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Energy Systems

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NTD-NRC-94-4249
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Docket No.: STN-52-003

July 29, 1994

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

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclos re three copies of the Westinghouse responses to NRC requests for additional information on the ,. 600 from your letters of April 7, 1994, April 15, 1994, April 29, 1994, May 5, 1994, May 11, 1994, May 16, 1994, May 23, 1994 and June 8, 1994. In addition, revisions of responses previously submitted are provided.

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

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

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

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory And Licensing Activities

/nja

Enclosure

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

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NTD-NRC-94-4249
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JULY 29 1994

RAI No.	Issue
210.090	COL applicant process for seismic interaction
230.063R02	Soil column properties for horiz. & vert. models
260.020	Revision to SSAR Table 16.2-1
410.110	SWS radioactive leakage detection, control, isol.
410.112	SWS valve position instrumentation
410.128	TB CCWS temperature indications
410.133	TC CCWS isophase bus cooling
410.136	Steam & power conversion system protective feature
410.144	URD requirements for TG
410.150	Auxiliary steam system codes & standards
410.151	Auxiliary steam system initial test program
410.153	Compressed & instrument air system contaminants
410.158	Figure 9.3.1-1
410.159	Conformance of CAS to ANSI/ISA-S7.3-1975 (R1981)
410.161	CAS testing
410.163	Agreement of text and figures in Section 9.2.9
410.176	Fuel oil S&T system illustration
410.185	Flood levels, methods of draining
410.187	Effects of CWS piping failure
410.190	SFCV closure actuation & leakage test requirements
410.208	Protection of safety-related SSCs
410.213	Protection of defense-in-depth equipment
410.219	Safe-shutdown equipment in containment
410.243	NI non-radioactive ventilation system
435.075	Regulatory oversight of offsite power sources
440.054	Potential rapid boron dilution scenarios

NTD-NRC-94-4249
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JULY 29 1994

RAI No.		Issue
440.058	:	Tech Spec changes to deal with shutdown operations
440.078	:	LTOP analysis
440.122	:	DWS makeup isolation actuation logic
440.128	:	RNS capabilities with single failure
440.131	:	RNS design requirements under RTNSS
440.136	:	PXS performance upon RNS operation
440.181	:	Time available for manual actuation of ADS
440.190	:	Effects of design changes on PRA success criteria
440.196	:	Success paths for MSLB upstream of MSIV
440.198	:	Credit for PRHR control in RTNSS evaluation
440.199	:	Recirculation in tree SIS
480.055	:	Inidentified NRHR penetrations
480.076	:	Containment penetrations beyond "state of art"
720.262R01	:	Spectrum shape used in seismic margins analysis
720.263R01	:	SSC fragilities and HCLPFs
720.264R01	:	Plant HCLPF
720.265R01	:	Failure of non-seismically qualified SSCs
720.266R01	:	Effect of seismic failure of non-seismic equipment
720.268R01	:	Plant HCLPF during shutdown
720.269R01	:	Seismically induced ATWS events
720.270R01	:	Systems modelled in seismic margins analysis
720.271R01	:	Initiating events with HCLPF greater than 0.5g

Question 210.90

The criteria in Sections 3.7.2.8 and 3.7.3.13 of the SSAR, relative to protecting certain SSCs from adverse seismic interactions, are used for the design of the AP600. However, during the construction phase, interferences from field run items may lead to such interactions. To identify and correct such potentially adverse interactions, provide a statement in the SSAR that the COL applicant should describe the process for completion of the design of balance-of-plant and non-safety systems to minimize II/I interactions and propose procedures to be used for performing an assessment of the as-built plant to verify that the interaction of non-seismic SSCs with seismic SSCs does not affect the safety function of the seismic SSCs.

Response:

The criteria for protection against seismic interaction are defined in SSAR Subsection 3.7.3.13. The design will be pre-engineered and there will be few, if any, field run items. The criteria are also applicable to the process for completion of detailed design, including reconciliation of as-procured equipment. SSAR Subsection 3.7.3.13 is revised as shown below to include the confirmatory analyses performed by the Combined License applicant. Subsection 3.7.3.13.2 is revised to include the criteria to determine if a nonseismic source is located in a manner where its failure could result in impact on a safety-related item.

SSAR Revision:

Add item No. 3.13 in Table 1.8-1 (Sheet 3 of 8) as follows:

3.13	Seismic interaction review	Requirement of AP600	Combined License applicant program	3.7.3.13

Revise Subsections 3.7.3.13 through 3.7.3.13.2 as shown below:

3.7.3.13 Interaction of Other Systems with Seismic Category I Systems

The safety functions of seismic Category I structures, systems, and components are protected from interaction with nonseismic structures, systems, and components; or their interaction is evaluated. **The safety-related systems and components required for safe shutdown are described in Section 7.4. This equipment is located in selected areas of the auxiliary building and inside containment.** The primary means of protecting safety-related structures, systems, and components from adverse seismic interactions are discussed in the following paragraphs in the order of preference.

Interaction of ~~other~~ **connected** systems with seismic Category I piping is considered by including the other piping in the analysis of the seismic Category I system. This is discussed in Subsection 3.7.3.13.3.

The containment and each room outside containment containing safety-related systems or equipment, as identified in Table 3.7.3-1, are reviewed for potential adverse seismic interactions to demonstrate that systems, structures, and components are not prevented from performing their required safe shutdown functions. In addition, the review identifies the protection features required to mitigate the consequences of seismic interaction in an area that contains safety-related equipment. The seismic interaction review is performed on a room-by-room basis and includes the following steps:

- Identification of the non-seismic sources
- Identification of essential equipment in the area
- Evaluation of effects on essential equipment.

