Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

ET-NRC-92-3789
NSRA-APSL-92-0270
Docket No. STN-52-003
Proj. 676
December 22, 1992

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: DR. THOMAS MURLEY

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

Dear Dr. Murley:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of September 23, 1992, October 1, 1992, October 9, 1992 and October 28, 1992. This transmittal is a partial response to those letters. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A. The Westinghouse responses to the remainder of the requests for additional information contained in your letters of September 23, 1992 and October 1, 1992 will be provided prior to January 23, 1993.

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

Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
F. Hasselberg - NRR

E004
1/1

0733A

ET-NRC-92-3789
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED DECEMBER 22, 1992

RAI No.	Issue
100.003	Operational assessment
210.001	Safety and seismic classification
210.004	Reg Guide 1.26 compliance
210.008	Piping analysis
210.011	ISM piping analysis
210.015	Computer codes for stress analyses
220.015	Containment penetration reinforcement
230.006	Damped seismic design response
250.009	Pipe weld CUF
250.011	Access points
250.012	Tubesheet handholdes
250.013	Indexing
250.014	U-bend area access
250.015	Remote inspection features
250.017	Eddy current inspection
250.018	Leak corrective measures
250.019	Material strength properties
250.020	Exceptions to Reg Guide 1.121
251.032	SRP compliance
252.003	Analyses
252.004	Piping material property verification
252.005	Class 2 & 3 piping
252.006	Piping outside containment
252.010	Piping stresses for different sites
252.011	RCS piping stresses
252.047	ASS carbon content analysis

ET-NRC-92-3789
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED DECEMBER 22, 1992

RAI No.	Issue
252.060	Inaccessibility to cavities/chambers
252.107	Tube plugging criteria
252.108	Facilitating SG tube fusion techniques
252.109	Tube vibration wear
252.111	Misplaced anti-vibration bars
252.112	Delta-75 SG primary side manway size
252.113	Records/material archive program
252.114	AVT secondary water chemistry with Inconel 690
252.115	Stress corrosion cracking in Inconel 690
252.116	Inconel 690 resistance to corrosion
252.137	Materials
252.143	System failure jeopardizing safe plant shutdown
420.005	Annunciators, manual actions & defense-in-depth
420.006	ICS common mode failures
435.005	Post-72 hour portable diesels
435.007	Reg Guide 1.75 separation of nonsafety ac division
435.009	Diesel generator maintenance/testing program
435.012	Load sizing
435.017	Lightning protection of main setup transformers
435.024	Overvoltage protection of batteries
435.025	Ground detection of Class 1E dc system
435.032	Reg Guide 1.75 compliance
435.033	dc power system transient response
435.065	Circuit breakers used for electrical isolation
435.068	1E/non1E boundary ac power source/120 Vac bus
435.069	Emergency lighting requirements

ET-NRC-92-3789
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED DECEMBER 22, 1992

RAI No.	Issue
440.022	Fuel design and analysis
440.023	Mid-loop operations
440.030	Interfacing system LOCA
450.009	Sump pH control
471.002	Radiation zone designations
471.003	ALARA concerns
480.001	Operator actions during 72-hours post-accident
480.003	Failure modes for PCS
720.003	Failure modes
720.006	PRA methodology
720.007	Truncation limits
720.010	Human error probabilities
720.016	PRA assumptions
720.023	PRA assumption concerning PCS availability
720.026	Containment drain design
720.030	MAAP 4.0
720.035	Hot leg creep rupture
720.036	Hot leg creep rupture
720.038	Fuel-coolant interactions
720.039	Core-concrete interactions
720.045	Fission product holdup
720.046	Fission product transport and retention
720.047	Xenon & Krypton release fractions
720.050	Sensitivity & uncertainty for MAAP analysis
720.057	PRA
720.058	Data files

Question 100.3

An applicant for a standard design certification is required by 10 CFR Part 52 to address issues identified as a result of the TMI-2 event (10 CFR 50.34(f)), unresolved safety issues (USIs) and generic safety issues (GSIs) that have been prioritized as either high or medium, as well as all applicable rules and regulations. The NRC staff has, as part of its overall design certification review process, been reviewing operational experience information contained in NRC bulletins and generic letters issued since January 1980[1] to identify any potential safety issues that should be considered during the NRC staff's design certification review but may not be specifically addressed thru any of the mechanisms highlighted above.

The staff has reviewed the SSAR to determine whether operational experience information has been effectively incorporated into the AP600. The staff has concluded that the discussion provided in Section 1.9 of the SSAR does not contain sufficient detail for the staff to determine that operational experience information has been effectively incorporated into the AP600.

Provide additional details regarding incorporation of operational experience information into the AP600 design. Specifically, incorporate a tabulation of NRC bulletins and generic letters issued since January 1980 into Section 1.9 of the SSAR. This tabulation should include the applicability of each bulletin and generic letter to the AP600, as well as its disposition. Typical dispositions could include a) a determination that the issue identified in bulletin or generic letter is not applicable to the AP600 and the basis for that determination, b) a determination that the issue identified in the bulletin or generic letter will be addressed as part of a USI, GSI, Rule, Regulatory Guide, or TMI Action Item, including references to both the applicable USI, GSI, Rule, Regulatory Guide, or TMI Action Item and the appropriate section of the SSAR where the issue is or will be addressed, or c) a determination that the issue identified in the bulletin or generic letter is or will be covered by interface requirements for the COL applicant. In addition, a description of any enhancements made to the AP600 based on other sources of operational experience information (with appropriate references to the SSAR) should also be provided.

Response:

This RAI has been addressed by:

WCAP-13559, "Operational Assessment for AP600", December, 1992. Submitted December 15, 1992 in letter ET-NRC-92-3784.

SSAR Revision: NONE

[1]NUREG-4690, Vol. 1, Rev.1 "Generic Communications Index" provides a listing of NRC generic communications (bulletins, generic letters, circulars, and information notices) issued from 1971 to 1989.

Question 210.1

Discuss the justification for the safety and seismic classification of structures, systems, and components (SSCs) that are unique to the passive design of the AP600 (i.e. Passive Core Cooling System, Passive Containment Cooling System, etc.). The staff is concerned that there is no previous experience with these systems and they fall outside the structures, systems, and components that have traditionally been classified utilizing the guidelines set forth in Regulatory Guides 1.26, and 1.29 for safety and seismic classification, respectively (Section 3.2).

Response:

The safety philosophy of the AP600 is basically the same as current PWRs, with the safety-related systems performing the safety-related functions. What is different is that the safety-related functions are performed by different systems that are made up of different arrangements of piping, valves and components. Regulatory Guides 1.26 and 1.29 are functionally defined for the most part and therefore they have been applied to the safety-related AP600 systems. There are two areas of the AP600 design where these Regulatory Guides do not allow a straight-forward classification, as discussed below.

Subsection 3.2.2 of the SSAR describes the AP600 classification system. This system is very similar to and is compatible with Regulatory Guide 1.26. Most of the AP600 structures, systems and components have the same Regulatory Guide 1.26 classification. For example, the AP600 RCS components and piping are class A, which is equivalent to Regulatory Guide 1.26 quality group A. The AP600 containment is class B, which is equivalent to Regulatory Guide quality group B. The AP600 classification system differs from Regulatory Guide 1.26 in the following two aspects:

1) Portions of safety-related systems that interface with the RCS or containment are considered to be class C if they are not part of the RCS pressure boundary and they do not recirculate fluids from the RCS / containment outside of the containment. Regulatory Guide 1.26 requires quality group B for an ECCS system, but only requires quality group C for an auxiliary feedwater system.

   The AP600 classification requires a portion of a safety-related system that is part of the RCS pressure boundary to be class A, as in the current Regulatory Guide 1.26. If it recirculates post-accident fluid outside of the containment, it would be class B, as in the current Regulatory Guide 1.26. None of the AP600 safety-related systems that interface with the RCS or the containment recirculate fluid outside containment. Several portions of the PXS systems are class A because they are part of the RCS pressure boundary, including the PRHR HXs and the CMTs. The IRWST and the accumulators are class C because they are ECCS components located inside containment which can not recirculate post accident fluid outside of the containment. The passive containment cooling system water storage tank and its associated piping and valves are class C because they provide a safety-related function and they are located completely outside of the containment. This system only contains nonradioactive water.

2) Nonsafety-related systems that mitigate events are not addressed in Regulatory Guide 1.26. The AP600 classification system addresses this type of equipment. It uses a special nonsafety-related classification,

Class D, which applies to both defense-in-depth equipment that reduces the potential for passive safety-related system actuation and to radioactive waste processing equipment. This class is equivalent to Regulatory Guide 1.26 quality group D. For example, the startup feedwater system is class D because it has no safety-related functions, but does provide decay heat removal for events such as transients and loss of outside power.

SSAR Table 3.2-1 provides a comparison of the AP600 classification system with Regulatory Guide 1.26, as well as ANSI 51.1. SSAR Table 3.2-3 provides a listing of the classification of the equipment and valves in the AP600 systems. Appendix 1A of the AP600 SSAR discusses compliance with Regulatory Guides.

SSAR Revision: NONE

Question 210.4

The last sentence of the first paragraph in Section 3.2.2.1 of the SSAR states that "These definitions are consistent with the draft ANS Definitions for LWR Standards." These definitions should be explicitly identified in the SSAR since the staff does not presently endorse an ANS standard for the classifications of structures, systems, and components. The staff relies on Regulatory Guide 1.26 for that purpose. Provide technical justification for any deviations from Regulatory Guide 1.26.

Response:

The definition of the AP600 classifications is provided in SSAR section 3.2.2. These definitions are similar to Regulatory Guide 1.26. Refer to the response to RAI 210.1 for a discussion of the differences. Appendix 1A of the AP600 SSAR discusses conformance with regulatory guides.

SSAR Revision: NONE

Question 210.8

Section 3.7.3.8.2.2 of the SSAR states that for ASME Class 1 piping equal to or less than one inch nominal pipe size and ASME Class 2 and 3 piping equal to or less than two inch nominal pipe size, one of the following three methods of analysis may be used:

a. The method for large diameter pipe described in Section 3.7.3.8.2.1 of the SSAR.

b. Equivalent static analysis.

c. Seismic qualification by experience based on the guidelines in EPRI Report NP-6628, "Procedure for Seismic Evaluation and Design of Small Bore Piping."

If the procedure for use of the equivalent static analysis as noted in Item b above is different from that described in Section 3.7.3.5 of the SSAR, revise Section 3.7.3.8.2.2 to provide a detailed description of the methodology to be used.

The staff is currently reviewing EPRI NP-6628 as a topical report, which was submitted to the staff by the Nuclear Management and Resources Council in a letter dated March 19, 1991. Pending completion of this review, the staff's position is that the methodology in this report is not acceptable. Revise Section 3.7.3.8.2.2 to remove the reference to EPRI NP-6628.

Response:

There are no differences between the equivalent static analyses described in Subsections 3.7.3.5 and 3.7.3.8.2.2. We believe the methodology presented in EPRI NP 6628 will be found acceptable by the NRC and should be included in the AP600 review and approval process.

SSAR Revision: NONE

Question 210.11

Clarify the discussion in Section 3.7.3.9 of the SSAR on the use of the independent support motion (ISM) method of modal analysis of piping systems to address the following concerns:

a. The proposed ISM method is inconsistent with the recommendations in Sections 2.3 and 2.4 of NUREG-1061, "Report of the USNRC Piping and Review Committee," Volume 4. Provide further technical justification. As a part of these recommendations, a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.

b. The damping values in Section 3.7.1 of the SSAR are referenced for use with the ISM method. This implies that the AP600 design incorporates the use of ASME Code Case (CC) N-411, "Alternate Damping Values for Response Spectra Analysis of Classes 1, 2, and Piping, Section III, Division 1" in conjunction with the ISM method. One of the conditions in RG 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1," relative to the use of CC N-411 is that the staff's acceptance of the use of the damping values in CC N-411 with the ISM method is pending further justification. Since the proposed ISM method is not in accordance with the recommendations in Item a above, provide further technical justification for this approach.

Response:

a. Technical justification for the proposed independent support motion (ISM) method is a comparison with test results as reported in EPRI NP-6153, "Seismic Analysis of Multiply Supported Piping Systems", Project 964-10, March, 1989. A support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.

b. ASME Code Case N-411 damping values will not be used with the ISM method. This satisfies the requirements of Regualatory Guide 1.84. Conformance to Regulatory Guides is addressed in SSAR Section 1.9. The damping values for piping systems that are analysed with the ISM method are 3% for piping larger than 12 inch diameter and 2% for smaller piping.

The following changes will be made to Subsection 3.7.3.9 of the SSAR:

SSAR Revision:

~~Subsection 3.7.1 presents the damping values.~~ A support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.The SSE damping values for piping systems that are analysed with the ISM method are 3% for piping larger than 12 inch diameter and 2% for smaller piping.

Question 210.15

The guidelines of Paragraph II.2 in Section 3.9.1 of the SRP state that a list of computer programs that will be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items, and the analyses to determine stresses should be provided. Provide such a list. Also, discuss the various programs' applicability and validity. At present, Section 3.9.1.2 of the SSAR only references the quality assurance program (as described in Chapter 17 of the SSAR) for this information.

Response:

The list of computer programs that are used for seismic Category I mechanical components is provided in Table 3.9-15 along with the application of each program. Other programs may also be used to complete the dynamic, static or stress analysis of these components. The computer programs will be listed in the ASME Code Design Report. The validation of each program is in accordance with an established quality assurance program and is available for NRC audit.

Subsection 3.9.1.2 of the SSAR will be revised and Table 3.9-15 added as follows:

SSAR Revision:

A number of computer programs that are used in the dynamic and static analyses of mechanical loads, stresses, and deformations of seismic Category I components and supports are listed in Table 3.9-15. A complete list of programs is included in the ASME Code Design Report.

The development process, verification, validation, configuration control and error reporting and resolution for computer programs used in these analyses for the AP600 are completed in compliance with an established quality assurance program. The quality assurance program is described in Chapter 17.

Table 3.9-15

Computer Programs for Seismic Category I Components

Program	Application
ABAQUS	Finite element structural analysis
ANSYS	Finite element structural analysis
CAEPIPE	Static analysis of piping systems
FATCON	ASME fatigue analysis of piping components
GAPPIPE	Static and dynamic analysis of piping systems
MAXTRAN	Transient stress evaluation of piping components
PIPSAN	Structural and ASME stress analysis of component supports
PS+CAEPIPE	Static and dynamic analysis of piping systems
STAAD-III	Static and dynamic analysis of structural frames
THERST	Transient heat transfer analysis of piping components
WECAN	Finite element structural analysis
WEGAP	Dynamic structural response of the reactor core
WECEVAL	ASME stress evaluation of mechanical components
WESTDYN	Static and dynamic analysis of piping systems

Question 220.15

Describe how penetrations and penetration reinforcements will be analyzed for buckling. The area-replacement rule may satisfy tensile strength requirements, but it does not necessarily satisfy buckling requirements (Section 3.8.2.4.2.5).

Response:

The penetrations and penetration reinforcements have been designed in accordance with the rules of ASME III, Subsection NE. References 220.15-1 thru 220.15-3 provide data which shows that penetrations with 80% or more of the area-reinforcement required for the internal pressure have the same buckling strength as an unpenetrated cylinder. Therefore, based on extensive experience with operating units and the results of in-house testing programs, no further buckling assessments of the penetrations and penetration reinforcements are deemed appropriate.

REFERENCES

220.15-1: C.D. Miller, "Experimental Study of the Buckling of Cylindrical Shells with Reinforced Openings," ASME/ANS Joint Conference, Portland, Oregon, July 26-28,1982.

220.15-2: C D. Miller, R.B. Grove, "Buckling of Cylindrical Shells with Reinforced Circular Openings under Axial Compression," Internal Report by Chicago Bridge & Iron Company, Plainfield, Illinois, March 14, 1980 (Letter Liparulo to Murley, ET-NRC-92-3778, December 2, 1992).

220.15-3: J.G. Bennett, R.C. Dove, T.A. Butler, "An Investigation of Buckling on Steel Cylinders in Circular Cutouts Reinforced in accordance with ASME Rules," Los Alamos Scientific Laboratory, LA-8853-MS, NUREG/CR-2165, June 1981.

SSAR Revision: NONE

Question 230.6

Section 3.7.1.2 of the SSAR states that the "TAFT" earthquake time history was used to generate synthetic time histories for AP600 seismic design. The SSAR presents spectrum comparison between the AP600 damped seismic design response spectra and the corresponding RG 1.60 response spectra anchored to 0.3 g for the damping ratios of 2, 3, 4, and 7% in Figures 3.7.1-6 through 3.7.1-8. However, the SSAR should also provide a spectrum comparison for the case with a damping ratio of 5%. Provide such a spectrum.

Response:

SSAR Figures 3.7.1-6 through 3.7.1-8 are revised to include the 5% damping response spectrum curves.

SSAR Revision:

(SSAR Figures 3.7.1-6 through 3.7.1-8 will be revised as shown in attached sheets.)

