second paragraph in Subsection 5.4.7.2.2 as follows:

Increased Design Pressure - The portions of the normal residual heat removal system from the reactor coolant system to the containment isolation valves outside containment are designed to the operating pressure of the reactor coolant system. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve are designed so that its ultimate rupture strength is not less than the operating pressure of the reactor coolant system. Specifically, the piping is designed as schedule 80S, and the flanges, valves, and fittings are specified to be greater than or equal to ANS class 900.

Question 210.71

Sheet 53 of Table 3.2-3 of the SSAR does not appear to contain the core barrel and the control rod drive mechanism (CRDM) housings as a part of the reactor system. Revise this table to include these two components. If the CRDM housings are considered a part of the reactor vessel, add a note to this effect.

Response:

The core barrel will be added to Table 3.2-3 of the SSAR.

The CRDM Latch Housing, Rod Travel Housing, Latch Assembly, Coil Stack Assembly and Drive Rod Assembly are the part of CRDM and they are listed in Table 3.2-3 of SSAR under Reactor System (RXS).

SSAR Revision:

The following entry will be added to Table 3.2-3, under Reactor System (RXS):

Description	Loc.	AP600 Class	Seismic Category	Principal Construction Code
Core Barrel	11	B	I	ASME III, Subsection NG

Question 210.74

In the SSAR, Appendix 1A takes exception to Position C.7.b in RG 1.124 relative to Service Level D allowable loads for ASME Class 1 linear-type component supports designed by the load rating method. Revision 1 to WCAP-13054 contains this same exception. The exception states that the AP600 will use rules in ASME Appendix F, Section F-1370(d). F-1370(d) has been replaced by the load rating rules in F-1332.7, which is acceptable to the staff. Section 3.9.3.4 of the SSAR states that all AP600 ASME Class 1 supports are designed to ASME III, Subsection NF and Appendix F. Revise the exceptions in Appendix 1A of the SSAR, WCAP-13054, and any other applicable SSAR section to reference F-1332.7 rather than F-1370(d).

Response:

The reference to F-1370(d) is incorrect. The exception to Position C.7.b in Regulatory Guide 1.124 identified in Appendix 1A of the SSAR will be changed to delete the ASME Code, Section III, Appendix F paragraph reference as shown below. The load rating method in Appendix F is found in Subparagraphs F-1332.7 and F-1334.8 and Paragraph 1333. The reference to the use of the load rating (test load) method in SSAR Subsection 3.9.3 does not identify the specific paragraph in ASME Code, Section III, Appendix F. The exception to the Position C.7.b in Regulatory Guide 1.124 will be revised in the next revision of WCAP-13054.

SSAR Revision:

In the Appendix 1A discussion of conformance with Regulatory Guide 1.124 revise the exception to position C.7.b as follows:

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
C.7.b		Exception	The AP600 uses the provisions of the ASME Code, Section III, Appendix F ~~F-1370(d)~~ to determine faulted condition allowable loads for supports designed by the load rating method. The method described in this regulatory position is conservative and inconsistent with the remainder of the faulted stress limits.

Question 210.75

In the SSAR, Appendix 1A takes exception to Position C.6.b in RG 1.130 relative to Service Level D allowable loads for ASME Class 1 plate-and-shell-type component supports designed by the load rating method. Revision 1 to WCAP-13054 contains this same exception. The exception presents an equation for an allowable load rating which is not consistent with ASME III, Appendix F, Section F-1332.7. Revise the exceptions in Appendix 1A of the SSAR, WCAP-13054, and any other applicable SSAR section to reference F-1332.7.

Response:

The reference to F-1370(d) is incorrect. The exception to Position C.6.b in Regulatory Guide 1.130 identified in Appendix 1A of the SSAR will be changed to delete the calculation and add a reference to the ASME Code, Section III, Appendix F as shown below. The load rating method in Appendix F is found in Subparagraphs F-1332.7 and F-1334.8 and Paragraph F-1333. The reference to the use of the load rating (test load) method in SSAR Subsection 3.9.3 does not identify the specific paragraph in ASME Code, Section III, Appendix F. The exception to the Position C.7.b in Regulatory Guide 1.124 will be revised in the next revision of WCAP-13054.