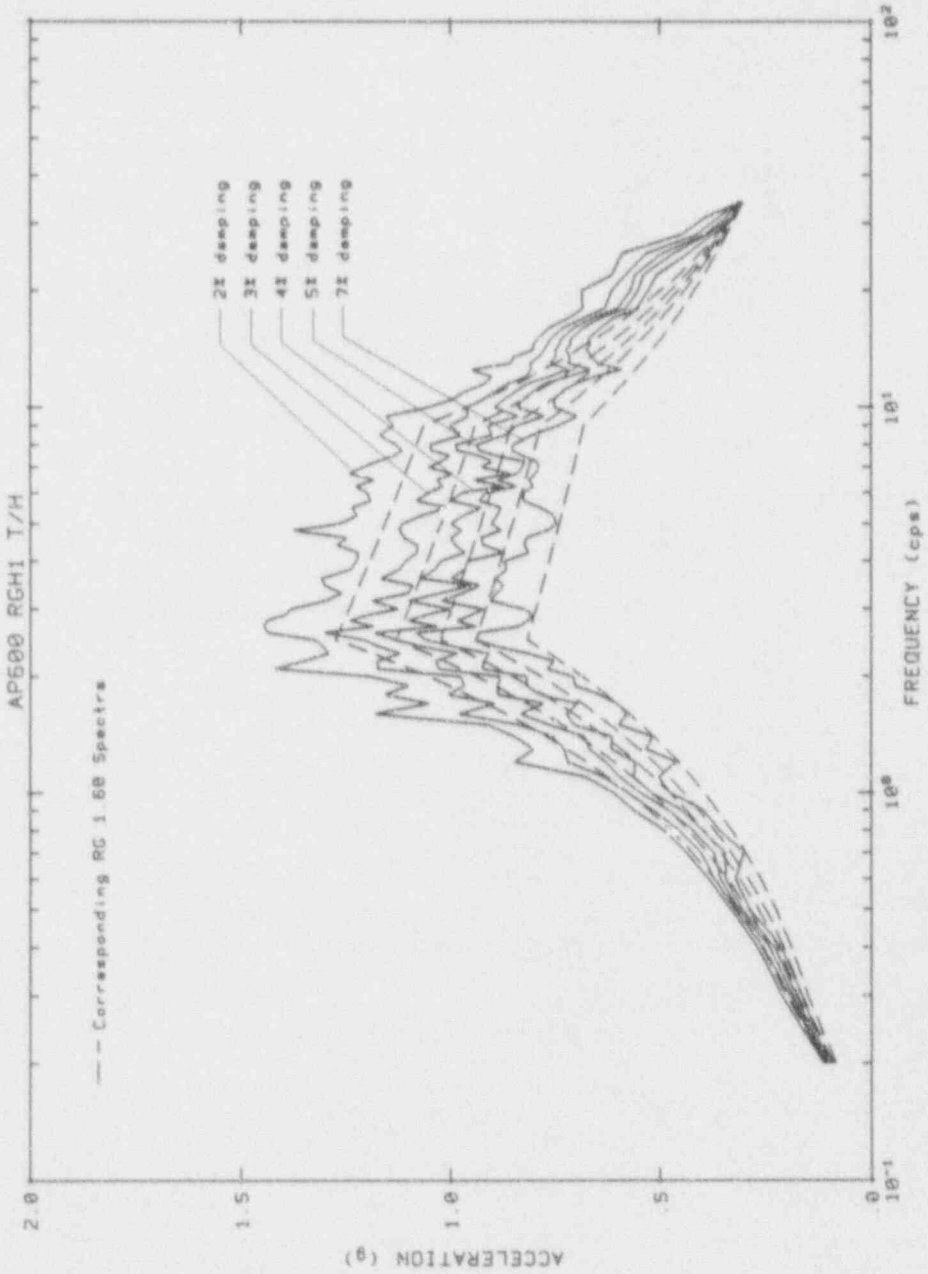

Figure 3.7.1-6 Acceleration Response Spectra of Design Horizontal Time History, "H1"

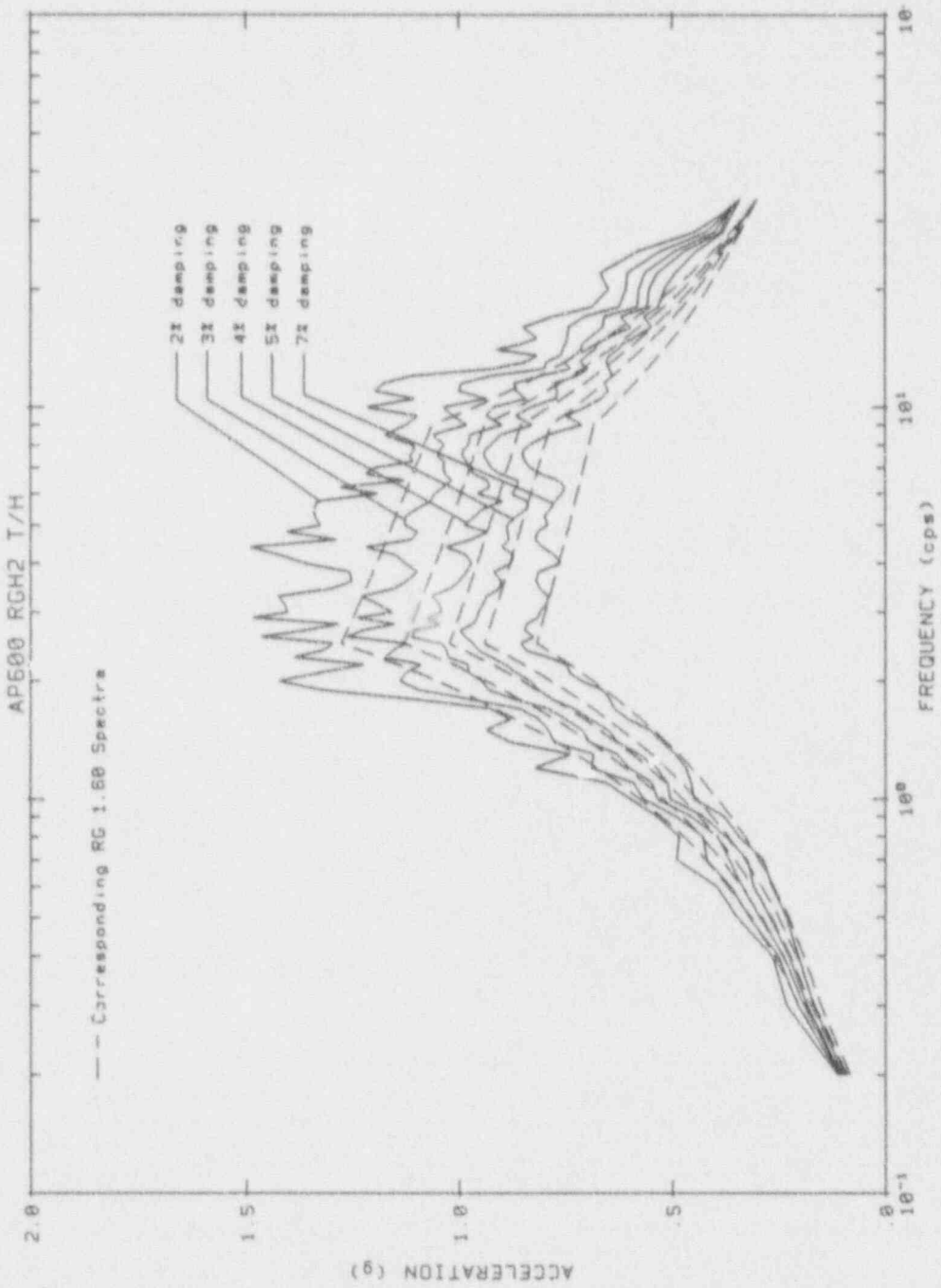

Figure 3.7.1-7 Acceleration Response Spectra of Design Horizontal Time History, "H2"

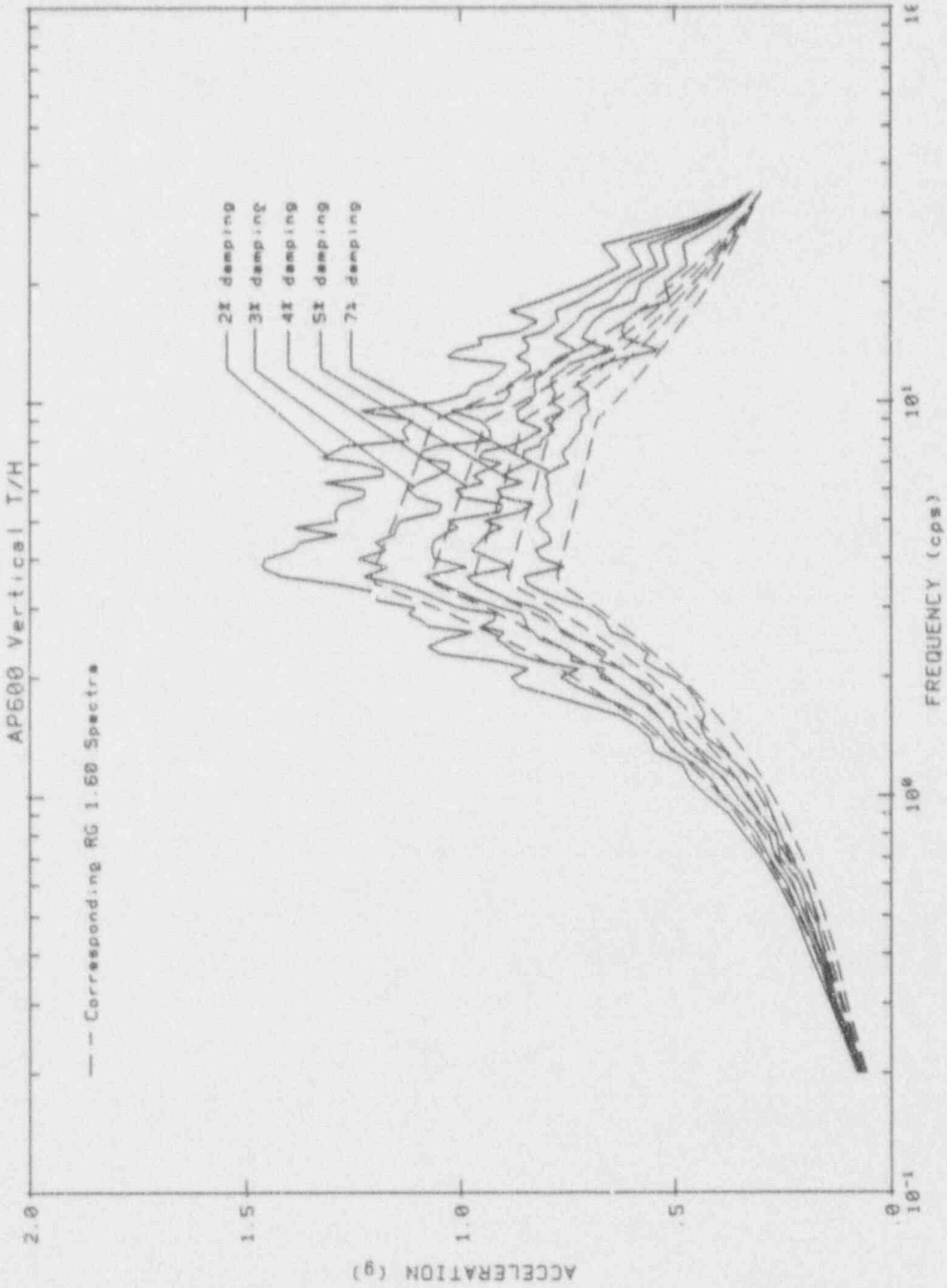

Figure 3.7.1-8 Acceleration Response Spectra of Design Vertical Time History

Question 250.9

Table IWB-2500-1 in Section XI of the ASME Code requires the examination of Class 1 piping welds, with a Section III fatigue cumulative usage factor (CUF) exceeding 0.4, at every inspection interval. Confirm that the value of CUF to be used will correspond to the projected 60-year plant design life (Section 5.2.4).

Response:

The Section III cumulative usage factor is calculated on the basis of a 60-year plant design life.

SSAR Revision: NONE

Question 250.11

Figure 5.4-2 in the SSAR does not show the orientation and location of all of the access points in the steam generator. Provide drawings to show the secondary side access points in the steam generator.

Response:

There are six 6 inch handholes located in the lower part secondary shell. There are four 90° apart (two on the tubelane and two at 90° from the tube lane) that provide access to the secondary face of the tubesheet. The other two are 180°apart on the tubelane and provide access to the top of the flow distribution baffle. Two 4 inch ports located on the secondary shell in line with the tubelane and above the top tube support plate provide access to the U-Bend area. Please see the answer to RAI 250.12.

Figure 5.4-2 will be updated to show the location. The last paragraph of SSAR Subsection 5.4.2.4.2 will be revised to reflect this response as follows:

SSAR Revision:

Several methods can be used to clean operating steam generators of secondary-side deposits. Sludge lancing is a procedure in which a hydraulic jet inserted through an access opening (handhole) loosens deposits and the loose material is flushed out of the steam generator. Six 6 inch access ports are provided for sludge lancing. Four of these are located above the tubesheet 90° apart (two on the tubelane and two at 90° from the tube lane) to provide access to the secondary face of the tubesheet, and two are 180°apart on the tubelane and provide access to the top of ~~above~~ the flow distribution baffle. Also two 4 inch ports located on the secondary shell in line with the tubelane and above the top tube support plate provide access to the U-Bend area. A blowdown pipe is provided to permit continuous blowdown and monitoring of secondary water chemistry. The blowdown piping suction is adjacent to the tubesheet and in a region of relatively low-flow velocity. This facilitates the removal of particulate impurities to reduce the accumulation on the tubesheet. The materials of the secondary side of the steam generator are also compatible with chemical cleaning.

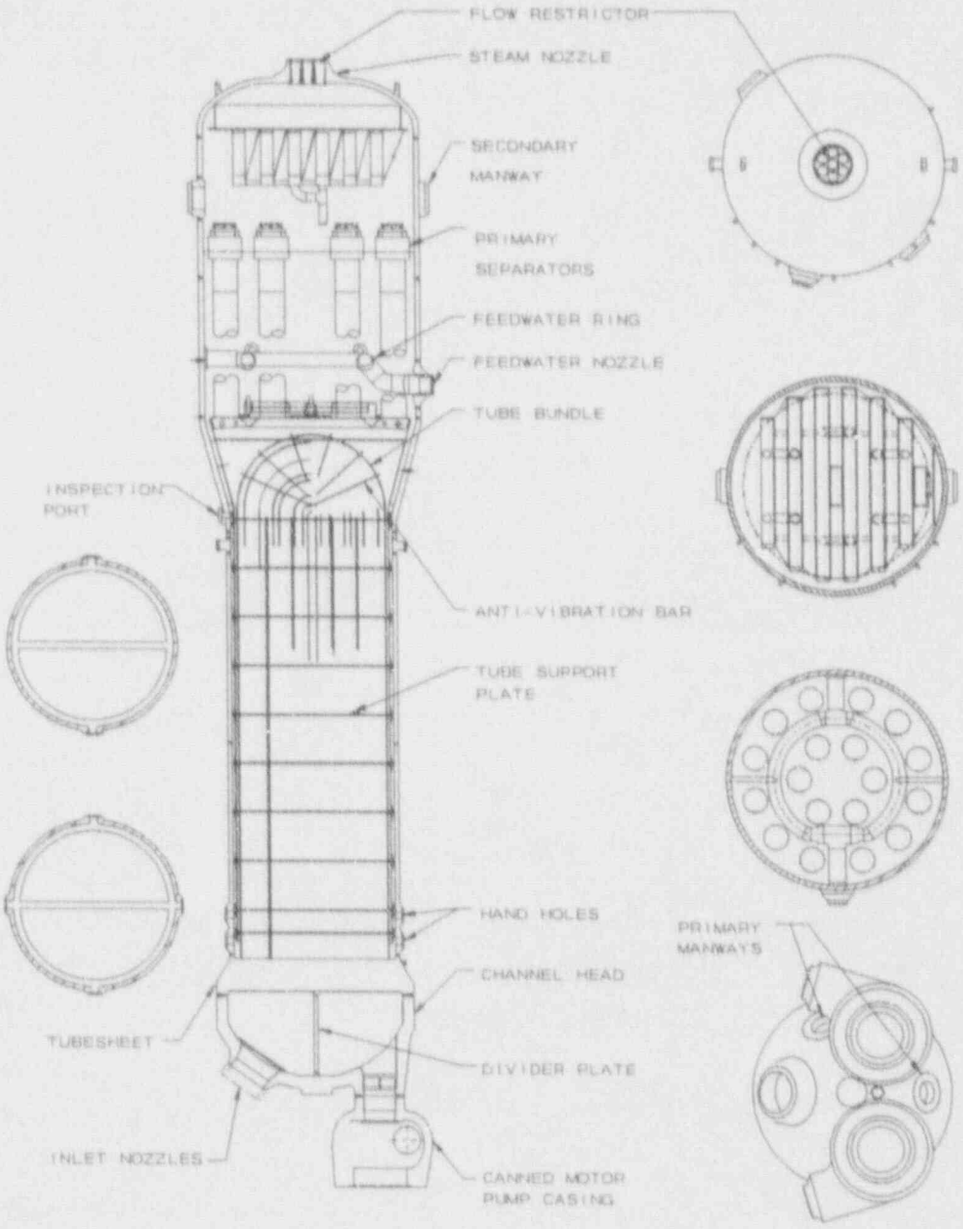

Figure 5.4-2 Steam Generator

Question 250.12

Discuss whether the four 15 cm (6 in) handholes located just above the tubesheet are of sufficient size to allow for effective sludge lancing, retrieval of loose parts, and/or inspection of the tube bundle by portable inspection equipment (e.g. video equipment) (Section 5.4.2).

Response:

Two of the four handholes just above the tubesheet are the same size and in the same relative location as the handholes in the Westinghouse designed Model F steam generators. Two additional handholes are provided at approximately 90° from the ends of the tubelane. The experience in operating plants has demonstrated that these handholes permit effective sludge lancing, retrieval of loose objects, and inspection of the tube bundle. Two 4 inch ports located on the secondary shell in line with the tubelane and above the top tube support plate provide access to the U-Bend area. Please see the response to RAI 250.11.

The last paragraph of SSAR Subsection 5.4.2.4.2 will be revised to reflect this response as follows:

SSAR Revision

Several methods can be used to clean operating steam generators of secondary-side deposits. Sludge lancing is a procedure in which a hydraulic jet inserted through an access opening (handhole) loosens deposits and the loose material is flushed out of the steam generator. Six 6 inch access ports are provided for sludge lancing, inspection of the tube bundle by portable inspection equipment, and retrieval of loose objects. Four of these are located above the tubesheet, and two are above the flow distribution baffle. A blowdown pipe is provided to permit continuous blowdown and monitoring of secondary water chemistry. The blowdown piping suction is adjacent to the tubesheet and in a region of relatively low-flow velocity. This facilitates the removal of particulate impurities to reduce the accumulation on the tubesheet. The materials of the secondary side of the steam generator are also compatible with chemical cleaning.

Question 250.13

Describe the design provisions for tube indexing for facilitation of tube identification and location during inservice inspections (Section 5.4.2).

Response:

Controls for the automated, robotic equipment typically used for tube inspection and repair activities deliver the inspection and service tooling to the proper tube location without the need for any visual numbering or other identification of the tubes. However, to facilitate tube identification for manual activities, if needed, the tube location for a large fraction of the tubes is scribed on the tubesheet adjacent to the tube. The scribing is done using laser scribing or other method chosen to minimize residual stress in the tubesheet cladding.

The fourth paragraph of SSAR Subsection 5.4.2.5 will be revised to reflect this response as follows:

SSAR Revision:

The steam generators are designed to permit access to tubes for inspection and/or repair or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. The AP600 steam generator includes features to enhance robotics inspection of steam generator tubes without manned entry of the channel head. These include a cylindrical section of the channel head and remote installation of nozzle dams. To facilitate tube identification for manual activities, the tube location for a large fraction of the tubes is scribed on the tubesheet.

Question 250.14

Describe the physical location of the internal deck plates used to gain access to the U-bend area. Clarify the statement in Section 5.4.2.5 of the SSAR that "for proper functioning of the steam generator, some of the deck-plate openings are covered with welded but removable, hatch plates."

Response:

The deck plate through which access is required to reach the steam generator tube U-bends is the deck plate located at the base of the primary separators at the top of the tube bundle wrapper. The deck plate openings in this deck plate may be opened by grinding off the welds retaining the hatch plates in place. Routine inspection requiring access through these hatches is not expected.

The second paragraph of SSAR Subsection 5.4.2.6 will be revised to reflect this response as follows:

SSAR Revision:

The design includes a number of openings to provide access to both the primary and secondary sides of the steam generator. The openings include four 18-inch diameter manways, one for access to each chamber of the reactor coolant channel head and two in the steam drum for inspection and maintenance of the upper shell internals. In addition, six 6-inch diameter handholes in the shell, four located just above the tubesheet secondary surface, and two located just above the flow distribution baffle, are provided. Two 4-inch diameter inspection openings are provided at each end of the tubelane between the upper tube support plate and the row 1 tubes. Additional access to the tube bundle U-bend is provided through ~~each of~~ the internal deck plates at the bottom of the primary separators. For proper functioning of the steam generator, some of the deck-plate openings are covered with hatch plates welded inplace that are ~~, but~~ removable by grinding, gouging, or other methods to cut off the welds.

Question 250.15

Describe the features incorporated in the design that enhance inspection of the steam generator tubes without manned entry. Discuss whether the design features support the use of current robotic equipment used in steam generator tube inspection and repair. In addition, discuss whether verification have been performed, by computer simulation and/or mockup, to ensure that the design will facilitate not only the use of robotic manipulators in inspecting all of the tubes within the steam generator but also in inserting the robotics into the steam generator (Section 5.4.2).

Response:

The cylindrical portion of the channel head just below the tubesheet facilitates the use of robotically delivered inspection and repair tooling to tube locations on the periphery of the tube bundle. The ability to reach all locations has been verified using computer simulations. The channel head and primary inlet and outlet nozzles have provisions to facilitate the robotic installation of nozzle dams. The use of 18 inch diameter manway openings provides that any equipment that is used in operating steam generators can be used in the AP600 steam generator.

The fourth paragraph of SSAR Subsection 5.4.2.5 will be revised to reflect this response as follows:

SSAR Revision:

The steam generators ~~are designed to~~ permit access to tubes for inspection, ~~and/or~~ repair, or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. The AP600 steam generator includes features to enhance robotics inspection of steam generator tubes without manned entry of the channel head. These include a cylindrical section of the channel head, 18 inch diameter primary manways, and provisions to facilitate the remote installation of nozzle dams. Computer simulation using designs of existing robotically delivered inspection and maintenance equipment verifies that tubes can be accessed.

Question 250.17

Provide clarification on what it considers "more capable equipment" or a "suitable eddy current inspection system" as compared with the equipment described in paragraph C.2.c of Regulatory Guide 1.83 (Section 5.4.2).

Response:

Eddy current inspection equipment currently in use typically includes multi-channel capability, software to screen indications, software to provide graphical representations of associated tube degradation and recording on media readable on personal computers or workstations. Strip charts and magnetic tape data recorders are typically not used and optical disk storage media may be used. As new equipment and methods are developed for inservice eddy current testing this equipment and testing may be used for baseline inspections. Any eddy current inspection performed in the manufacturing facility is conducted by personnel qualified to the requirements for inspectors performing inservice inspection of operating units. The manufacturing facility inspection is conducted using the same equipment as, or equipment similar to, that used during inservice inspection of operating units.

(See the response to RAI 250.16 for changes to the exceptions to Regulatory Guide 1.83 found in SSAR Appendix 1A.)

The third paragraph of SSAR Subsection 5.4.2.5 will be revised to reflect this response as follows:

SSAR Revision:

Regulatory Guide 1.83 provides recommendations on the inspection of tubes. The recommendations cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, and required actions based on findings. Any eddy current inspection performed in the manufacturing facility is conducted by personnel qualified to the requirements for inspectors performing inservice inspection of operating units. The manufacturing facility inspection is conducted using the same equipment as, or equipment similar to, that used during inservice inspection of operating units. Exceptions to Regulatory Guide 1.83 are noted in Subsection 1.9.1.

Question 250.18

Describe the corrective measures that will be implemented to disposition leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limits (Section 5.4.2).

Response:

The determination of the corrective measures to be used to disposition defective or degraded tubes is the responsibility of the combined license holder at the time such degradation or defects are observed. Tubes with eddy current indications in excess of the repair limit may be removed from service by the installation of mechanical tube plugs, by the installation of welded plugs meeting the requirements of the ASME Code, Section XI, IWB-4200, by the installation of tube sleeves meeting the requirements of ASME Code, Section XI, IWB-4300 or other tube repair methods authorized by the NRC. The AP600 steam generator accommodates current and anticipated repair methods and techniques. The tube repair criteria establishing the level at which degraded tube must be plugged or repaired is to be provided by the combined license applicant considering NRC requirements and industry recommendations. Please also see the response to RAI 250.19.

The fourth paragraph of SSAR Subsection 5.4.2.5 will be revised to reflect this response as follows:

SSAR Revision:

The steam generators are designed to permit access to tubes for inspection and/or repair or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. Tooling to install mechanical and welded plugs, tube repair sleeves, or effect other repair processes remotely can be delivered robotically. The AP600 steam generator includes features to enhance robotics inspection of steam generator tubes without manned entry of the channel head. These include a cylindrical section of the channel head and remote installation of nozzle dams.

Question 250.19

Provide clarification to exceptions to criteria C.2.a.(2) and C.2.a.(4) of Regulatory Guide 1.121. In particular, describe how the proposed change will affect the margin of safety currently observed. Describe the statistical analysis of the tensile test data that is used in the development of the expected material strength properties. Also, discuss whether the calculation of the tube minimum wall requirements will be based on the lowest values for the material properties, i.e., the lowest values from statistical analyses or from the ASME Code (Section 5.4.2).

Response:

The exceptions noted in Appendix 1A for Positions C.2.a.(2) and C.2.a.(4) are inconsistent with the methods used to develop tube plugging criteria for currently operating plants. Please note that Regulatory Guide 1.121 is used to address degradation of steam generator tubes in units that have entered service and has typically not been used to establish the basis for tube plugging criteria prior to operation. The tube repair criteria establishing the level at which a degraded tube must be plugged or repaired is to be provided by the combined license applicant considering NRC requirements and industry recommendations. Since the use of the recommendations of Regulatory Guide 1.121 to establish tube repair criteria during the Design Certification process is inappropriate, reference to the Regulatory Guide should be deleted.

The last paragraph of SSAR Subsection 5.4.2.5 will be revised to reflect this response as follows:

SSAR Revision:

The minimum requirements for in-service inspection of steam generators, including tube repair ~~plugging~~ criteria, is the responsibility of the combined license applicant considering NRC requirements and industry recommendations. ~~are established as part of the Technical Specifications. These requirements are consistent with the ASME Code, Section XI, and Regulatory Guide 1.121~~. Section XI of the ASME Code provides general acceptance criteria for indications of tube degradation in the steam generator. ~~Specific conformance with Regulatory Guide 1.121 is discussed in Section 1.9.~~

Question 250.20

Provide technical justifications for exceptions to criteria C.2.a.(5)-(6) and C.2.b of Regulatory Guide 1.121 (Section 5.4.2).

Response:

The AP600 position provided in Appendix 1A for Regulatory Guide 1.121 Regulatory Positions C.2.a.(5)-(6) should be "Conforms". The AP600 position for C.2.b. is that given for C.2.a.(5)-(6).

SSAR Appendix 1A and the last paragraph of SSAR Subsection 5.4.2.5 will be revised as follows:
(See the response to RAI 250.19 for related SSAR changes.)

SSAR Revision:

(Appendix 1A)

Reg. Guide 1.121, Rev. 0, 8/76 - Bases for Plugging Degraded PWR Steam Generator Tubes

C.2.a.(5)-(6)	~~Exception~~ Conforms	~~In cases where sufficient inspection data exist to establish degradation allowance, the rate used is an average time-rate determined from the mean of the test data. Where requirements for minimum wall are markedly different for various areas of the tube bundle, such as the U-bend area versus straight length in Westinghouse designs, separate plugging limits are established to address the varying requirements in a manner which does not require unnecessary plugging of tubes.~~
C.2..b.	Exception	In cases where sufficient inspection data exist to establish degradation allowance, the rate used is an average time-rate determined from the mean of the test data. Where requirements for minimum wall are markedly different for various areas of the tube bundle, such as the U-bend area versus straight length in Westinghouse designs, separate plugging limits are established to address the varying requirements in a manner which does not require unnecessary plugging of tubes.

(Subsection 5.4.2.5)

The minimum requirements for in-service inspection of steam generators, including tube plugging criteria, are established as part of the Technical Specifications. These requirements are consistent with the ASME Code, Section XI~~, and Regulatory Guide 1.121~~. Section XI of the ASME Code provides general acceptance criteria for indications of tube degradation in the steam generator. ~~Specific conformance with Regulatory Guide 1.121 is discussed in Section 1.9.~~

Question 251.32

Discuss conformance with guidance in Acceptance Criteria II.4.a, II.4.b, II.4.c, II.4.d, and II.4.e in Section 10.2.3 of the SRP (Section 10.2.3).

Response:

The referenced acceptance criteria are stated to apply to built-up turbine rotors. The rotors that would be supplied are fully integral (FI) and the following responses are applicable to FI rotors.

a. The design overspeed of the rotors is 120 percent of normal speed. The maximum anticipated overspeed, in the event of a trip from full load is 111 percent.

b. The combined stress is less than 50 percent of the bore yield strength (See response to RAI 251.27).

c. Turbine shaft bearings are sized to limit the normal gravity load pressures and load additions resulting from misalignments due to thermal effects in the foundation and vacuum loading to conservative levels based on actual bearing tests. The turbine shaft bearings do not experience any abnormal loads due to the transients or accidents as discussed in the RAI questions.

d. The natural critical frequencies are controlled to be removed from normal running speed by approximately 15 percent. The system is evaluated for sharpness of resonance response if separation margin not attained. Adequate model damping assures controllable response amplitudes through proper balancing procedures.

e. The FI rotor does not have any keyways. The rotor may be examined from the bore by existing ultrasonic technique techniques. This will enable full volume coverage of all highly stressed zones.

SSAR Revision: NONE

Question 252.3

Perform bounding LBB analyses for each of the LBB candidate piping, including evaluations for susceptibility to potential degradation mechanisms for the projected 60-year plant design life. Provide the analyses (Section 3.6.3).

Response:

Bounding leak-before-break analysis is not performed. Sample leak-before-break analyses are provided in Appendix 3B for the reactor coolant loop piping. The NRC staff should be able to assess the acceptability of the AP600 leak-before-break approach based on this sample and the criteria in Subsection 3.6.3. The following additional sample calculations will be performed by December, 1993.

Pipe Stress Analysis Sample Problems for Piping Evaluated to Leak-Before-Break

P&ID number	Description	Diameter(in)
RCS-M6-001	Primary coolant loops (1 and 2)	22 and 31
RCS-M6-002	Automatic Depressurization stage 1, 2, and 3 (A and B)	4, 8, and 14
SGS-M6-001	Main steam (1 and 2)	32
SGS-M6-001	Main feedwater (1 and 2)	16

The entire scope of candidate leak-before-break piping lines is provided in the response to RAI 210.6.

SSAR Revision: NONE

Question 252.4

Describe the procedures to be used by the COL applicant to verify that the actual material properties and final, as-built piping analyses are within the limits in the bounding LBB analyses (Section 3.6.3).

Response:

Bounding leak-before-break analysis is not performed. The leak-before break analysis is performed prior to construction and is based on the as-designed piping analyses and representative material properties. A report will be prepared by the COL applicant to reconcile the design analysis. This report will include a review of the as-built piping analyses and the Certified Material Test Reports that are associated with those piping systems that are qualified to the leak-before-break criteria. The procedure to develop this report is beyond the scope of design certification.

SSAR Revision: NONE

Question 252.5

Section 3.6.3 of the SSAR indicates that Class 2 and 3 piping are within the LBB scope. The staff has not approved the application of LBB for these piping for operating reactors. There are differences in ASME Code requirements between Class 1 and Class 2 and 3 piping. Discuss the significance of these differences on ensuring piping structural integrity and describe procedures to address them.

For example, the ASME Code does not require a fatigue analysis for Class 2 and 3 piping. Discuss how the fatigue resistance of the LBB Class 2 and 3 piping will be addressed. As another example, the in service inspection requirements for Class 2 piping is based on a sampling basis and Class 3 piping is based on visual inspections. Discuss any augmented in service inspection for Class 2 and 3 LBB piping.

Response:

The leak-before-break methodology is not applied to ASME Class 3 piping as presented in response to RAI 210.6. This methodology is applied to Class 2 portions of the main steam and feedwater piping inside containment. The main difference between the ASME Section III stress analysis requirements for Class 1 and Class 2 piping is that there is no cumulative fatigue damage calculation for Class 2 systems.The fatigue resistance of the Class 2 piping which meets the lea[illegible]-to-break criteria is verified by performing fatigue crack growth calculations for postulated part through-wall fl[illegible] The welds selected for examination as part of the ASME Section XI in service inspections will include a termina[illegible] weld at the steam generator nozzle for one main steam line and one main feedwater line. The terminal end locations typically hav[illegible] thermal and SSE stresses than other locations and are generally limiting for leak-before-break evaluatio[illegible] ne size of the sample and the frequency of inspection will not be increased beyond that required by the ASME Code.

SSAR Subsection 3.6.3.2 will be revised as follows:

SSAR Revision:

- ~~For ASME Code, Section III, Class 3 designed piping and piping not designed to ASME Code, Section III, the piping, supports, and structures are designed for the safe shutdown earthquake event. The pre-service and in-service inspection requirements for ASME Code, Section III, Class 2 piping are met.~~

- For ASME Class 2 piping a fatigue crack growth analysis is performed to verify the fatigue resistance of the piping system. In addition, the welds selected for examination for the ASME Section XI in-service inspection will include two terminal end welds to the steam generator nozzles: one for the main steam piping and one for the main feedwater piping.

Question 252.6

Section 3.6.3 of the SSAR indicates that LBB may be applied for portions of piping outside containment. Provide information to demonstrate the reliability, effectiveness, sensitivity, and timeliness of leakage detection methods and procedures selected for outside containment.

Response:

The leak-before-break methodology is not applied to piping outside of containment. This methodology is applied to piping systems that extend into the break exclusion zone of the main steam tunnel outside containment. This applies to the main steam and main feedwater piping as follows. The main steam piping from the steam generator outlet nozzle to the anchor downstream of the isolation valve is analyzed for applicable loadings including the SSE. This anchor is at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements of Standard Review Plan 3.6.2. The portion of this piping from the steam generator outlet nozzle to flued head inboard weld is evaluated to the leak-before-break methodology. The main feedwater piping from the steam generator inlet nozzle to the anchor upstream of the isolation valve is analyzed for applicable loadings including the SSE. This anchor is also located at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements of Standard Review Plan 3.6.2. The portion of the piping from the steam generator inlet nozzle to the flued head inboard weld is evaluated to the leak-before-break methodology.

SSAR Subsections 3.6.3 and 3.6.3.1 will be revised as follows:

SSAR Revision:

(Subsection 3.6.3, paragraph 7)

~~High-energy ASME Code, Section III, Class 1, 2, and 3 piping of four-inch nominal diameter or larger is evaluated for compliance with leak-before-break criteria. For piping that penetrates the containment vessel, the evaluation is continued to the first anchor outside containment and includes any branch connections between the penetration and anchor. For those piping systems or portions of systems for which it is not practical or economical to satisfy the mechanistic pipe break criteria, the requirements and criteria discussed in Subsections 3.6.1 and 3.6.2 for the analysis and location of postulated pipe ruptures apply.~~

High-energy ASME Code Section III piping that is evaluated to the leak-before-break methodology is identified in Appendix 3B. This applies to the main steam and main feedwater piping as follows. The main steam piping from the steam generator outlet nozzle to the anchor downstream of the isolation valve is analyzed for applicable loadings including the SSE. This anchor is at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements described in Subsection 3.6.2. The portion of this piping from the steam generator outlet nozzle to flued head inboard weld is evaluated to the leak-before-break methodology. The main feedwater piping from the steam generator inlet nozzle to the anchor upstream of the isolation valve is analyzed for applicable loadings

including the SSE. This anchor is also located at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements described in subsection 3.6.2. The portion of the piping from the steam generator inlet nozzle to the flued head inboard weld is evaluated to the leak-before-break methodology. High energy piping that does not satisfy the leak-before-break criteria is designed to the requiremements discussed in Subsections 3.6.1 and 3.6.2.

(Subsection 3.6.3.1, paragraph 9)

~~The feedwater and steam piping inside containment to the first anchor outside containment is also designed to the criteria developed to provide leak-before-break characteristics in high energy lines.~~

(Subsection 3.6.3.1, paragraph 15)

~~Outside containment, visual observation or local instrumentation is used to provide detection equivalent to Regulatory Guide 1.45. Outside containment monitoring of the main steam and feedwater lines is supplemented by periodic inspection and humidity measurements in the areas adjacent to the lines and the isolation valves.~~

Question 252.10

Section 3B of the SSAR discusses the LBB evaluation for the reactor coolant loop piping. The SSAR indicates that two different soil conditions have been considered in deriving piping stresses. Discuss how these piping stresses represent the worst condition of all potential sites within the scope of AP600 applications.

Response:

As described in Appendix 3B of the SSAR, two soil conditions for the reactor coolant loop pipe stress analyses were chosen to provide preliminary stresses for a sample of the application of the leak-before-break methodology. Analyses for other soil conditions will be performed by December,1993. These loop analyses will represent the worst condition of all potential sites within the scope of AP600 applications.

SSAR Revision: NONE

Question 252.11

Tables 3B-3 and 3B-4 of the SSAR give stresses used in the LBB evaluation of the reactor coolant loop piping. Provide information to clarify whether the stresses are from the stress analysis of routed or unrouted reactor coolant loop piping.

Response:

The stress analysis for the reactor coolant piping is included in the SSAR Appendix 3B to provide a sample for the application of the leak-before-break methodology. This analysis is based on routed reactor coolant piping which is supported by the primary equipment supports; the connecting piping (e.g. surgeline) is not included in the model.

SSAR Revision: NONE

Question 252.47

Discuss whether the carbon content of austenitic stainless steel in the reactor internals and core support structures will be limited to less than 0.02% as recommended in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," January 1988. Provide a technical discussion on why this limit is not relevant to the proposed use if it is not used (Section 4.5.2).

Response:

The carbon content of austenitic stainless steel will not be limited to less than 0.02%C in the reactor internals and core support structures. NUREG-0313 is concerned with Boiling Water Reactors where presence of oxygen, from radiolytic decomposition of water in the core, can lead to IGSCC. EPRI NP-6780-L recommends a limit of 0.035% as suitable for Pressurized Water Reactors where the presence of oxygen is suppressed by the hydrogen overpressure. The EPRI industry guidance will be followed where L or LN grades are to be used.

SSAR Revision: NONE

Question 252.60

Section 5.2.3.4.3 of the SSAR indicates that there may be inaccessible cavities or chambers in the RCPB. Discuss considerations to eliminate these conditions. If these conditions cannot be avoided, provide accesses for future inservice inspection to monitor the conditions in these cavities or chambers. Discuss the associated augmented inservice inspection program.

Response:

The discussion of inaccessible cavities and chambers is related to the requirement for a test for sensitization for stainless steel parts. Those components that are not simple shapes are subject to the test. A configuration that precludes rapid cooling of a part when water quenched necessarily preclude or complicate the subsequent inservice inspection of the pressure boundary. As noted in 5.2.4.2, the Class 1 components can be inspected per the requirements of the ASME Code, Section XI. The first paragraph of 5.2.3.4.3 will be revised to reflect this response as follows:

SSAR Revision:

Austenitic stainless steel materials of product forms with simple shapes need not be corrosion-tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as plates, sheets, bars, pipe, and tubes, as well as forgings, fittings, and other shaped products that do not have inaccessible cavities or chambers that would preclude rapid cooling when water-quenched. This characterization of cavities or chambers as inaccessible is in relation to the entry of water during quenching and is not a determination of the component accessibility for inservice inspection.

Question 252.107

The proposed new steam generator tube plugging criteria in Section 5.4.2 of the SSAR would place increased emphasis for steam generator integrity on primary to secondary leakage monitoring relying on increased sensitivity and on-line real time read-outs. Describe Westinghouse's proposed plans on implementing this monitoring.

Response:

The tube repair criteria and inservice inspection program is the responsibility of the Combined License holder. The SSAR does not delineate a tube repair criteria that would rely on leak monitoring. The inservice inspection program is expected to follow the industry guidelines for steam generator tube inspection contained in the EPRI steam generator examination guidelines report. Please also see RAI Questions 250.10 and 250.21 for a discussion of the inservice inspection program. See response to RAI 250.19 for a discussion of the development of tube repair criteria.

The AP600 has radiation monitoring capability in the steam generator blowdown, main steam lines, and condenser air removal discharge to detect radiation due to primary to secondary side leakage. The radiation monitoring capability for the secondary system in the AP600 is in conformance with industry requirements for advanced light water reactors. The monitoring system is in conformance with the recommendations in NRC Information Notices 88-99 and 91-43 to provide for the detection of increases in primary-to-secondary leak rate. The radiation monitoring system is discussed in Subsection 11.5 of the SSAR.