SSAR Revision:

In the Appendix 1A discussion of conformance with Regulatory Guide 1.124 revise the exception to position C.6.b as follows:

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
C.6.b	ASME Code, Section III, NF-3262.1	Exception	The limit based on the test load given in this regulatory position is overly conservative and is inconsistent with ASME Code requirements. The AP600 uses the provisions of ASME Code, Section III, Appendix F to determine faulted condition allowable loads for supports designed by the load rating method. ~~The AP600 uses the limit as calculated by T.L. x 0.8 S'u/Su.~~

Question 210.91

Revision 1 to WCAP-13054 lists an exception to Section C.4.1.(c) of Appendix A to Section 3.9.3 of the SRP, that implies that since the AP600 does not have an FSAR, information relative to how the criteria in Sections 1 and 2 of Appendix A have been implemented will not be in the SSAR. The staff's position is that this information should be available for the reactor system and most of the reactor coolant system, and should be discussed in the SSAR. Revise this exception in WCAP-13054 to identify those components/systems for which this information will be provided and revise the applicable SSAR section to include this information.

Response:

A number of design specifications for reactor system and reactor coolant system components have been developed. In conformance with the guidelines of the Standard Review Plan, these are available for NRC review or audit. The final design reports and other information listed in the referenced position of the Standard Review Plan can only be completed when the final as-built information is available. An example of intermediate information available for review or audit is the summary of the methods and results for the reactor coolant loop piping is found in Appendix 3C of the SSAR. Revision of the SSAR is not required to provide the NRC an opportunity to review and audit design specifications and preliminary design analysis.

The position on Section C.4.1.(c) of Appendix A to Section 3.9.3 of the Standard Review Plan will be revised in the next revision of WCAP-13054.

SSAR Revision: NONE

Question 220.78

For the modular construction design, provide the following information:

a. a copy of the detailed design drawings for one of the common M-subunit modules,

b. the Modular D sign Criteria Document, and

c. the current version of the construction procedures related to modular construction.

Response:

a. Detailed drawings for submodules are available for review at the Westinghouse offices. The details that are shown in the submodule drawings are typical of M-type structural modules.

b. The General Module Design Criteria and the Structural Module Design Criteria documents have been submitted via Westinghouse letter NTD-NRC-94-4097 dated April 14, 1994. The Design Methodology of Structural Modules is available for review at the Westinghouse offices.

c. The AP600 Construction Plan describes the construction procedures to be used with modules. This document is available for the NRC staff and consultants to review at the Westinghouse Rockville Nuclear Licensing Center. In addition, the AP600 modular construction approach is described in the paper "Modular Construction Approach for Advanced Nuclear Plants" as published in the proceedings for the November 1988 international conference of the American Nuclear Society and European Nuclear Society.

SSAR Revision: NONE

Question 410.126

Describe how the AP600 CCS is designed to minimize the potential for water hammer.

Response:

The operating temperatures of the components will normally be well below 200°F . The low pressure point in the component cooling water system, the suction of the component cooling water pumps, is maintained at a pressure greater than atmospheric by the combined surge tank elevation head and nitrogen cover gas pressure. Because the component cooling water system will normally operate at temperatures and pressures which prevent formation of steam bubbles, water hammer issues are avoided

SSAR Revision: NONE

Question 410.127

Provide information with respect to the analysis of postulated cracks in moderate-energy piping systems for the CCS.

Response:

The component cooling water system is a nonsafety-related system. The nonsafety-related portions of the CCS do not require analysis of the affects of postulated piping cracks in moderate energy piping systems. The safety-related portions of the CCS (the containment penetrations and isolation valves) are subject to such analysis as described in SSAR section 3.6.

SSAR Revision: NONE

Question 410.146

Section 10.3.3 of the SSAR states that Section 3.2 provides the quality group classification, the required design and fabrication codes, and the seismic category applicable to the safety-related portion of the main steam supply and supporting systems. Section 3.2.4 states that Table 3.2-3 lists mechanical and fluid system components and their associated equipment class and seismic category as well as other related information. However, the staff cannot find the information on the quality group classification, the code requirements, and seismic category for the main steam supply system in Table 3.2-3 or Section 3.2. Provide the above information.

Response:

The safety-related portions of the main steam supply and the main and startup feedwater supply are included in the steam generator system (SGS). The steam generator system is included in Table 3.2-3. As noted in Subsection 3.2.4, systems that contain no components in AP600 equipment Classes A, B, C, or D, which includes the main steam system, have no list of components in Table 3.2-2.

SSAR Revision: NONE

Question 410.216

How are the main control room (MCR) and remote shutdown workstation (RSW) protected from internally-generated missiles (outside containment)?

Response:

Both the main control room and remote shutdown workstation are located in the auxiliary building. These areas are protected from internally-generated missiles outside containment by the structural concrete walls (internal and external) and floors of the auxilliary building.