SSAR Response: NONE

Question 252.108

Describe how the "Delta-75" steam generator design proposed for the AP600 will facilitate the implementation of in-situ fusion techniques for steam generator tube repair. Also, discuss how the selection of materials for the tube support structures and the tubesheet will preclude deleterious effects on material toughness caused by in-situ fusion heat effects (Section 5.4.2).

Response:

The steam generator design features provide for tube repair with automated equipment without manned entry. The repair equipment and the robotic delivery equipment that has been developed for use in operating steam generators can be used in the AP600 steam generator. Please also see the response to RAI 250.15. The steam generator uses materials for the tubesheet and tube supports plates that have been used in previous steam generators in the same applications. With respect to the use of repair techniques the material selection and design of the steam generator for the AP600 is the same as or similar to previous steam generator designs. Specific testing of the effects on tubesheet and tube support plate materials to support the application of in-situ fusion techniques for steam generator tube repair in the AP600 steam generator is not required.

See RAI 250.15 for suggested SSAR changes related to provisions for the use of robotic equipment for steam generator repair.

SSAR Revision: NONE

Question 252.109

Section 5.4.2.3.3 of the SSAR indicates that tube vibration has potential to cause wear. Discuss in detail the potential for wear degradation with emphasis on the AP600 features that are designed to mitigate this concern.

Response:

The potential for tube wear is dependent on the amplitude of vibration of the tube, the material couple, and the configuration and surface condition of the supports being impacted by the tubes. The amplitude of vibration is determined, in part, by the size of the gaps between the supports and the spacing between supports. The AP600 steam generator includes a number of features that minimize the potential for tube wear at tube supports and antivibration bars. Provisions to minimize the potential for wear include the spacing between the tube supports, the configuration of the broached hole through the support plate, the clearance between the tube and the hole in the tube support plate, tube support plate material selection, and the configuration of the anti-vibration bar assemblies.

The seventh paragraph of SSAR Subsection 5.4.2.3.3 will be revised to reflect this response as follows:

SSAR Revision:

Tube vibration response is shown to have wear potential within available design margins even for limiting tube fit-up conditions, based on pervious experience in fabricating steam generators with fit-up control typical of the AP600 steam generator. The AP600 steam generator includes a number of features that minimize the potential for tube wear at tube supports and antivibration bars. Provisions to minimize the potential for wear include the spacing between the tube supports, the configuration of the broached hole through the support plate, the surface finish of the broached hole in the tube support plate, the clearance between the tube and the hole in the tube support plate, tube support plate material selection, and the configuration of the anti-vibration bar assemblies.

~~Corresponding~~ Tube bending stresses corresponding to tube vibration response remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. These analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

Question 252.111

Recent plant operating experience disclosed the possibility of miss-placed anti-vibration bars (AVBs) and the possible severe consequences. Discuss how the proper location of AVBs will be ensured (Section 5.4.2).

Response:

The potential for misplaced anti-vibration bars is minimized by an in-process dimensional inspection of the tube U-bends and the anti-vibration bars. As discussed in NRC Bulletin 88-02, misplaced anti-vibration bars have resulted in adverse consequences only in conjunction with dented or corrosion packed top tube support plates. The AP600 steam generator is not expected to be subject to these conditions due to the design and material selection of the support plates.

The last paragraph of SSAR Subsection 5.4.2.3.3 will be revised to reflect this response as follows:

SSAR Revision:

The U-bend fatigue (discussed in NRC Bulletin 88-02) is not a consideration in the AP600 steam generators. The mechanism considered in Bulletin 88-02 requires denting of the top tube support plate. But this is not expected with the stainless steel tube support plates in the AP600 steam generator. Additionally, the location of anti-vibration bars is controlled by in-process dimensional inspection.

Question 252.112

Industry recommendations and other vendors' improved steam generators designs incorporate primary side manways having a minimum inner diameter of 53 cm (21 in). Discuss Westinghouse's technical basis for limiting the ports in the "Delta-75" steam generator to 46 cm (18 in) in diameter as indicated in Section 5.4.2.5 of the SSAR.

Response:

The function of primary manways is to provide access to the channel head for inspection and repair, if necessary. The steam generator provides full access to the tubes with automated, robotically delivered inspection and repair equipment. Use of this equipment minimizes or eliminates manned entry into the channel head for routine inspection and repair activies. The 18 inch diameter manway is large enough for use of the inspection and repair equipment. A larger diameter manway opening would increase the radiation levels in the areas adjacent to the manway when open. The 16 inch manways in operating steam generators have been found to be adequate, if not optimum, to permit entry into the channel head. The 18 inch diameter manway size is based on considerations that include access, space available, stress levels in the channel head, size of closure fasteners, occupational radiation exposure, and handling of covers.

SSAR Revision: NONE

Question 252.113

Experience has shown the advisability of complete records and archive materials to investigate corrosion and mechanical damage which may occur during service. Industry recommendations suggest archiving at least 2 m (7 ft) of each heat of row 1 and 2 "U-bends" prior to final heat treatment and following the mill anneal, and production samples containing tubes from each heat expanded in a tube sheet mockup. Archive samples should be maintained to support future chemical cleaning programs and for possible defect calibration samples for inservice inspection. Describe Westinghouse's program to retain records and archive materials (Section 5.4.2).

Response:

The heat and lot of tubing material for each steam generator tube is recorded and documented. Archive samples of each heat and lot of steam generator tubing material are provided to the combined license holder for use in future materials testing programs or inservice inspection calibration standards. The archive samples are subject to the same manufacturing processes and inspections as the installed tubing. A minimum of seven feet of tubing in the final heat treat condition is supplied.

The archive samples to be provided do not include tubes expanded into tubesheet mockups. The archive samples provided are consistent with the ALWR utility requirements for advanced light water reactors and EPRI guidelines for PWR steam generator tubing specifications (EPRI Report NP 6743-L) which do not include a requirement for tubesheet mockup archive samples. Tubesheet mockup samples have generally not been provided with replacement steam generators.

The following paragraph will be added to SSAR Subsection 5.4.2.4.1 to reflect this response:

SSAR Revision:

The heat and lot of tubing material for each steam generator tube is recorded and documented as part of the quality assurance records. Archive samples of each heat and lot of steam generator tubing material are provided to the combined license holder for use in future materials testing programs or as inservice inspection calibration standards. A minimum of seven feet of tubing in the final heat treat condition is supplied.

Question 252.114

Provide detailed discussion on the extensive operating experience and laboratory testing (including model boiler tests) to justify the use of all volatile treatment (AVT) secondary water chemistry with Inconel 690 for the proposed 60-year plant design life (Section 5.4.2).

Response:

Background information on the material and corrosion properties of nickel-chromium-iron Alloy 690 may be found in EPRI report NP-6997-M, "Alloy 690 for Steam Generator Tubing Applications." The report is a compilation of published and previously unpublished data on the testing of Alloy 690 and includes information on corrosion behavior and a comparative ranking of Alloy 690 with nickel-chromium-iron Alloy 600 and nickel-iron-chromium Alloy 800. In this ranking, Alloy 690 was found to provide either comparable or additional corrosion resistance for a wide variety of postulated steam generator crevice environments relative to the other candidate tubing materials. The report concluded that Alloy 690 is "the material of choice for steam generator tubing applications because of its corrosion resistance in a variety of environments."

The industry has also recently completed a review of secondary water operating chemistry guidelines from the viewpoint of limiting secondary-side initiated corrosion concerns. The review, presented in "Interim PWR Secondary Water Chemistry Recommendations for IGA/SCC Control," EPRI TR-101230, September, 1992, reaffirms the use AVT secondary water chemistry in plants which have not experienced corrosion. This recommendation does not preclude the possibility of adopting an alternate water chemistry as technology evolves, nor does conformance to the guidelines assure that steam generator tubing integrity will be maintained for the 60-year operating design objective. Nevertheless, selection of Alloy 690 tubing and adherence to the AVT water chemistry guidelines provide reasonable assurance for maintaining the long-term integrity of the steam generator tubes under current technology assumptions.

In addition to the water chemistry and the tube alloy material selection, steam generator design features have an affect on the tube integrity of operating steam generators. Currently, there are 106 operating steam generators utilizing, to some degree, AP-600 type design features. After approximately 700 cumulative years of operating experience with Alloy 690 and Alloy 600 tubes, not one tube has been removed from service due to secondary-side initiated corrosion.

SSAR Revision: NONE

Question 252.115

Address the potential for primary water stress corrosion cracking in Inconel 690 for the proposed 60-year plant design life (Section 5.4.2).

Response:

The resistance to primary water stress corrosion cracking (PWSCC) of thermally treated nickel-chromium-iron Alloy 690, chosen for the steam generator heat transfer tubing, has been the subject of extensive corrosion testing in simulated and highly accelerated primary side environments. These corrosion test programs were conducted individually and in joint programs by reactor vendors and primary metals suppliers in the U.S., France, Sweden and Japan over more than a ten year period.

The major results of these evaluations have been summarized in a recent EPRI report (NP-6997). The unanimous consensus of these efforts is that Alloy 690 appears to be highly resistant to PWSCC at the temperatures and operating conditions appropriate to the application in steam generators. See the response for RAI 252.114 for additional background of the tube material selection. Periodic inservice inspection of the steam generator tubing affords an opportunity to see that the integrity of the steam generator tubing is maintained for continued operation of the steam generators.

SSAR Revision: NONE

Question 252.116

Address the resistance to corrosion of Inconel 690 in upset water chemistry conditions which would take place over the proposed 60-year plant design life (Section 5.4.2).

Response:

As discussed in the response to RAI 252.114, nickel-chromium-iron Alloy 690 tubing provides comparable or additional corrosion resistance over a wide range of postulated crevice environments relative to other tested candidate tubing materials. These environments bracket those which may be formed as a result of operation under upset water chemistry conditions.

Tube material selection and design features alone will not assure nor guarantee that the steam generator tube integrity will be maintained for the 60-year operating design objective of the steam generator without defects or significant degradation. Periodic inservice inspection of the steam generator tubing affords an opportunity to see that the integrity of the steam generator tubing is maintained for continued operation of the steam generators.

SSAR Revision: NONE

Question 252.137

Identify the steam and feedwater system materials and provide information to demonstrate that the materials meet the requirements of Section III of the ASME Code (Section 10.3.6).

Response:

Subsection 10.3.6 specifies that material selection for ASME Code, Section III, Class 2 and 3 components in the main steam and feedwater systems are addressed in Subsection 6.1.1.1. where commitment is made to compliance with the ASME code.

> "The material specifications for pressure-retaining materials in each component of an engineered safety features system meet the requirements of Article NC-2000 of the ASME Code, Section III, Class 2, for Quality Group B; Article ND-2000 of the ASME Code, Section III, Class 3, for Quality Group C components; and Article NE-2000 of the ASME Code, Section III for containment pressure boundary components."

The ASME Section III portion of the main steam line is currently specified as carbon steel A/SA-106 Grade B, whereas the feedwater line is currently specified as alloy steel A/SA-355 Grade F22. The SSAR commitment is that the materials used for these lines will comply with the ASME Code as cited above. Material selection could be affected by ongoing piping analyses and leak before break qualification analyses for these lines.

SSAR Revision: NONE

Question 252.143

Although the condensate polishing system serves no safety related function, show that failure of any of its components will not cause damage to the systems required for safe plant shutdown (Section 10.4.6.1.1).

Response:

As described in Subsection 10.4.6 of the AP600 SSAR, the nonsafety-related condensate polishing system is a partial flow system that operates in parallel with main condensate system flow. Condensate polishing flow can be isolated if necessary during power operation. The condensate polishing system piping and components are located in the turbine building and are completely isolated from the safety-related equipment required for safe shutdown of the plant. The systems required for safe shutdown are listed in Section 7.4 of the AP600 SSAR and their successful operation is not impacted by the failure of components in the condensate polishing system.

SSAR Revision: NONE

Question 420.5

There is no discussion on annunciator system and the guidance for manual action in Section 7.5 of the SSAR. There are no analyses to address defense-in-depth design to protect against the common mode failures in the Integrated Protection System and the Integrated Control System. Provide such analyses.

Response:

1) Annunciator system and guidance for manual action.

The AP600 annunciator system is described in Section 18.9.2 of the SSAR. The following clarification describes the way expected operator response to plant alarms will be addressed in the context of the AP600 Man-Machine Interface.

The AP600 M-MIS utilizes the wall panel information station for display of high level alarm indications. Once an alarm is activated, the operator can retrieve information related to the alarm from the operational displays at the operator's console and can query the alarm system. The appropriate alarm response procedure can be called up at the operator workstation and appropriate controls can be accessed. This operating philosophy is expected to be applied to all operational modes in which the alarm system remains functional. In the event of a loss of the alarm system, the Class 1E qualified display processing system is used.

The alarms are also embedded in the operational displays. Thus if the wall panel information station is not available, the information available on the wall panel information station is also available at the operator's console. If an operator's console is inactive, the supervisor's console has the information available, and the plant control functions can be switched over to the supervisor's console.

If the operators' consoles, the supervisor's console, and the wall panel information station are not available, the operator moves to the center bridge console between the operators' consoles to use the qualified display processing system and dedicated hard controls, guided by the emergency operating procedures, to bring the plant to safe shutdown and to maintain that condition. The operator can use paper alarm response procedures and the dedicated controls, with feedback from the qualified display processing system, to maintain the plant in a safe condition. If any alarms are identified through the function based task analysis that prompt a required operator action to bring the plant to a safe shutdown condition and maintain it, they will be provided in both the qualified display processing system and in the operational display system.

Particular guidance concerning when to take manual action is provided through the alarm response procedures, as well as the operating procedures (normal, abnormal, emergency). These are derived from the output of the function based task analysis, after the task allocation between human and computer has been made.

2) Analysis of defense-in-depth design.

A defense-in-depth analysis of the protection and safety monitoring system, as described in NUREG-0493, is currently being performed, and will be submitted upon completion.

In the case of a common mode failure in the plant control system, the protection and safety monitoring system will still be available, and will provide protection for the plant. In the case of a common mode failure that affects both the protection and safety monitoring and plant control systems, the diverse actuation system will be available to protect the plant. Common mode failures of the protection and safety monitoring system and diverse actuation system were included in the Probabilistic Risk Assessment (PRA), therefore, a defense-in-depth and diversity assessment is not required for the plant control system.

SSAR Revision: NONE

Question 420.6

The AP600 design implemented an integrated control system (ICS) as indicated on Figure 7.1-1 of Section 7.7 of the SSAR. However, there is no discussion in Section 7.7 to describe the ICS, no analysis to address protection against common mode failures in the ICS. Provide this information.

Response:

The integrated control system is implemented by the Plant Control System described in Section 7.1.3 of the AP600 SSAR.

The analysis to protect against common mode failure in the Plant Control System was done as part of the Probabilistic Risk Assessment (PRA). The specific fault trees developed to address common cause failures of the plant control system are:

CLCCX	- Common cause failures of control logic cabinet.
CMUXCCX	- Common cause failures of control multiplexer cabinet
CGCCX	- Common cause failures of control group cabinets
SEGSELCX	- Common cause failures of signal selector subsystems

In the PRA, failures of the Plant Control System, including common cause failures, were analyzed together with failures, including common cause failures, of the Protection and Safety Monitoring System and the Diverse Actuation System. This analysis of the AP600 instrumentation and control systems is described in the PRA in Appendix C20, "Protection and Safety Monitoring System, Plant Control System", Appendix C12, "Diverse Actuation System", and Appendix E.3.4.6, "Evaluation of Common Cause Failure for Instrumentation and Control." The conclusion is that the nonsafety-related Plant Control System, taken together with the safety-related Protection and Safety Monitoring System and the nonsafety-related Diverse Actuation System is, by PRA analysis, sufficient to protect the plant for the analyzed events.

SSAR Revision: NONE

Question 435.5

The non-safety ac power systems may fail during a seismic event or fire, in which case, the installed non-safety ac power systems may not be available beyond 72 hours. Section 8.3.1.1.1 of the SSAR states that a provision of two 480 V non-class 1E transportable diesel generators (150 KW each) is made to meet the post-72 hour power requirements following extended loss of all electric power sources. Discuss the provisions for ensuring the availability of the transportable diesel generator to the safety system.

Response:

Two transportable, ac diesel generators will be connected directly by prefabricated cables to the Class 1E regulating transformers, hydrogen recombiners, and temporary equipment transported to the site as listed on SSAR Figure 8.3.1-4. These two transportable diesel generators will be stored at a location far enough from the site so that they remain unaffected by events such as earthquake and explosions and will be transportable to the site within 72 hours.

The storage and transportation details are covered by the combined license applicant.

SSAR Table 1.8-1 will changed to include the following:

SSAR Revision:

8.9	Storage of transportable AC diesel generators	NNS	Combined License applicant coordination	8.3

Question 435.7

Section 8.3.1.1.2.1 of the SSAR states that each standby diesel generator is dedicated to one of the two divisions of permanent non-safety loads. Describe whether the two non-safety ac divisions will meet the physical and electrical independence requirements of RG 1.75.

Response:

SSAR Table 8.1-1 provides the information on the Regulatory Guides and their applicability to the electrical systems design. As per Table 8.1-1, RG 1.75 is not applicable to the SSAR Subsection 8.3.1 which addresses the non-safety ac power supply system design. The nonsafety-related ac power is for invesment protection only and is not required to meet the electrical independence requirement of RG 1.75

The folowing statements provide the electrical separation aspects of the nonsafety-related ac power supply system design:

- Non-Class 1E circuits are electrically isolated from the Class 1E circuits in compliance with the RG 1.75 stipulations, and IEEE Std. 384 requirements.
- Nonsafety-related ac power system design includes two divisions of permanent nonsafety-related loads each supported by its own onsite diesel generator unit.
- Each onsite diesel generator unit is located in a separate enclosure.
- The ac switchgear units pertaining to each of the nonsafety-related ac divisions are located in separate rooms in the Annex Building.
- The control power for the control of the nonsafety-related ac switchgear breakers is provided from separate nonsafety-related dc power sources.
- The detailed raceway and circuit design for the two non-safety ac divisions is yet to be completed. Regulatory Guide 1.75 separation criteria is not going to be implemented for the nonsafety-related ac raceway design.

SSAR Revision: NONE

Question 435.9

Periodic testing and test loading of a diesel generator in a nuclear power plant is a necessary function to demonstrate the operability, capability, and availability of the unit on demand. Periodic testing coupled with good preventive maintenance practices will assure optimum equipment readiness and availability on loss of offsite power. To achieve this optimum equipment readiness, the following items should be considered:

a. The equipment should be tested with a minimum loading of 25 percent of rated load. No load or light load operation will cause incomplete combustion of fuel and the consequences could be the potential equipment failure due to the gum and varnish deposits and fire in the engine exhaust system.

b. Periodic surveillance testing should be performed in accordance with the recommendations of the engine manufacturer.

Provide a discussion of the maintenance and testing program for the diesel generators.

Response:

A discussion of the proposed maintenance and testing program for the diesel generators is provided below:

Maintenance

The diesel generators are not safety-related and will be maintained in accordance with the requirements of the overall plant maintenance program. This program will cover the preventive, corrective and predictive maintenance activities of the plant systems and equipment and will be presented in the combined license application.

Periodic Testing

The periodic testing program will be performed in accordance with the recommendation of the engine manufacturers.

The specific engine loading level will be determined based on the engine manufacturer's recommendations such that the load operation will not cause incomplete fuel combustion that may result in gum and varnish deposits in the engine exhaust system.