SSAR Revision: NONE

Question 410.221

The January 22, 1993, response to Q410.65 implies that mass around the impeller and rotating parts of the motor is the primary means used to prevent missiles in the shaft seal pump. Is this interpretation correct?

Response:

Yes, the basis for not considering the impeller and rotating parts of the motor of a shaft seal pump to be credible missiles is the retention provided by the strength of the mass around these parts.

SSAR Revision: NONE

Question 440.164

Revision 1 to Regulatory Guide (RG) 1.133, "Loose-Part Detection Program for the Primary System Light-Water-Cooled Reactors," was issued as part of resolution of TMI Task Action Plan Items B-60, "Loose Parts Monitoring Systems," and C-12, "Primary System Vibration Assessment." Provide a discussion of compliance of the AP600 design with the guidance of RG 1.133 regarding a loose parts monitoring system.

Response:

The digital metal impact monitoring system is described in Subsection 4.4.6.4. The conformance of the digital metal impact monitoring system with the regulatory positiors in Regulatory Guide 1.133 Rev. 1 is dicussed in Appendix 1A.

SSAR Revision: NONE

Question 440.125

Section 5.4.3.1 of the SSAR classifies the small lines in the RC system with a 3/8-inch or less flow restricting orifice as AP600 equipment Class B because, if one of these lines breaks, the chemical and volume control charging pumps are capable of providing makeup flow while maintaining pressure water level. 10 CFR 50.55a(c)(2)(i) allows the components that are connected to the RCS and are part of the RC pressure boundary (RCPB) to be excepted from meeting the Class A requirements for the RCPB if, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooldown in an orderly manner, assuming makeup is provided by the RC makeup system.

a. The CVCS in the AP600 design is a non-safety related system. What is the justification for taking credit for the CVCS makeup capability so that the small lines in the RCS can be excluded from meeting the Class A requirement for the RCPB?

b. What regulatory measures are proposed to ensure availability and reliability of the CVCS when called upon?

Response:

a. The requirements of 10 CFR 50.55a(c)(2)(i) do not require that the "normal makeup system" be safety related. The use of a nonsafety-related chemical and volume control system to supply makeup to the reactor coolant system is consistent with practice for some operating plants. In the AP600, the safety-related makeup for the reactor coolant system is supplied by the core makeup tanks.

   The core makeup tanks provide reactor coolant system makeup during transients or accidents when the normal reactor coolant system makeup supply from the chemical and volume control system is unavailable or is insufficient. The core makeup tanks have sufficient capacity to supply makeup to shut down and cool down the reactor in an orderly manner for any design basis leak or break without makeup supply from the chemical and volume control system. Section 6.3 of the SSAR describes how the passive core cooling system operates to shut down and cool down the reactor without any of the active systems. The chemical and volume control system is not relied on to supply makeup in the accident analyses documented in Chapter 15 of the SSAR.

b. Those portions of the chemical and volume control system required to perform safety-related functions (see SSAR Subsection 9.3.6.1.1) are classified as safety-related. See SSAR Section 3.2 for the equipment classification for the chemical and volume control system. The response to RAI 100.11 identifies the safety-related functions of the chemical and volume control system. Reactor coolant system makeup is not a safety-related function of the chemical and volume control system. No regulatory measures are required to provide for the availability and reliability of the nonsafety-related portions of the chemical and volume control system. The process used to determine that no additional regulatory measures are required for the chemical and volume control system is documented in Reference 440.25-1.

References

440.25-1. WCAP-13856, Revision 0, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process Summary Report, September, 1993.

SSAR Revision: NONE

Question 480.51

Sheet 2 of Figure 9.2.2-2 of the SSAR contains a note indicating that component cooling water system isolation valves close on trip of all the reactor coolant pumps. Other information in the SSAR indicates that component cooling system isolation results from other signals. is this note correct? If not, correct this note.

Response:

Note 5 of Figure 9.2.2-2, Sheet 2 of the SSAR reads as follows: 'Close on "S" signal (Trip of all RCPs).' The "S" signal is the safety injection signal. This signal initiates several safety actions including tripping the reactor coolant pumps. This note is consistent with SSAR sections 7.3, subsections 9.2.2.2, 9.2.2.3.4 and 9.2.2.4.5.2 and Table 6.2.3-1 which all say that the component cooling system containment isolation valves close upon safety injection "S" signal.

SSAR Revision: Remove "(Trip of all RCPs)" from Note 5 of Figure 9.2.2-2, Sheet 2.