SSAR Revision: NONE

Question 435.12

Provide information on the maximum loading of each supply circuit during normal and abnormal operating conditions, including accident conditions and plant shutdown conditions, to demonstrate the adequacy of the lines, circuit breakers, and transformers of the offsite power system.

Response:

The design of the offsite power system is a site specific issue. The information requested can not be determined until the physical characteristics of the offsite power system have been designed. This information will be determined by the Combined License applicant during site selection and design of the transmission system. For further information, please refer to the response for RAI 435.4.

SSAR Revision: NONE

Question 435.17

The frequency of catastrophic failures of the main step-up transformers due to lightning has been greater than what was anticipated. Describe the design for lightning protection of the main step-up transformers.

Response:

The main step-up transformers are protected from lightning coming from two sources. Lightning can affect transformers by both a direct strike to the transformer and also by lightning propagating to the transformer over the transmission lines connected to it. Therefore, two means of protection are used. Grounded shield wires are located above the equipment in the transformer area, including the main step-up transformers, to intercept lightning strikes in the area and conduct them to ground. Suitably rated surge arresters are located on the high voltage side of the main step-up transformers to reduce the magnitudes of incoming lightning caused voltage surges to levels which are well within the insulation withstand capability of the transformer.

SSAR Revision: NONE

Question 435.24

Provide the details of the design of the dc power system that assures equipment will be protected from damaging overvoltages from the battery chargers that may occur due to faulty regulation or operator error.

Response:

The battery chargers are provided with a High DC Voltage Charger Shutdown feature, which includes an alarm relay and indicating light. The High DC Voltage Charger Shutdown functionally disables and locks-out the battery charger whenever the output dc voltage exceeds the preset upper limit of charging voltage. DC equipment and components are rated for maximum equalization voltage of 140V. Thus, the dc system equipment is protected from overvoltage damages.

SSAR Revision: NONE

Question 435.25

Section 8.3.2.2 of the SSAR states that the Class 1E dc system is ungrounded; thus, a single ground fault does not cause immediate loss of the faulted system. However, a ground fault followed by a second ground can produce ground currents of sufficient magnitude to initiate operation of de-energized dc loads or inhibit drop-out of energized dc loads. Detection with alarms is provided for each division of power so that ground faults can be located and removed before a second fault could disable the affected circuit. This has been the subject of Information Notice 88-86 and 88-86, Supplement 1. Describe the ground detection system for the ungrounded dc auxiliary system.

Response:

During the detail design phase, the concerns expressed in Information Notice 88-86 and 88-86, Supplement 1 will be considered for specifying the Class 1E dc ground detection system. Also, plant procedures will be established so that prompt action is taken to clear any ground fault on the Class 1E dc system.

SSAR Revision: NONE

Question 435.32

Describe how the Class 1E power systems meet RG 1.75.

Response:

Compliance with RG 1.75 is discussed in Section 1A of the SSAR. The AP600 Class 1E power system design will meet the physical independence requirement of RG. 1.75 as described in detail in Sections 8.3.2.3 and 8.3.2.4 of the SSAR. AP600 design will be based on IEEE Standard 384-1981 - "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits". A brief outline of the AP600 design is presented below:

The separation of safety-related system will be maintained by the physical layout and the unique identification of equipment, raceways and cables. Different color codes will be used for different safety-related electrical equipment, power, control and instrumentation cables and raceways. In addition, each circuit and cable will carry a unique identification number which will provide a means of distinguishing between circuits of different separation divisions.

Cables associated with the same division will require no physical separation and may be routed in common raceway. Signal cables associated with the same division will require no physical separation and may be routed in common raceways.

Class 1E cables and electrical and I&C equipment (panels, cabinets, components, etc.) will be identified with the appropriate division identifier (A, B, C, or D).

Raceways and cables associated with a specific division (A, B, C, or D) will be physically separated from all other divisions, and from non-safety raceways and cables, as described in SSAR Section 8.3.2.4.

The independence of non-Class 1E circuits from Class 1E circuits will be achieved by complying with the requirements addressed in IEEE 384-1981, paragraph 5.6.

Physical separation between electrical equipment and components associated with redundant divisions will be consistent with the criteria established in IEEE 384-1981, Sections 5 & 6. The Class 1E equipment of each division will be located in safety-related structures.

SSAR Revision: NONE

Question 435.33

Show that the design of the dc power system will ensure the reliable operation of the system's loads and equipment for the full range of operating voltages, including charging, equalizing, and end-of-discharge. The analyses should include the steady state and switching transients.

Response:

The AP600 dc power system design is based on "IEEE Recommended Practice for the Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations - IEEE Std 946-1985". The batteries have been sized in accordance with IEEE Std 485-1983. The battery duty cycles (load profiles) used for sizing the batteries are shown in Tables 8.3.2 - 1, 2, 3 & 4 of the SSAR. These duty cycles include steady state and switching transients as indicated by 0-1 min, 1-1440 min, and 1440-4320 min kW power requirements. The number of plates selected for each cell is based on battery manufacturer's curve for end-of-discharge cycle voltage of 1.75V/cell, i.e., battery end-of-discharge voltage of 105V. Also, aging factor (1.25), temperature factor (1.064) for minimum operating temperature of 67°F, and design margin of 1.1, as recommended by the IEEE Standard 485, have been used to calculate the number of plates. The selected batteries and battery charger sizes and the Class 1E dc power distribution configurations are shown on SSAR Figure 8.3.2-1 (Sheets 1 and 2).

The operating voltage range under all operating modes, including charging, equalizing, and end-of-discharge is 105 to 140Vdc. The maximum equalizing charge voltage for batteries is 140Vdc and the end-of-discharge voltage is 105Vdc. The nominal system voltage is 125Vdc.

The operating voltage range for the equipment and the associated loads will be specified in accordance with Table 1 of IEEE Std 946-1985 to ensure reliable operation of the dc power system for the full range of operating voltages, including charging, equalizing and end-of-discharge. Also, as described above and in the response to RAI Question 435.28, the batteries have been sized to provide steady state and switching transient power within the required voltage range of 105 to 140Vdc.

SSAR Revision: NONE

Question 435.65

Section 8.3.2.2 of the SSAR states that the Class 1E battery chargers and Class 1E transformers have built-in circuit breakers at the input and output sides for protection and isolation. In some cases, they are serving as isolation devices between Class 1E and non-Class 1E circuits. This is not in accordance with Position C.1 of RG 1.75, which precludes the use of interrupting devices actuated only by fault current as acceptable devices for isolating non-Class 1E circuits from Class 1E or associated circuits. Provide acceptable isolation for the Class 1E circuits in accordance with Position C.1 of RG 1.75, or provide justification for deviation from this position.

Response:

Conformance with RG 1.75 is addressed in SSAR Section 1A. The purpose of the built-in circuit breakers provided at the input and output sides of the Class 1E battery chargers and the Class 1E regulating transformers is for the protection of the Class 1E equipment against a fault. In addition, the input side breakers equipped with shunt trip devices will be used to isolate the Class 1E battery chargers and the Class 1E regulating transformers in the event of out-of-tolerance power outputs from the non-Class 1E sources. These breakers are actuated not only by fault current and therefore, may be considered as isolation devices. However, the primary isolation functions are performed by the Class 1E battery chargers and the Class 1E regulating transformers.

The Class 1E battery chargers receive 480V non-Class 1E ac input power through isolation transformers which are built into the battery charger. In addition, the battery chargers are provided with blocking diodes to prevent accidental discharge of the batteries in the event of a fault at the non-Class 1E side of the battery chargers.

The backup ac power to Class 1E UPS loads is provided through 480-208/120V Class 1E regulating transformers. In this mode of operation, the inverter/battery will be disconnected from the UPS system. The regulating transformers are provided with isolation transformers and power conditioners which protect the Class 1E 120Vac system from brownouts, high line voltages, surges, and electrical noise.

In view of the above, the Class 1E battery chargers and the Class 1E regulating transformers together with the built-in circuit breakers meet the isolation device requirements in accordance with RG 1.75. In addition, IEEE Std 384-1981, Section 7.1.2.3 lists inverters, regulating transformers, and battery chargers as an acceptable isolation device

SSAR Revision: NONE

Question 435.68

Section 8.3.2.1.1.2 of the SSAR states that if an inverter in the UPS system is inoperable or the Class 1E 125 Vdc input to the inverter is unavailable, the power to the 120 Vac bus is transferred automatically to the backup ac source. The backup ac power supply in Figure 8.3.1.1 (Sheet 2 of 2) of the SSAR is shown to be a non-Class 1E source. Traditionally, this ac source has been classified as a Class 1E source in existing plants and in the evolutionary designs. Where is the Class 1E/non-Class 1E boundary? Does it allow connection of the transportable diesel generator through the Class 1E system only? The regulating transformer assembly feeding the Class 1E battery charger should be classified as Class 1E. Revise the SSAR accordingly, or provide justification for not doing so.

Response:

The Class 1E and non-Class 1E boundary is at the regulating transformer. This makes the backup ac source to the inverter Class 1E. The transportable diesel generators are connected through the Class 1E system only. As indicated by Note 2 on Figure 8.3.2-2 of the SSAR, the regulating transformers are Class 1E qualified. Also, it is further clarified in Section 8.3.2.1.1.2 of the SSAR which states "The backup power is received from the diesel generator backed non-Class 1E 480Vac bus through the Class 1E regulating transformer. A reference may be made to the responses to RAI 435.5 and RAI 435.65 for further clarifications.

SSAR Revision: NONE

Question 435.69

Identify the vital areas and hazardous areas where emergency lighting is needed for safe shutdown of the reactor and the evacuation of personnel in the event of an accident. Describe the lighting provided to accommodate those areas so identified.

Response:

As described in Section 9.5.3.2.2 of the SSAR, the emergency lighting will be provided in the vital areas and hazardous areas as follows:

- The main control room and remote shutdown areas have been identified as vital areas where emergency lighting is required. Lighting in these areas consists of 120 Vac fluorescent lighting fixtures which are supplied from the Class 1E dc & UPS system. There are no other areas required to achieve safe shutdown of the reactor.

- Emergency lighting in plant areas for safe ingress and egress of personnel following loss of all ac power is provided with sealed beam fixtures. The fixtures have self-contained battery with eight hour rating and battery charger powered from both onsite and offsite AC power. Power to these units automatically switch to their internal dc source once normal ac power is lost. Ingress/egress routes are used for evacuation of personnel in the event of an accident. See also question RAI 435.72.

SSAR Revision: NONE

Question 440.22

Various computer codes have been referenced in Chapter 4 of the SSAR, most of which have been used to analyze different kinds of Westinghouse fuels, i.e., the STD 17X17, OFA and VANTAGE 5. Although a cursory review of chapter 4 indicates that there are great similarities between the AP600 fuel and previously designed Westinghouse fuel, minor differences do exist.

a. Can these previously approved codes accommodate/address the differences that exist between the AP600 fuel design and earlier Westinghouse designs?

b. Does sufficient data (experimental or otherwise) exist to demonstrate the capabilities of these established codes to handle any AP600 differences?

c. Describe the data base that justifies the DNB correlation and limits applied to the AP600 fuel.

Response:

Parts a. and b.

With regard to the codes used in Chapter 4 for nuclear design, thermal-hydraulic analyses and fuel design, the previously approved codes referenced in this Chapter can accommodate the very minor differences in the AP600 fuel design. Relative to the previously approved VANTAGE 5H design (Section 4.1, Reference 1), the only changes in the AP600 fuel design within the active fuel region of the fuel assembly is the addition of one low pressure drop structural grid and one IFM grid. The neutronic, thermal-hydraulic and mechanical design characteristics of the AP600 fuel design are essentially identical to that of previous proven designs. Sufficient experimental and operating data exists which has been used in the qualification of the nuclear design, thermal-hydraulic design, mechanical design, and fuel design codes to handle minor differences in the AP600 design.

Part c.

The WRB-2 correlation will be used to analyze the AP600 fuel. The database for this correlation is described in WCAP-10444-P-A. The AP600 fuel geometry is within the licensed parameter range of the WRB-2 correlation. The flow rates at the time of minimum DNBR of the loss of flow event are somewhat below the previously licensed WRB-2 lower limit on flow ($G = 0.9 \times 10^6$ lb/ft^2-hr). DNB testing is being done to verify the extension of the WRB-2 correlation to these lower flows. See the response for RAI 440.1 for information on the DNB testing. The W-3 correlation will be used where the WRB-2 correlation is not applicable, e.g., steamline break.

SSAR Revison: None

Question 440.23

Section 5.4.7.2.1 of Chapter 5 of the SSAR provides "a summary of the specific AP600 design features that address SECY 88-017 regarding mid-loop operations."

a. The staff believes that the reference to "SECY 88-017" stated in this section is intended to mean "Generic Letter (GL) 88-17." Please confirm.

b. In addition to GL 88-17, the staff also addressed its concerns regarding shutdown and low power operations in NUREG-1410 and provided guidance in draft NUREG-1449. Are the AP600 design and the proposed Technical Specifications consistent with the findings and recommended industry actions discussed in NUREG-1410 and draft NUREG-1449?

c. The staff is aware that many issues related to shutdown and low-power operations are the plant owner's responsibility as they are related to operation, maintenance and refueling plan, procedures and risk management. What are the specific guidance or requirements that must be implemented by the AP600 plant owners to fulfill their responsibilities with regard to shutdown or low power operations?

d. Section 5.4.7.7 of the SSAR states that the motor-operated valves connected to the hot leg are interlocked to prevent them from opening when RCS pressure exceeds 450 psig, and to prevent their being opened unless the isolation valve from the IRWST to the NRHRS pump suction header is closed. Confirm that there is no autoclosure interlock for the AP600 NRHRS isolation valves that could result in loss of decay heat removal due to unplanned activation.

e. Are there any systems or components needed for shutdown cooling which are deenergized or have power locked out during plant operation? If so, discuss what actions have to be taken to restore operability to the components or systems, and describe where the actions must be taken.

Response:

a. The reference to "SECY 88-017" stated in this section is intended to mean "Generic Letter (GL) 88-17." The SSAR will be updated to reflect this change.

b. In addition to the design features described in Section 5.4.7.2.1 of Chapter 5 of the SSAR which provided "a summary of the specific AP600 design features that address GL 88-017 regarding mid-loop operations," the AP600 contains passive safety grade protection against a loss of normal RHR during shutdown. The recommendations of the specified NUREGs are incorporated into the design and technical specifications of the passive safety systems. During mid-loop operations, the IRWST is operable, and the ADS valves connected to the pressurizer are open. If the normal RHR system is lost, the IRWST will provide core cooling by gravity injection. Technical Specification 3.5.4 addresses this issue.

c. The combined license applicant is responsible to ensure that the AP600 design requirements, system operating procedures, and maintenance and testing recommendations are properly implemented during plant operations, including specific requirements related to shutdown, refueling, and low power operation. The requirements for shutdown operations are contained in the AP600 Technical Specifications. Please see the response to RAI 440.28 for a description of the requirements during shutdown operations.

d. There is no autoclosure interlock for the AP600 NRHRS isolation valves that could result in loss of decay heat removal due to unplanned activation.

e. The normal RHR system contains valves that isolate the system from the reactor coolant system hot leg (RNS-V001A,B RNS-V002A,B) have power locked out at the motor control center during plant operation. Prior to initiating normal RHR cooling, the operator will be required to go to the motor control center to restore power to the valves. Note that this will occur prior to mid-loop operation.

SSAR Revision:

5.4.7.2.1 Design Features Addressing Mid-loop Operations

The following is a summary of the specific AP600 design features that address ~~SECY-88-017~~ Generic Letter (GL) 88-17 regarding mid-loop operations.

Question 440.30

The AP600 design does not fully comply with the staff position stated in SECY-90-015 with regard to interfacing system LOCA. For example, it is stated in Section 1.9.5.1 of the SSAR that the normal residual heat removal system pump seal is not designed for the higher NRHRS piping design pressure and could fail if overpressurized. Though the AP600 design of NRHRS limits the leakage to within the capacity of the chemical and volume control system to allow the plant to be place in safe shutdown, the leakage from the pump seal will bypass the containment. As stated in the draft Commission paper entitled "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," (Letter from D.M. Crutchfield to E.E. Kintner dated February 27, 1992), the staff concludes that, to the extent practicable, all elements of the low pressure system, including pump seals, should be designed to withstand the full RCS pressure. For those systems and components that do not meet this requirement, justifications must be provided as to why it is not practicable to do so.

a. Provide a list of systems and components interfacing with the RCS that are not designed to have ultimate rupture strength at least equal to the full RCS pressure.

b. For each of these systems or components in Item (a), either (1) confirm that it will be redesigned to meet the ultimate rupture strength criteria, or (2) provide justification as to why it is not practicable to meet this criterion, and confirm that these systems or components will include the capability for isolation valve leak testing, valve position indication, and include high pressure alarms as discussed in the draft Commission paper.

Response:

a. The AP600 design features that address intersystem LOCA are described in SSAR section 5.4.7.2.2. The AP600 position with regards to this issue is contained in SSAR section 1.9.5. The normal residual heat removal system is designed to the extent practible, to have an ultimate rupture strength equal to the full RCS operating pressure. The normal RHR pump seal is the only component or system connected to the RCS that is not designed to have an ultimate rupture strength equal to the normal RCS operating pressure.

b. The following is a discussion of the practicability of designing the RHR pump seal to full RCS operating pressure.

The rupture pressure of the RHR pump mechanical seal was investigated with suppliers of nuclear grade mechanical seals. The rupture pressure of the mechanical seals in the AP600 RHR pumps is approximately 670 to 700 psig with the pump operating. The rupture pressure is somewhat higher when the pump is not operating.

There is a fundamental problem with designing an RHR pump seal that can withstand full RCS pressure. Any type of seal that can withstand the full RCS pressure will likely have abnormally fast wear of the seal faces during normal plant operation at low seal pressures. This increased wear at normal plant operating conditions could well prevent the seal from maintaining the pressure boundary if ever exposed to the full RCS pressure. Use of the high

pressure seals will also require more frequent maintenance during normal operation. In simple terms, the seal can be designed for normal low pressure operation or for full RCS pressure, but it is unlikely that a seal can be designed to operate desirably at both conditions.

It is the AP600 position that the best solution is to utilize the existing single seal design in conjunction with a sturdy disaster bushing design. This combination will maximize the reliability of the seal during normal RHR operation and minimize maintenance and associated radiation exposure. Furthermore, this design approach will minimize the leakage from the normal RHR pump in the event that the mechanical seal fails. The leakage from this design can be controlled so that only a small portion of the water that leaks past the primary seal faces escapes to the pump cubicle and the majority of the leakage is piped to a controlled drain. This is more favorable than a seal specially designed for full RCS pressure at the expense of normal condition reliability.

Because of the design features of the AP600 which prevent the likliehood of exposing the normal RHR system to full RCS pressure, and because of the limited leakage from the normal RHR pump seal, the current PRA analyses has shown that the interfacing system LOCA accident has been eliminated (contribution to CMF $< 10^{-9}$). Therefore, the current AP600 design approach should be retained.

In addition, provisions for leak detection and position indication for the isolation valves that separate the normal RHR system from the RCS are consistent with those recommended by the NRC. Furthermore, high pressure alarms will be added to the normal RHR pump suction to comply with the NRC position.

The following paragraph will be added to SSAR Subsection 5.4.7.2.2:

SSAR Revision:

RCS Pressure Indication and High Alarm - The AP600 Normal RHR system contains an instrumentation channel that indicates pressure in each RHR pump suction line. A high pressure alarm is provided in the main control room to alert the operator to a condition of rising RCS pressure that could eventually exceed the design pressure of the normal RHR system.

Question 450.9

As stated in Section 6.5.2, Item II.1, Revision 1 of the SRP, the pH of the aqueous solution collected in the containment sump after completion of injection of the containment spray and ECCS water, and all additives for reactivity control, fission product removal, or other purpose, should be maintained at a level sufficiently high to provide assurance that significant long-term iodine re-evolution does not occur. Long-term iodine retention with no significant re-evolution may be assumed only when the equilibrium sump pH, after mixing and dilution with the primary cooling and ECCS injection, is above 8.5. Sections 6.5.2 and 6.5.3 of the SSAR do not indicate that the long-term pH of the sump water will be maintained at a minimum of 7.0. It is understood that the phrase "by the onset of the spray recirculation mode" is not applicable to the AP600 design; however, long-term iodine re-evolution is a concern. Also, WCAP-13053 indicates that there will be additives for the adjustment of the sump solution pH. Demonstrate how long-term iodine retention is achieved and maintained, precluding any significant re-evolution of iodine (Section 6.5.2).

Response:

The RAI states that long-term iodine retention with no significant re-evolution can be assumed only when the equilibrium sump solution pH is above 8.5. This is not consistent with Revision 2 of Section 6.5.2 of the Standard Review Plan which states that long term retention of iodine may be assumed if the sump solution is adjusted to 7.0 or higher.

Adjustment of the sump pH to 7.0 or greater is accomplished by the passive core cooling system. This function is discussed in Subsection 6.3.2.1.4 of the SSAR.

The SSAR will be modified by the addition of the paragraph below to the end of Subsection 6.5.2.

SSAR Revision:

Much of the non-gaseous airborne activity would eventually be deposited in the containment sump solution. Long term retention of iodine in the containment sump requires adjustment of the sump solution pH to 7.0 or above. This pH adjustment is accomplished by the passive core cooling system and is discussed in Subsection 6.3.2.1.4.

Question 471.2

Section 12.3 of the SRP specifies that the SSAR should contain radiation zone designations (including zone boundaries and normal traffic patterns) on the plant layout drawings. The zone maps are laid out very well, except that Very High Radiation Areas as defined in the revised 10 CFR Part 20 are not identified during normal and anticipated operational occurrences. Also, there are no traffic patterns identified for normal traffic flow or for access to vital areas during accident operations. This information is needed by the staff to ensure all areas having potentially lethal levels of radiation are identified and controlled. Provide this information.

Response:

This is an interim response to the referenced question.

Section 12.3 will be revised to describe the "Very High Radiation " area locations and "Vital Areas". The traffic patterns during normal and accident operations will be reflected on revised Figures 12.3-1 and 12.3-2.

A final response to the question including figures will be submitted on 2/26/93.

SSAR Revision: NONE AT THIS TIME

Question 471.3

Provide expected peak airborne radioactivity concentrations, estimated man-hours of occupancy, and estimated inhalation exposures for all areas of the plant accessed by plant personnel. This information is required by the Standard Review Plan and is needed by the staff to ensure that the plant's ventilation flow is sufficient to maintain airborne radioactivity levels ALARA.

Response:

This is an interim response to the referenced question.

Section 12.4 will be revised to incorporate the expected peak airborne radioactivity concentrations, estimated man-hours of occupancy, and estimated inhalation exposures for all areas of the plant accessed by plant personnel. The final response to the question will be submitted on 2/26/93.

SSAR Revision: NONE AT THIS TIME

Question 480.1

Section 1.2.1.4.1 of the SSAR states that the number and complexity of operator actions required to control the safety systems are minimized. One of the principal design criteria of the EPRI ALWR Requirements Document for passive plants is that the core must be cooled and containment integrity maintained for 72 hours without reliance on ac power and operator action. This criteria is stated in Section 2.3.2.9 of Chapter 1 and Section 1.2.1.1. of Chapter 5 of the EPRI Passive Requirements Document. The extent of this commitment regarding operator action in the AP600 appears to be less than that in the EPRI Requirements Document.

It is the staff's understanding that the AP600 will comply with the EPRI ALWR Requirements Document for passive plants. Clarify your position with regard to this matter. Will the AP600 meet the above cited EPRI criteria without modification? If the AP600 takes a different position, identify the differences and state all the operator actions during the 72-hour period.

Response:

Consistent with ALWR requirements for Licensing Design Basis events, the AP600 safety-related systems are designed to maintain containment integrity and core cooling without operator action or ac power for at least 72 hours.

SSAR Revision: NONE

Question 480.3

Compare the failure probability of the PCS in an AP600 design to the failure probability of the containment heat removal systems in a typical Westinghouse plant. What is the failure probability of the air operated valves in the PCS? Compare the consequence of the failure of the PCS to the consequence of the failure of the containment heat removal systems in a typical Westinghouse plant. In Section 1.2.1.4.1 of the SSAR, it is stated that with only air cooling, the containment pressure does not exceed its ultimate pressure during a core melt scenario. Does this correspond to the worst consequence of a total failure of PCS? What is the value of the containment ultimate pressure? What are the bases for choosing this value (Section 6.2.2)?

Response:

The failure probability of the AP600 passive containment cooling system (PCS) is estimated to be 7.49E-4 failures per demand. The dominant contributor to PCS failure is the common cause failure to open of the fail safe air-operated valves and this failure probability is 7.1E-4.

The failure probability of containment heat removal systems in a typical Westinghouse plant is: 1) Failure of 2 out of 2 fan cooler units is 4.E-3 (assuming that all support systems are available) and 2) Failure of 2 out of 2 containment water recirculation systems is about 2.E-3 for the equipment. Note that operator action is required, in a typical plant, to align the recirculation system and to align component cooling water to the residual heat removal system heat exchangers. Either one containment fan cooler or one RHR heat exchanger operating in recirculation is typically sufficient to preclude containment failure.

A very important advantage of the AP600 PCS, not reflected by a comparison of containment heat removal systems failure probabilities, is that PCS failure in the AP600 is independent of the failures that cause core damage. The only commonality is the actuation logic but in the AP600 design, this failure requires failure of the plant monitoring system, the diverse actuation system, and hard-wired manual actuation, which uses information provided by diverse indication from the diverse actuation system.

Failure of the water cooling system for the outside of the containment shell, which results in only air cooling being available to cool the containment, is the worst PCS failure evaluated in the PRA. Even in this case, sufficient cooling from only air maintains the containment pressure below Service Level C. It could be postulated that air cooling of the containment could be reduced by blocking the air flow path through the annulus with water accumulating at the bottom of the annulus if the annulus drains were plugged. The probability of such a scenario is negligibly small and the failure of the annulus drains is independent of any failures that lead to core damage. Provisions to preclude blockage of these drains are discussed in response to RAI 720.26.

The value of containment ultimate pressure and the bases for this value are provided in SSAR Subsection 3.8.2.4, "Design and Analysis Procedures" (for the containment vessel).

SSAR Revision: NONE

Question 720.3

The AP600 design incorporates a number of novel features (e.g. new controls) and passive devices. The PRA is not capable of portraying a realistic risk profile for the AP600 design unless a comprehensive and systematic search was conducted to assure that adverse failure modes of these new features were integrated into the PRA models. By adverse failure modes, we do not only mean failures that result in challenging safety systems or obstruct the shutdown process, but also those failures that may motivate the operators to take wrong actions. Provide information detailing your efforts in searching for these failure modes.

Response:

A comprehensive and systematic search was conducted for adverse failure modes of both novel features of the AP600 and of other features which are also found in current plants. The systematic strategy that was used to identify potential errors of commission is described in Section D.1, "Errors of Commission", of the PRA report.

The first part of this approach, which consists of identification of the systems and of the modification of their operating status with potential influence of the successful performance of a safety function, was used to also identify hardware spurious failures that could aggravate the accident progression.

PRA Revision: NONE

Question 720.6

The AP600 PRA does not provide a concise summary of sequences in which recovery actions were credited. These recovery actions are important to the staff's understanding of system performance and required operator response. Provide a listing of top accident sequences in which recovery was credited and characterize these actions. By characterizing these actions, the staff means identifying error probability, locations, describing the performance shaping factors that were used, listing the control room information motivating the recovery action, etc. Was credit for recovery handled before or after truncation, and was recovery integrated in the analyses at the cutset or sequence level?

Response:

For events evaluated at power, the dominant sequences are shown in Appendix F, Table F-7. The operator recovery actions for these events are identified as LPM-MAN01, LPM-MAN03, LPM-MAN04, ATW-MAN03, and REG-MAN00. and shown in the following 41 sequences: 5, 6, 12, 16, 18, 23, 24, 25, 28, 30, 32, 34, 35, 37, 38, 42, 53, 56, 58, 60, 62, 63, 65, 66, 68, 69, 70, 74, 75, 76, 78, 80, 81, 82, 83, 89, 91, 93, 96, 97 and 100.

For events evaluated at shutdown, the dominant sequences are shown in Appendix F, Table F-10. The operator recovery actions for these events are identified as IWN-MAN00, CCN-MAN02, RHN-MAN03, and PRN-MAN02S, and shown in the following 21 sequences: 11, 16, 17, 18, 19, 30, 31, 34, 35, 48, 49, 62, 66, 70, 80, 81, 84, 86, 88, 94 and 95.

For events evaluated in the containmant analysis, the dominant sequences are shown in Appendix G, Table G-5. The operator recovery actions for these events are identified as CIX-MAN00, SFN-MAN00, and ZON-MAN02, and shown in the following 33 sequences: 1, 4, 11, 16, 26, 33, 36, 38, 45, 64, 76, 100, 112, 113, 132, 136, 138, 139, 151, 163, 168, 169, 172, 175, 176, 179, 180, 181, 185, 189, 190, 258 and 259.

Table 720.6-1 provides the recovery actions, the descriptions, the expected cues for the operators, the performance shaping factors used in the evaluation, the location where the action is performed (that is; in control room or locally), and the estimated human error probability. Two recovery actions, REG-MAN00 and ZON-MAN02 are performed locally; the other recovery actions are carried out in the control room. The THERP methodology was used to evaluated the recovery actions, except CIX-MAN00 and ZON-MAN02; the HEPs for these two actions are based solely on engineering judgement, and are believed to be conservative.

The recovery actions were integrated into the PRA at the fault tree or cutset level. Therefore, credit for recovery was handled before truncation.

PRA revisions: NONE

Table 720.6-1 - Recovery Actions

Identifier	Operator Action	Cues	PSF	Location	HEP	Note
LPM-MAN01	Failure to recognize the need for reactor coolant system depressurization during a small loss of coolant accident or loss of high-pressure heat removal system	Low SG wide range level; high hot leg temperature; low hot leg water level	Long procedure; Time window (Tw) = 30 min; high stress level	Control room	2.20E-03	1
LPM-MAN03	Failure to recognize the need for reactor coolant system depressurization when only the diverse actuation system is providing information during a small loss of coolant accident or transient	Low hot leg water level; jammed instrument	Long procedure; Tw = 15 min; high stress level	Control room	8.30E-02	2
LPM-MAN04	Failure to recognize the need for reactor coolant system depressurization when only the diverse actuation system is providing information during a medium loss of coolant accident	Low hot leg water level; jammed instrument	Long procedure; Tw = 15 min; high stress level	Control room	8.30E-02	2
ATW-MAN03	Failure to recognize the need and failure to manually trip the reactor within 1 minute, given anticipated transient without scram	Low narrow range level in steam generators; high pressurizer pressure; flow mismatch between feedwater flow and turbine inlet pressure	Short procedure; Tw = 1 min; moderate stress level	Control room	1.53E-02	3
REG-MAN00	Failure to regulate the startup feedwater following full opening of the regulating valves after a loss of compressed air	Estimated 7 to 10 different alarms indicating loss of compressed air	Short procedure; Tw = 50 min; high stress level	Local	2.10E-01	4

Table 720.6-1 - Recovery Actions

Identifier	Operator Action	Cues	PSF	Location	HEP	Note
IWN-MAN00	Failure to manually open two motor-operated valves during shutdown conditions with the normal residual heat removal system unavailable	1 group of alarms indicat-ing failure of the normal residual heat removal system; low hot leg water level	Long procedure; Tw = 60 min; high stress level	Control room	1.15E-03	5
CCN-MAN02	Failure to exclude heat exchanger H001A and align H001B during normal operation	High temperature on the line downstream of the heat exchanger	Long procedure; Tw = 60 min; moderate stress level	Control room	2.62E-02	6
RHN-MAN03	Failure to recognize the need and failure to manually restart the normal residual heat removal system pumps following grid recovery within two hours after a loss of offsite power and failure of both automatic and manual transfer onto a diesel generator, during shutdown	Voltage return on the grid; no flow from the normal residual heat removal system; low hot leg temperature	Short procedure; Tw = 120 min; high stress level	Control room	1.97E-03	7
PRN-MAN02S	Failure to actuate the passive residual heat removal system (PRHR) air operated valves, V108A and V108B following loss of offsite power during shutdown	RHR pumps trip; RCS pressure increase (up to normal RHR relief valves opening setpoint); low pressurizer level; low hot leg water level	Long procedure; Tw = 30 min; high stress level	Control room	2.55E-03	8
SFN-MAN00	Failure to recognize the need and failure to perform IRWST makeup with spent fuel or chemical and volume control system after containment isolation following gravity injection	Low IRWST level; low reactor cavity level	Long procedure; Tw > 120 min; high stress level	Control room	9.40E-04	9

Table 720.6-1 - Recovery Actions

Identifier	Operator Action	Cues	PSF	Location	HEP	Note
CIX-MAN00	Failure to detect loss of all instrument-ation and control, and diverse indication system alarms	Loss of plant power generation	N/A; engineering judgement applied	Control room	1.00E-01	10
ZON-MAN02	Failure to start diesel generator locally in order to provide power for long term RCS makeup following loss of control room control	Loss of AC power with loss of remote operation capability	N/A; engineering judgement applied	Local	1.00E-02	11

NOTES

1. The LPM-MAN01 operator action represents diagnosis of the event to perform actions that could be associated with three systems or subsystems, (the ADS, RCS and CMT). There are cues that are unique to actions for each system, as well as cues that are common to particular cases for the different systems. The PRA models considered the cases in which common cues are provided to the operators for the different system failures. These cases are believed to be representative of all possible cases.

   In general, the human reliability analysis considers the presence of the reactor operator and the senior reactor operator throughout an emergency. A moderate dependency is assumed between the operators. Recovery is applied for the presence of the shift supervisor and shift technical advisor (STA) for time windows greater than 10 minutes.

2. The LPM-MAN03 operator action represents diagnosis of the event when only the diverse actuation system is providing information. It is believed that the signals for this event will be generated much later than the signals for the event LPM-MAN01, discussed above. It is estimated that a time window of 15 minutes exists for recognizing the diagnostic cue for this event. No credit is taken for the presence of STA in the analysis, due to the lack of instrumentation and the relatively short time window.

   The factors that are considered in the evaluation of LPM-MAN03 are applied to LPM-MAN04.

3. The ATW-MAN03 operator action to trip the reactor has a time window of 1 minute. No recovery credit is applied in the evaluation of this event, due to the 1 minute time window.

4. The REG-MAN00 operator action is performed locally. The flow must be regulated by stroking a manual valve, which is considered to be relatively difficult to perform. No credit is taken for the presence of the STA in the evaluation of this local action.

5. The IWN-MAN00 operator action is carried out within a short procedure, and is required to be completed in an estimated time window of 1 hour. Slack time is believed to exist; therefore, recovery credit is applied for the presence of the shift supervisor and STA.

6. The CCN-MAN02 operator action is carried out within a somewhat lengthy process. The estimated time window for this task is about 1 hour. Slack time is believed to exist; therefore, recovery credit is applied for the presence of the shift supervisor and STA.

7. The RHN-MAN03 operator action is carried out within a short procedure, and is required to be completed in an estimated time window of 2 hours. Slack time is believed to exist; therefore, recovery credit is applied for the presence of the shift supervisor and STA.

8. The PRN-MAN02S operator action is carried out within a long procedure. However, there are few steps to be completed in an estimated time window of 30 minutes. Slack time is believed to exist; therefore, recovery credit is applied for the presence of the shift supervisor and STA.

9. The SFN-MAN0O operator action is carried out within a long procedure, and is required to be completed in an estimated time window greater than 2 hours. Slack time is believed to exist; therefore, recovery credit is applied for the presence of the shift supervisor and STA.

10. The CIX-MAN00 operator action considers the loss of plant power generation. For this scenario, plant instrumentation is not providing information to the operators. It is estimated that the most limiting time window for detecting this failure is 40 minutes for a "cold leg break" accident; a time window of about 2 hours is estimated for the transient event. The shorter time window is reflected in the HEP of 1.00E-01; this value is believed to be conservative.

11. The ZON-MAN02 operator action is a local action to start the diesel generator. The time window to complete this task is estimated to be 24 hours. No other performance shaping factors are considered in the evaluation. The HEP of 1.00E-02 is assigned as a screening value based on engineering judgement.

Question 720.7

Provide information describing how cutsets were truncated and how the truncation limits were selected.

Response:

The cutset generation was done by the WESCUT/WESLGE codes on a IBM-PC 486 with a software limit of 1000/2000 cutsets per fault tree. Following this the fault tree linking process was done. This process consists of first a "REDUCTION" (linking subtree cutsets into a system cutset file) of system cutset files, then the accident sequence quantification (multiplication of fault trees making up accident sequences). The WLINK code system was used for fault tree linking. This code system has a limit of 9900 cutsets on the PC and 100,000 cutsets on the UNIX workstation. Most of the "REDUCTION" was done on the IBM-PC, some of the reduction and all of accident sequence quantification where done on a UNIX workstation, where software limits on the number of cutsets are larger.

A. For fault tree quantification, a 1.0 E-09 cutoff probability was used. When this cutoff caused the software to go out of limits, one of the following two actions were taken:

   1. For highly redundant systems, such as ADS, the original fault tree was broken into artificial subtrees, which were individually quantified and later their cutsets were linked.

   2. For other fault trees, the cutoff probability was increased until the software ran and provided an ample number of cutsets representing the system failure.

B. For fault tree "REDUCTION" the objective was to use a cutoff probability of 1.0 E-12, to the extent possible. If the software limit for number of cutsets was exceeded, the cutoff probability was increased until the software ran, and an ample number of cutsets representing the system failure were obtained.

   For some larger system cutset files, the reduction was done on a UNIX workstation where a low cutoff probability can be chosen, since the software limit for number of cutsets is 100,000 on the Unix workstation.

C. For event tree sequence quantification by fault tree linking, cutoff probabilities ranging from E-10 to E-15 were used, based on the initiating event frequency. For lower initiating event frequencies, a lower cutoff value was used to allow system level linking to occur at least at the E-10 level or lower.

Based on the care provided in this process, and the involvement of various cognizant engineers in review of the output cutset files, there is confidence that all dominant cutsets were captured in the output files.

D. Precautions were also taken in construction of the fault trees to avoid unnecessarily challenging the number of cutset limits of the software. This is done as follows:

In construction of the fault trees, implied OR logic in constructing some basic events was used, thus reducing the number of independent basic events to a minimum. ("implied OR logic", combines as many independent failure modes of a component as possible, into a single basic event.) When the number of basic events are reduced to a minimum set in the fault tree, the resulting number of cutsets representing the fault tree is also minimized, and the failure probability of each cutset is increased (in effect, some cutsets now represent "families of cutsets").

This modeling process does two things:

1. It allows representing the same system with a lower number of cutsets (avoids exceeding software cutset limits);

2. It allows picking up more cutsets at a given cutoff probability (because more cutsets now have higher probabilities since they actually represent families of cutsets).

An excessive number of cutsets was not required to adequately represent a fault tree. The dominant core damage cutsets were captured using this process.

PRA Revision: NONE

Question 720.10

Several of the core damage sequences result from the failure of the NRHR or CVS to inject coolant into the vessel. These systems must be aligned for injection, and these actions result in containment bypass if the pumps fail or suction is not available. The staff is concerned with the potential to bypass the containment from failure of the NRHR or CVS when it is aligned to inject coolant into the vessel during LOCA or transient sequences that lead to core damage. In order to prevent containment bypass, the operators must isolate the system from the reactor vessel when these systems fail to inject. These operator actions were not evaluated by Westinghouse. Describe the human actions that are planned to isolate NRHR and/or CVS upon failure to inject and how these associated human error probabilities will be quantified.

Response:

Injection from the RNS and the CVS are taken credit for in the AP600 PRA. The potential to bypass the containment due to failure of the RNS or the CVS does not increase because of the following:

CVS - The system can only inject water from tanks located outside of the containment; it does not have suction connections from the IRWST or the RCS. Note that water can be removed from the RCS by the CVS, but water cannot be recirculated through the CVS, as in current plants. Instead it would be sent to the waste processing system. Following actuation of an "S" signal, the RCS letdown and the connection to the waste processing system are isolated and these connections are not reopened to provide RCS injection from the CVS.

The process of aligning the CVS to provide injection does not increase the chance of a containment bypass because all of the valves between the CVS makeup pumps and the RCS are normally open except one check valve. In addition there are eight isolation valves in this path between the RCS and the makeup pumps. These isolation valves include simple check valves, fail open air operated valves, dc powered motor operated valves and stop check valves. With the redundancy and diversity of valve types, in particular the different types of check valves (simple and stop) the probability of a containment bypass occurring on failure of injection would not contribute to the frequency of release from the plant.

RNS - The RNS can inject water into the RCS from the IRWST located inside the containment. Failure of the RNS to inject is almost entirely due to either failure of the operators to start the system, failures of the pumps to start, or failures of MOVs to open. None of these failures lead to containment bypass because either the RNS would remain isolated from the containment or, if the RNS was aligned and the pumps failed to start, the pressure boundary of the RNS would still be intact. Note that since the RNS pressure boundary is Regulatory Guide 1.26, quality group C / seismic category I and it is designed for 900 psig, the chance of its rupturing and causing containment bypass is insignificant. Also, note that the RNS does not have connections that are used to transfer water to or from other systems outside of containment.

AP600

As discussed above, it is unnecessary for the operators to take actions to prevent containment bypass, when they attempt to use the CVS or the RNS to provide RCS injection and these systems fail. As a result, the AP600 PRA does not consider operator action to isolate these systems from the RCS or the containment.

PRA Revision: NONE

Question 720.16

The staff observes that the severe accident initiators for the external events analyses for the AP600 design were either screened out based on qualitative analyses or they had unusually low frequencies when compared with similar sequences for the current generation of operating plants. Provide a listing and characterization of AP600 design features that have a major contribution to reducing risk from external events to the level estimated in the AP600 PRA, as well a comparison between the major assumptions in the analyses and those generally used in published PRAs.

Response:

a. AP600 design features which reduce risk:

a1. The AP600 is a passive plant. Major contributors to plant risk in many previous PRAs have been the failures of pumps to provide decay heat removal. Because the AP600 does not rely on active components, such as pumps, to remove decay heat, fire induced failure mechanisms of active components are not present. Active components are credited in the AP600 PRA, but decay heat removal can be achieved without their success.

a2. The AP600 incorporates spatial separation in its layout, which prevents a single event from disabling a function. For instance, of the six vital battery rooms, four are located on one elevation in separate rooms, while the other two are located on a different elevation in separate rooms. There is also a spare battery room in addition to the six mentioned above, which is located in a separate room.

a3. Most of the safety systems which are required to reach a safe stable state are located inside of the containment, and are not subject to external event effects.

b. Comparison between major assumptions used in AP600 PRA and existing PRAs

b1. Internal flooding:

i) Assumptions which were used in the AP600 internal flooding PRA were similar to those used for flooding PRAs of present day plants. In addition, flooding hazards during reduced power and shutdown operations were considered for the AP600 PRA. Most existing PRAs consider at-power hazards only, consistent with the requirements of NUREG-1335.

b2. Internal fire:

i) Barrier failure was not considered credible in the AP600 fire PRA, whereas it is considered in NUREG/CR-4840. Barrier failure is not considered in other methodologies approved by the NRC (e.g., NUREG/CR-2300, Fire Induced Vulnerability Evaluation). The combustible loading in areas separated

by barriers is low, so barrier integrity will not be challenged by a maximum fire. Moreover, all barriers proposed for the AP600 are designed based on NFPA standards. For these reasons, barrier failure was not considered to be credible.

(ii) Offsite power was assumed to be available during a fire event. Appendix R analyses for existing plants assumed a loss-of-offsite power concurrent with a fire. The AP600 PRA, however, considered whether a fire could cause a loss-of-offsite-power, and it was concluded that this was not a likely event.

(iii) Containment fires were not quantitatively analyzed for the AP600 PRA. Oil spills from reactor coolant pumps have historically been the source of containment fires. The "canned motor pump" design of the AP600 reactor coolant pumps prevents oil spills and fires. In addition, the AP600 containment does not have any confined spaces where a damaging hot gas layer could form. Due to the absence of risk initiators, containment fires were not considered in the PRA.

Please note that the other external events like high winds, tornadoes beyond the AP600 design basis as well as transportation and nearby facility accidents, etc. are screened out per the guidelines described in NUREG-1407 on the basis of hazard frequency being acceptably low (less than 1 x 10-6 per year). Such an assurance of low hazard frequency is obtained during the selection of the site and complying with the site selection criteria as described in the SSAR.

PRA Revision: NONE

Question 720.23

It is stated on page 2-4 of the PRA that air cooling of the containment is sufficient to maintain containment pressure below the yield stress, yet the success criteria for the passive cooling system (page C7-2) requires operation of one of two branches of the water supply system. Resolve this apparent inconsistency.

Response:

The success criteria discussed in section C-7 describes the success criteria for the design basis of the PCS system. However, it is true that air cooling of the containment is sufficient to prevent the containment from reaching the yield stress and from failing. The operation of the passive containment cooling system water has a significant effect on the containment pressure and is required for design basis accidents to keep the containment pressure below the design pressure. When it is not operating, the long-term containment pressure exceeds the design pressure, but remains below the containment ultimate capacity. The PCS node is on the containment event tree because the containment pressure, and therefore the PCS water operation, impacts two important downstream event tree results:

1. The success criteria for the ex-vessel debris coolability node is different when the PCS water is operational than the success criteria when the PCS water is not operational. Debris is considered to be coolable in the AP600 cavity when the PCS water is on and 3 out of 4 water sources (core makeup tanks and accumulators) or in-containment refueling water storage tank (IRWST) water have been injected into the reactor cavity. If PCS water is not operational, 4 out of 4 water sources or IRWST water are required to maintain water coverage of the ex-vessel debris. The difference is due to the amount of water that is present as steam in the pressurized containment atmosphere, and is therefore unavailable to be in the cavity pool cooling debris.

2. The fission product release from a containment for which the PCS water has maintained the long-term pressure below the design pressure is assigned to the OK release category. The release from a containment in which the long-term pressure exceeds the design pressure due to a failure of the PCS water is assigned to the OKP release category. The difference is due to the increased leakage from a pressurized containment. Since the offsite boundary doses for the release criteria are so small (1 rem and 25 rem), the difference between design basis leakage and pressurized leakage needs to be addressed in the containment event tree.

PRA Revision: NONE

Question 720.26

Provide additional information regarding the drains in the annular region, including: (1) the normal position, and means of actuation of any valves in the lines, (2) the size, number, and rated flow capacity of the lines, and (3) provisions for preventing blockage of the sump and lines.

Response:

Two annulus drains are located at low points in the upper annulus floor; the first located in the north sector of the shield building and the second located in the south sector. The drain sump is covered by a coarse screen to preclude large materials from blocking the pipe inlet. Each drain is always open (without isolation valves) to the storm drain system and sized to accept maximum PCS system flow assuming a full passive containment cooling water storage tank without consideration of any evaporation and with a minimum flooding of the annulus required. Additionally, if the storm drains are blocked, e.g. frozen, the interface between the annulus drain and the storm drain system is an open connection such that the annulus drain would simply overflow the connection into a catch basin or into a ground level storm drain to assure positive drainage.

SSAR Subsection 6.2.2.2.4 will be modified as follows:

SSAR Revision:

The cooling water not evaporated from the vessel wall flows down to the bottom of the inner containment annulus into floor drains. The drains route the excess water to storm drains. The drain line is always open (without isolation valves) and is sized to accept maximum passive containment cooling system flow. The interface with the storm drain system is an open connection such that any blockage in the storm drains would result in the annulus drains overflowing the connection.

Question 720.30

The review of the AP600 PRA indicates that MAAP 4.0 was used in the analysis. It is of concern to the staff that the code was used without support from more detailed mechanistic analysis for items that are controlling containment performance. In this regard, provide additional information regarding any supporting analyses performed to confirm the adequacy of the MAAP treatment of the following items: (1) external cooling of the reactor vessel, (2) temperature induced hot leg failure, (3) mode of reactor vessel lower head failure (creep rupture versus local failure), (4) early containment challenges, such as direct containment heating, and fuel coolant interactions, (5) hydrogen combustion, (6) coolability of core debris in the cavity, (7) fission product decontamination factors, and (8) molten core concrete attack and non-condensible gas generation.

Response:

(1) The MAAP 4.0 models calculate that the lower head of the reactor vessel will not fail when it is covered by water in the reactor cavity. Supporting analyses for the external cooling of the vessel lower head can be found in WCAP-13388, "AP600 Phenomenological Evaluation Summaries", section 2, "A Phenomenological Evaluation Summary on External Cooling of the RPV in Support of the AP600 Risk Assessment".

(2) The MAAP 4.0 analyses assumed that a hot leg failure would occur when the hot leg metal temperature exceeded 1100°K at an RCS pressure of 2500 psi. This criterion was determined in an analysis that is described in greater detail in the response to RAI 720.35.

(3)(4) MAAP 4.0 assumes a local creep failure of the vessel that is ablated as the debris is ejected. The initial radius of the failure is a user input. Because of the depressurization system, and hot leg temperature induced failure, vessel failures occur at low pressure, and the results are not very sensitive to vessel failure mode. High pressure mode of reactor vessel failure and direct containment heating are addressed in WCAP-13388, section 3, "A Phenomenological Evaluation Summary on High Pressure Melt Ejection and Direct Containment Heating in Support of the AP600 Risk Assessment".

(4) MAAP 4.0 assumes critical heat flux (CHF) quenching of debris in-vessel and ex-vessel. Supporting adiabatic quench calculations show that the peak pressure from quench is well below the ultimate pressure of the containment. Explosive fuel coolant interaction is addressed in WCAP-13388, section 1, "A Phenomenological Evaluation Summary on Steam Explosions in Support of the AP600 Risk Assessment".

(5) MAAP 4.0 treats hydrogen combustion using the same models as MAAP 3.0B. The code contains a hydrogen-air-steam flammability curve, which is used to determine the time at which a node may become flammable. Supporting hydrogen mixing and combustion analyses are presented in more detail in PRA report chapters 14, and 15, appendices N and O, and WCAP-13388, section 5, "A Phenomenological Evaluation Summary on the Probability and Consequences of Deflagration and Detonation of Hydrogen in Support of the AP600 Risk Assessment".

(6)(8) MAAP 4.0 treats debris coolability and molten core concrete interaction using the same models as MAAP 3.0B. Supporting analyses of debris coolability, core concrete interaction and non-condensible gas generation are

addressed in WCAP-13388, section 4, "A Phenomenological Evaluation Summary on Molten Core-Concrete Interaction in Support of the AP600 Risk Assessment.

(7) MAAP 4.0 treats fission product transport and deposition with the same non-dimensional empirical correlation that were developed for MAAP 3.0B. Supporting analyses for fission product decontamination factors are presented in WCAP-13388, "A Phenomenological Evaluation Summary on Fission Product Retention Capability in Support of the AP600 Risk Assessment".

PRA Revision: NONE

Question 720.35

Describe and justify the models, input assumptions, and failure criteria used in the AP600 hot leg creep rupture assessments. Discuss the influence of the AP600 improvements in vessel and piping reliability on these calculations.

Response:

In order to determine the reactor coolant system creep rupture criteria for the AP600, plant specific geometric information of the RCS piping and the Larson-Miller parameters (ref. 1) of 316 stainless steel are used to generate "time at temperature" creep rupture curves for the hot leg nozzle, cold leg nozzle, surge line and steam generator tubes. Although the steam generator tubes are Inconel 690, not stainless steel, the rupture curves are conservative because the Inconel has significantly greater strength than steel at high temperatures (ref. 2).

The analysis shows that the hot leg was the most likely place for the failure to occur. At a pressure of 2500 psia, all strength in the hot leg is lost at a temperatures greater than 1200°K. Creep rupture failure of the hot leg was modeled in MAAP 4.0 using a hot leg metal temperature threshold of 1100°K. When this temperature was reached, the hot leg was failed in the analysis. The plot of hot leg temperature and RV lower head temperature vs. time in the AP600 PRA does not show the temperature of the hottest of the two hot leg nozzles. This figure will be corrected in the document.

A major uncertainty in the temperature induced failure of the hot leg was the break size, which was estimated low in the base case to prevent the IRWST water from injecting and terminating the accident. This uncertainty was addressed in the MAAP4.0 sensitivity analyses.

References

1. Larson, F.R., Miller, J., "A Time-Temperature Relationship for Rupture and Creep Stress", Transactions of the American Society of Mechanical Engineers, pp. 765-775, July 1952.

2. Harrold, D.L., et. al., "The Temperature Dependence of the Tensile Properties of Thermally Treated Alloy 690 Tubing", Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Monterey, CA, August 25-29, 1991.

PRA Figure L-97 will be replaced with the following:

PRA Revision:

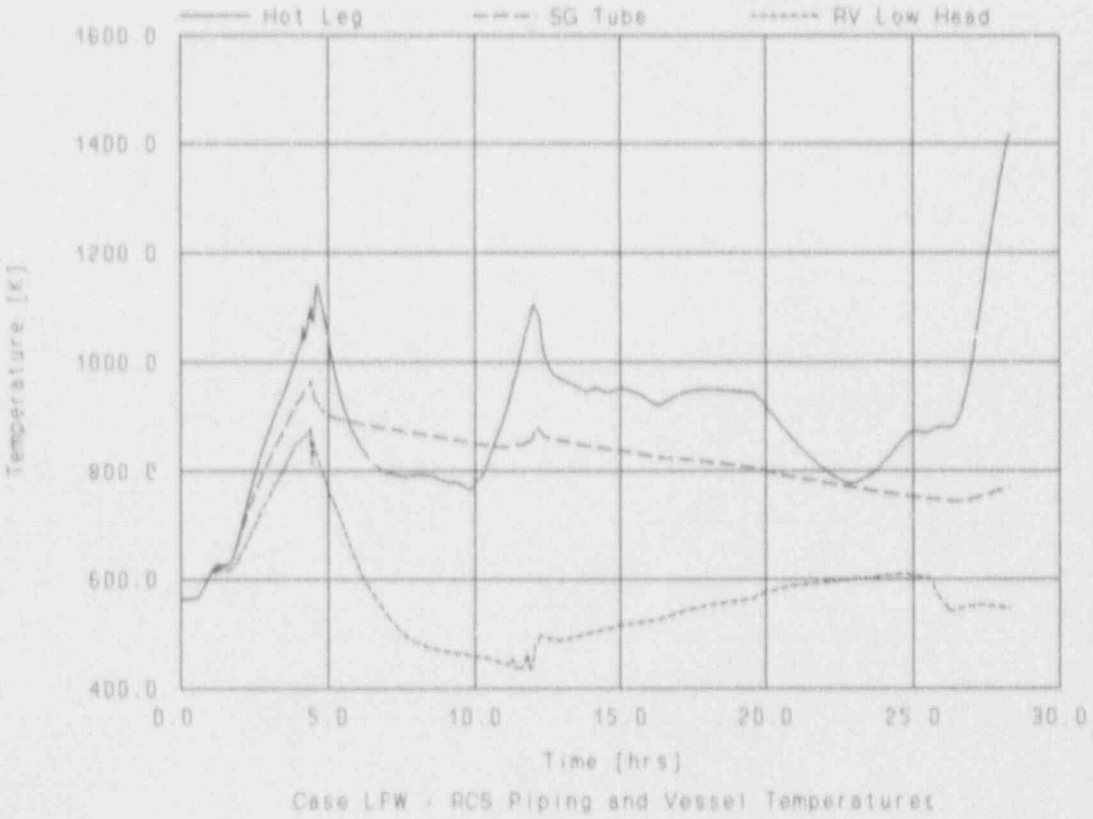
Hot Leg
SG Tube
RV Low Head
1600.0
1400.0
1200.0
1000.0
800.0
600.0
400.0
Temperature [K]
0.0
5.0
10.0
15.0
20.0
25.0
30.0
Time [hrs]
Case LFW - RCS Piping and Vessel Temperaturet

Figure L-97

Question 720.36

In addition to ADS, the PRA credits hot leg creep rupture as a means of reducing the occurrence of reactor vessel failure at high pressure. In the event of the failure of the ADS, the operator is instructed to flood the cavity under certain conditions. Describe the influence of the cavity flooding feature on the creep rupture of the hot leg. Of particular interest is the estimated probability and timing of creep rupture failure with and without cavity flooding, and how these values compare to the probability and timing of vessel failure under flooded cavity conditions.

Response:

The operator action to flood the cavity has little effect on the timing of the high temperature/pressure induced failure of the hot leg. The hot leg failure occurs during the period of rapid oxidation of the zircaloy cladding in the core (Figures L-90, L-94, and L-97). During this period, the core is just beginning to melt, and little, if any, of the fuel containing decay heat has relocated to the lower head of the reactor vessel (Figure L-99). Therefore, the reactor vessel lower head integrity is not jeopardized until well after the failure of the hot leg has occurred and the reactor coolant system is depressurized. These results are consistent with what is expected to occur since the temperature at which the cladding begins to rapidly oxidize ($\sim 1200°K$) is much lower than the melting temperature of the uranium-dioxide/zirconium eutectic ($\sim 2500°K$). If the lower head of the vessel is covered with water, then the reactor vessel will not eject debris into the containment. This position is supported in WCAP-13388, "AP600 Phenomenological Evaluation Summaries", section 2, "A Phenomenological Evaluation Summary on External Cooling of the RPV in support of the AP600 Risk Analysis".

In summary, in the AP600 PRA, for a high pressure RCS, in post-core uncovery conditions, the probability of a temperature induced hot leg failure prior to vessel failure is 1.0, independent of cavity flooding. The probability of vessel failure is 1.0 if the vessel has not been flooded by IRWST water in the cavity, and 0.0 if the IRWST water has not been injected. This treatment is consistent with the treatment of other phenomena on the containment event tree. The phenomena probabilities are 1.0/0.0, but are based on the availability of AP600 systems which have finite failure probabilities.

PRA Revision: NONE

Question 720.38

The staff is concerned with the potential for fuel-coolant interactions (FCIs) in the flooded reactor cavity. In this regard, provide the results of calculations that show the impact of ex-vessel FCIs in a flooded cavity on containment performance. In particular, demonstrate that the effects of an FCI or a rapid steam generation event do not damage equipment or structures required to maintain passive containment cooling. As part of this assessment, the potential for and consequences of ex-vessel FCIs for both low pressure and high pressure reactor vessel failure should be considered.

Response:

As a part of the AP600 PRA, an evaluation was carried out to examine the potential for and consequences from fuel-coolant interactions. It is documented in the Westinghouse document WCAP-13388, "AP600 Phenomenological Evaluation Summaries" in a section entitled "A Phenomenological Evaluation Summary on Steam Explosions in Support of the AP600 Risk Analysis." Apart from the containment shell, all the equipment and structures required to maintain passive containment cooling are outside the containment and, therefore, they are not exposed to the fuel-coolant interaction effects. Based on the evaluation described in WCAP-13388, it is concluded that loss of containment function due to a fuel-coolant interaction event is not credible for the AP600 design.

PRA Revision: NONE

Question 720.39

Provide an assessment of the axial and radial ablation of concrete that would occur in the event that water is not added to the reactor cavity.

Response:

The response of the AP600 containment design to axial and radial ablation of the concrete for the highly unlikely case of a dry reactor cavity is addressed in the Westinghouse document WCAP-13388, "AP600 Phenomenological Evaluation Summaries" in a section entitled "Phenomenological Evaluation Summary on Molten Core-Concrete Interaction in Support of the AP600 Risk Analysis." The assessment concludes that concrete ablation (MCCI) can be excluded as a credible containment failure mechanism due to the long time (days) required for a sustained dry condition to breach the AP600 containment. Details of the assessment are presented in WCAP-13388.

PRA Revision: NONE

Question 720.45

The fission product release fractions provided in Chapter 11 of the PRA appear inappropriate for accidents which bypass containment and do not benefit from fission product holdup and retention in the containment. In this regard, an additional release class(es) should be added to cover: (1) LOCAs outside containment, (2) interfacing system LOCAs, and (3) steam generator tube ruptures. Provide estimates of source terms for these 3 additional containment bypass sequences, and an assessment of these source terms on the AP600 risk profile.

Response:

The major goal of the PRA is to demonstrate that the AP600 meets the design criterion of limiting the mean whole body offsite dose at the site boundary to less than 25 rem at a frequency greater than 1.0E-6. The release categories were determined with this criterion in mind. Containment isolation failure, and containment bypass (SGTR and ISLOCA) both result in large source terms with significant consequences exceeding the dose criterion, and therefore were lumped together. The containment isolation release was used for the dose calculation since it represented the most likely release in the CI release category at 47% of the release category frequency. Steam generator tube rupture represents 20% of the release category frequency. The remaining 33% is excessive leakage (non-containment failure) sequences resulting in greater than 1 rem at the site boundary. LOCA outside containment and ISLOCA are not dominant initiating events in the AP600 PRA (see Table 7-1 of PRA report) since the interfacing systems which extend outside containment have an additional valve and are designed to better withstand the pressurization expected from multiple valve failures. The most likely SGTR sequences result in no releases to the environment (AP600 PRA report section L.2.6) but are classified as core damage events due to a conservative success criterion used in the SGTR core damage frequency calculation. A SGTR sensitivity case which results in large releases is analyzed and discussed in section L.3.7 of the PRA report. The excessive leakage release are much smaller than release from a containment isolation failure as they result in releases greater than 1 rem whole body at the site boundary. Therefore, the representative release in CI is appropriate, and significantly overestimates the releases for over half the sequences in the release category. No additional fission product source term estimates are required.

PRA Revision: NONE

Question 720.46

To better understand the transport and retention of fission products in the MAAP calculations, provide a breakdown of the distribution of fission products within the RCS, containment, environment, etc. for each release class in a manner similar to reported in Table 5.8 of NUREG/CR-4624, Volume 1.

Response:

The tables are attached in the following pages, and will be included in the PRA report in Chapter 11.

The last paragraph of the following Subsections will be revised as follows:

PRA Revision:

(Subsection 11.2.1)

Figures 11-1 through 11-12 present the fission product release fractions as functions of time for release category OK. Because of the influence of water in the containment, there is essentially no difference in fission product release if the debris remains in the vessel or is released to the containment. The final release fractions, at 24 hours after core damage, are presented in Table 11-1. The distribution of the fission products throughout the containment in case BC1 is summarized in Table 11-2.

(Subsection 11.2.2)

Figures 11-13 through 11-24 show the fission product release fractions as functions of time for release category OKP. Because of the influence of water in the containment, there is essentially no difference in fission product release if the debris remains in the vessel or is released to the containment. The final releases, at 24 hours after core damage, are presented in Table 11-1. The distribution of the fission products throughout the containment in case OKP is summarized in Table 11-3.

(Subsection 11.2.3)

Figures 11-25 through 11-36 show the fission product release fractions as functions of time for the CC release category. The final release fractions, at 24 hours after core damage, are presented in Table 11-1. The distribution of the fission products throughout the containment in case CC is summarized in Table 11-4.

(Subsection 11.2.4)

Figures 11-37 through 11-48 show the fission product release fractions as functions of time for the CI release category. The final release fractions, at 24 hours after core damage, are presented in Table 11-1. The distribution of the fission products throughout the containment in case CI is summarized in Table 11-5.

Table 11-2

Distribution of Fission Products By Group -- Release Category OK

Species	RCS	Debris	SG Rooms	Upper Compt	Lower Compt	Middle Annulus	Environment
Xe, Kr	2.8E-3	0.0	4.1E-2	8.4E-1	1.1E-1	9.2E-4	4.2E-5
CsI	3.8E-1	9.1E-4	5.5E-2	3.8E-1	1.7E-1	1.4E-5	5.6E-7
SrO	1.5E-2	9.7E-1	1.2E-3	8.8E-3	4.7E-3	7.5E-7	3.2E-8
MoO_2	2.6E-1	4.6E-1	2.5E-2	1.8E-1	9.4E-2	1.3E-5	5.6E-7
CsOH	3.7E-1	5.7E-4	5.5E-2	3.9E-1	1.7E-1	1.6E-5	5.8E-7
BaO	1.4E-1	7.3E-1	1.2E-2	8.3E-2	4.4E-2	6.9E-6	2.9E-7
La_2O_3	8.0E-3	9.8E-1	7.8E-4	5.5E-3	3.0E-3	4.8E-7	2.0E-8
CeO_2	1.9E-2	9.6E-1	2.1E-3	1.5E-2	8.2E-3	1.4E-6	5.9E-8
Sb	1.7E-1	2.6E-1	3.7E-2	4.0E-1	1.3E-1	3.7E-5	1.0E-6

Table 11-3

Distribution of Fission Products By Group -- Release Category OKP

Species	RCS	Debris	SG Rooms	Upper Compt	Lower Compt	Middle Annulus	Environment
Xe, Kr	2.9E-3	0.0	4.2E-2	8.4E-1	1.1E-4	9.6E-4	1.1E-4
CsI	4.5E-1	1.2E-3	4.6E-2	3.4E-1	1.5E-1	3.6E-5	2.0E-6
SrO	3.5E-2	9.5E-1	1.3E-3	8.7E-3	4.7E-3	1.0E-6	8.0E-8
MoO_2	4.6E-1	2.8E-1	2.3E-2	1.5E-1	8.1E-2	1.3E-5	9.6E-7
CsOH	4.3E-1	1.2E-3	4.8E-2	3.6E-1	1.6E-1	3.8E-5	2.0E-6
BaO	2.8E-1	5.9E-1	1.2E-2	7.6E-2	4.1E-2	8.4E-6	6.5E-7
La_2O_3	2.7E-2	9.7E-1	6.4E-4	4.5E-3	2.4E-3	6.9E-7	5.5E-8
CeO_2	6.8E-2	9.1E-1	2.2E-3	1.5E-2	8.0E-3	2.0E-6	1.6E-7
Sb	2.7E-2	1.7E-1	5.8E-2	5.1E-1	2.2E-1	8.8E-5	4.8E-6

Table 11-4

Distribution of Fission Products By Group -- Release Category CC

Species	RCS	Debris	SG Rooms	Upper Compt	Lower Compt	Middle Annulus	Environment
Xe, Kr	2.8E-3	0.0	4.2E-2	8.3E-1	1.1E-1	9.6E-4	6.5E-5
CsI	5.0E-1	0.0	4.9E-2	3.0E-1	1.5E-1	1.5E-5	7.9E-7
SrO	2.2E-2	9.5E-1	1.5E-3	9.4E-3	5.2E-3	7.6E-7	4.9E-8
MoO_2	3.6E-1	3.9E-1	2.5E-2	1.5E-1	8.4E-2	1.0E-5	6.5E-7
CsOH	4.6E-1	0.0	5.1E-2	3.2E-1	1.6E-1	2.0E-5	9.0E-7
BaO	2.0E-1	6.6E-1	1.3E-2	8.2E-2	4.6E-2	6.5E-6	4.2E-7
La_2O_3	1.6E-2	9.7E-1	9.0E-4	5.7E-3	3.2E-3	4.8E-7	3.1E-8
CeO_2	3.7E-2	8.9E-1	3.2E-3	2.0E-2	1.1E-2	1.6E-6	1.1E-7
Sb	3.7E-1	1.4E-1	3.8E-2	2.5E-1	1.3E-1	2.2E-5	1.1E-6

Table 11-5

Distribution of Fission Products By Group -- Release Category CI

Species	RCS	Debris	SG Rooms	Upper Compt	Lower Compt	Middle Annulus	Environment
Xe, Kr	9.0E-2	0.0	9.7E-3	5.1E-1	4.4E-2	1.4E-5	3.4E-1
CsI	1.9E-1	1.6E-3	5.7E-1	2.0E-1	6.3E-3	5.6E-7	3.7E-2
SrO	5.1E-3	9.6E-1	3.7E-2	1.1E-3	7.8E-5	1.3E-9	6.7E-5
MoO_2	6.1E-2	4.5E-1	4.7E-1	1.7E-2	8.0E-4	1.9E-8	1.5E-3
CsOH	1.8E-1	1.6E-3	5.8E-1	2.0E-1	5.1E-3	4.1E-7	3.7E-2
BaO	3.7E-2	6.8E-1	2.7E-1	7.3E-3	5.1E-4	9.2E-9	4.8E-4
La_2O_3	2.0E-3	9.8E-1	1.3E-2	3.0E-4	2.6E-5	4.8E-10	2.0E-5
CeO_2	3.0E-3	9.7E-1	2.3E-2	4.7E-4	3.7E-5	6.9E-10	2.8E-5
Sb	4.7E-2	3.3E-1	6.1E-1	6.3E-3	1.4E-4	9.7E-10	1.1E-3

Question 720.47

For the CI release category, provide a time dependent radionuclide distribution for the containment volumes and primary system. Show the decontamination factors for primary coolant system and containment. Indicate the extent to which the various deposition mechanisms contribute to decontamination. In particular, explain why are the xenon and krypton release fractions less than 100% for a failed containment. The 24 hour source term calculation for this accident needs to be justified.

Response:

The time dependent distribution of the fission products is included as Tables 720.47-1 through 720.47-4, each for different times in the analysis of case LFW, which was the representative sequence for release category CI. The four times in the table relate to: 4 Hours - after core damage and before the temperature induced hot leg failure, 8 Hours - after the temperature induced hot leg failure, 20 Hours - long-term releases, 28 Hours - end of the analysis (~24 hours after initial core damage)

The tables show the fraction of the noble gas, volatile and non-volatile fission products in the core debris (core region and lower head of the reactor vessel), the RCS piping, the various compartments of the containment, and environment. In the tables, noble gases are represented by the Xe, Kr fission product group, volatile aerosols are represented by the CsI fission product group and the non-volatile aerosols are represented by the SrO fission product group. Also included in the tables are the decontamination factors of the RCS and the containment. The individual deposition rates for the various aerosol removal mechanisms are not available from the MAAP 4.0 output, but the overall deposition rates are available, expressed as aerosol deposition lambdas. For the LFW case, the lambdas are:

at hot leg failure	$\lambda = 3.0\ hr^{-1}$
after hot leg failure	$\lambda = 0.5\ hr^{-1}$

The reason that the noble gas release is less than 100% for the CI release category is that the containment pressure is very near to atmospheric pressure for most of the accident time after core damage, and therefore, the driving force for containment leakage is low. Most of the release occurs as the hot leg fails due to high temperature and pressure creep rupture, fission product are released to the containment, and simultaneously the containment is pressurized. After the pressure subsides, much of the decay heat goes into heating RCS metal and into the cavity water which is subcooled water injected from the IRWST. Therefore, long-term containment pressurization is slow, and the rate of the release of fission products to the environment after the failure of the hot leg is slow.

The 24 hour release is justified since the goal of the PRA is to demonstrate that mean whole body offsite doses greater than 1 rem at the site boundary have a frequency less than 1.0×10^{-6} per reactor-year. At 24 hours after core damage, the containment isolation failure cases have significantly exceeded a 1 rem offsite dose, and therefore, the fission product source term analysis no longer needs to be continued. However, the AP600 meets the above criterion as the frequency of the CI release category is 3.0×10^{-8} per reactor-year.

PRA Revision: NONE

Table 720.47-1

Fission Product Distribution - Release Category CI
Time = 4 Hours, Prior to Hot Leg Failure

Species	Debris	RCS Piping	SG Rooms	Lower Compt	Upper Compt	Mid Annulus	Enviro	RCS DF	Cont DF
Nobles	8.6E-1	1.1E-1	8.0E-4	2.0E-3	2.4E-3	8.8E-8	1.2E-3	5.0	23.5
Volatile	9.1E-1	8.6E-2	4.3E-5	8.3E-5	9.7E-4	3.0E-9	4.1E-5	76.6	27.7
Non-volatile	1.0	0.0	0.0	0.0	0.0	0.0	0.0	-	-

Table 720.47-2

Fission Product Distribution - Release Category CI
Time = 8 Hours, After Hot Leg Failure

Species	Debris	RCS Piping	SG Rooms	Lower Compt	Upper Compt	Mid Annulus	Enviro	RCS DF	Cont DF
Nobles	2.9E-1	4.9E-4	1.0E-2	3.5E-2	5.1E-1	3.8E-6	1.6E-1	1.0	4.4
Volatile	2.6E-1	3.0E-1	2.1E-1	5.5E-3	1.9E-1	5.5E-7	3.6E-2	1.7	12.2
Non-volatile	9.9E-1	4.5E-4	4.9E-3	3.2E-6	2.2E-4	9.6E-11	1.7E-5	1.1	310

Table 720.47-3

Fission Product Distribution - Release Category CI
Time = 20 Hours, Long Term Release

Species	Debris	RCS Piping	SG Rooms	Lower Compt	Upper Compt	Mid Annulus	Enviro	RCS DF	Cont DF
Nobles	9.0E-2	0.0	1.1E-2	5.0E-2	5.6E-1	1.1E-5	2.8E-1	1.0	3.2
Volatile	1.6E-3	2.0E-1	5.6E-1	6.2E-3	2.0E-1	5.6E-7	3.7E-2	1.3	21.5
Non-volatile	9.6E-1	4.5E-3	3.2E-2	4.6E-5	8.2E-4	8.1E-10	4.8E-5	1.1	675

Table 720.47-4

Fission Product Distribution - Release Category CI
Time = 28 Hours, 24 Hours After Core Damage

Species	Debris	RCS Piping	SG Rooms	Lower Compt	Upper Compt	Mid Annulus	Enviro	RCS DF	Cont DF
Nobles	9.0E-2	0.0	9.7E-3	4.4E-2	5.1E-3	1.4E-5	3.4E-1	1.0	2.6
Volatile	1.6E-3	1.9E-1	5.7E-1	6.2E-3	2.0E-1	5.6E-7	3.7E-2	1.2	21.9
Non-volatile	9.6E-1	5.1E-3	3.7E-2	7.8E-5	1.1E-3	1.3E-9	6.7E-5	1.1	556

Question 720.50

On page 10-1 of the PRA, it is stated that NUREG-1335 was used as a guide for selecting model parameters to be varied in the sensitivity studies. Clarify whether the EPRI guidance document on MAAP sensitivity analyses for the IPE was also used, and why it was not, if that is the case.

Response:

The EPRI guidance document "Recommendations on the Use of the MAAP 3.0B Code in Individual Plant Evaluations" was used to a certain extent in the selection of the AP600 PRA sensitivity studies. However, the PRA was performed with MAAP 4.0, not MAAP 3.0B, and the AP600 does not always respond in the same manner as a conventional plant, for which the guidance was written. For example, the discussions of sensitivity due to the MAAP 3.0B core blockage model do not apply to MAAP 4.0, since there is no core blockage model in MAAP 4.0. Similarly, the sensitivity to containment failure parameters do not apply to AP600, since the containment integrity is not threatened in sequences in which the containment is not failed prior to, or at the initiation of an accident.

Analyses to address the severe accident issues recommended in the guidance document have been performed in the PRA. Specifically, the sensitivities of MAAP 4.0 parameters which affect the rate, or occurrence of core-concrete interaction, in-vessel hydrogen generation, and hydrogen combustion were performed. Supporting analyses were performed to address timing and location of high pressure and temperature induced failures of reactor coolant system piping. The above mentioned and other severe accident issues, such as high pressure melt ejection, fuel coolant interactions, and fission product deposition are addressed in WCAP-13388, "AP600 Phenomenological Evaluation Summaries". In addition, in the WCAP, external cooling of the reactor vessel lower head is also addressed, and not specifically recommended in the EPRI guidance document.

PRA Revision: NONE

Question 720.57

Confirm that the June 26, 1992 PRA reflects all of the changes made to the AP600 design, as presented in the SSAR submitted on June 26, 1992, or identify the differences between the design assumed in the PRA and that of the design application.

Response:

There are cases where differences exist between the design assumed in the PRA model and the design described in the AP600 SSAR. These differences, and projected effects, are identified in Appendicies C8, C9, C11, C12, C13, C15, C16, C17, C21, and C22 of the PRA report.

PRA Revision: NONE

Question 720.58

To perform confirmatory analyses of the Westinghouse AP600 PRA results, the staff is planning to upload the Westinghouse AP600 PRA onto the IRRAS computer program. To upload the Westinghouse PRA onto IRRAS, the staff needs to have the following files on electronic media in ASCII format, unless otherwise stated. Provide this information.

a. For all of the fault trees generated from Grafter, the staff needs to have all of the treename.txt and treename.dat files converted to SETS input using the SETSIN2 program as described in the GRAFTER Users Manual (WCAP-11693). The subtrees designated as SUB-XXXX also need to be converted to SETS input. The staff also needs a copy of all of the fault tree output files from GRAFTER (treename.cut), including the SUB-XXXX subtrees.

b. The staff needs the Master Data file from GRAFTER used to quantify the basic events that are described in the fault trees. The staff believes that the file is called SIMON.DAT or SIMON.CUT.

c. Based on conversations with Westinghouse, the GRAFTER computer output given to the staff contains incomplete system cutsets and incomplete system unavailabilities. These systems may contain basic events that are designated as SUB-XXXX that represent smaller subtrees that are given dummy probabilities. The staff understands that these system fault trees are reduced (the SUB-XXXX basic events are replaced with cutsets) in the SUBA option in the WLINK computer code. Therefore, the staff needs the fault trees output after the SUBA option is used in WLINK that reduces all of the SUB-XXXX events. The staff believes that these are treename.wlk files. The staff also needs to have a copy of the accident sequence output files from the SEQ OPTION in WLINK. The staff believes that they are called XXXX.out files.

Response:

The requested files were provided on diskettes in letter ET-NRC-92-3774 on November 25, 1992.

PRA Revision: NONE