The three-dimensional computer model and composites developed for the nuclear island are used during the design process of the systems and components in the nuclear island, to aid in evaluating and documenting the review for seismic interactions. This ~~detailed~~ review is performed using the design criteria and guidelines ~~that follow~~ described in Subsections 3.7.13.1 through 3.7.13.3.

The seismic interaction review will be updated by the Combined License applicant. This review is performed in parallel with the seismic margin evaluation (see PRA Report, Appendix H, Subsection H.2.5). The review is based on as-procured data, as well as the as-constructed condition.

3.7.3.13.1 Separation and Segregation

Separation - The general plant arrangement provides physical separation between the seismic Category I and nonseismic structures, systems, and components to the maximum extent practicable in the nuclear island. The objective is to assist in the preclusion of a potential adverse interaction if the nonseismic structures, systems and components were to fail during a seismic event. Whenever possible nonseismic pipe, electrical raceway, or ductwork is not routed above or adjacent to safety-related equipment, pipe, electrical raceway, or ductwork thereby eliminating the possibility of seismic interaction.

Segregation - Where separation by physical means cannot be accomplished and it becomes necessary to locate or route nonseismic structures, systems, and components in or through~~, as the case may be,~~ safety-related areas, the nonseismic structures, systems and components are segregated from the seismic Category I items to the extent practicable.

3.7.3.13.2 Impact Analysis and Seismic Category II

Impact Analysis - A specific nonseismic structure, system or component identified as a source to a specific safety-related component can be acceptable without being classified as seismic Category II~~. Or it can be relocated~~ if an analysis demonstrates that the weight and configuration of the source, relative to the target, and the trajectory of the source are such that the interaction would not cause unacceptable damage to the target. For example, a nonseismic instrument tube routed above a seismic Category I electrical cable tray would not pose a hazard and would be acceptable ~~without additional supports~~.

Nonseismic equipment can overturn as a result of a safe shutdown earthquake. The trajectory of its fall is evaluated to determine if it poses a potential impact hazard to a safety-related structure, system, or component. If it poses a hazard, the equipment is relocated, or it is classified and supported as seismic Category II.

Nonseismic walls, platforms, stairs, ladders, grating, handrail installations, or other structures next to safety-related structures, systems, and components are evaluated to determine if their failure is credible.

Should a nonseismic structure, system, or component be capable of being dislodged from its supports, the trajectory of its fall is evaluated for potential adverse impacts. If these present a hazard, the structure, system or component is relocated or classified and supported as seismic Category II. Impact is assumed for sources within an impact evaluation zone around the safety-related equipment. The impact evaluation zone is defined as the envelope around the target for which a source, if located outside of the envelope, would not impact the target during an SSE

in the event the supports of the source were to fail and allow the source to fall. The impact evaluation zone is defined by the volume extending 6 ft horizontally from the perimeter of the Seismic Category I object up to a height of 35 ft. The impact evaluation zone above 35 ft is defined by a 10-degree cone radiating vertically from the foot of the object, projected from its perimeter. This definition of the impact evaluation zone is illustrated in Figure 3.7.3-1. The impact evaluation zone need not extend beyond seismic Category I structures such as walls or floor slabs.

Seismic Category II

Where the preceding approaches of separation, segregation, or impact analysis cannot prevent unacceptable interaction, the source is classified and supported as seismic Category II. The seismic Category II designation provides confidence that these nonseismic structures, systems, and components can withstand the forces of a safe shutdown earthquake in addition to the loading imparted on the seismic Category II supports due to failure of the remaining nonseismically supported portions. However, the functionality of these seismic Category II sources does not have to be maintained following a safe shutdown earthquake.

Table 3.7.3-1

Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12101	Division A battery room 1	Batteries
12102	Division C battery room 1	Batteries
12103	Spare battery room	Spare batteries
12104	Division B battery room 1	Batteries
12105	Division D battery room 1	Batteries
12113	Spare battery charger room	
12162	RNS pump room A	RNS pressure boundary
12163	RNS pump room B	RNS pressure boundary
12201	Division A DC equipment room	DC equipment
12202	Division C battery room 2	Batteries
12203C	Division C DC equipment room	DC equipment
12203B	Division B DC equipment room	DC equipment
12204	Division B battery room 2	Batteries
12205	Division D DC equipment room	DC equipment room
12211	Corridor	Divisional cables
12212	RCP trip switchgear room B	RCP trip switchgear
12253	Pipe chase	RNS containment isolation valves
12255	CVS makeup pump room	CVS isolation valves Divisional cable to reactor trip swgr
12256	Lower annulus	CVS/WLS containment isolation valves RNS piping
12257	Pipe chase	CVS/WLS containment isolation valves RNS piping

Room No.	Room Name	Equipment Description
12300	Corridor	Divisional cable
12301	Division A I&C room	Divisional I&C
12302	Division C I&C room	Divisional I&C
12303	Remote shutdown workstation	Remote shutdown workstation
12304	Division B I&C/penetration room	Divisional I&C/electrical penetrations
12305	Division D I&C/penetration room	Divisional I&C/electrical penetrations
12306	Valve/piping penetration room	CCS/CVS/DWS/FPS/SGS containment isolation valves
12311	Corridor	Divisional cabling
12312	RCP trip switchgear room A	RCP trip switchgear
12313	Division C I&C/penetration room	Divisional I&C/electrical penetrations
12321	I&C/non 1E penetration room	Divisional cabling
12351	Maintenance floor and staging area	Divisional cabling (ceiling)
12352	Personnel hatch	Personnel airlock (interlocks)
12354	Rad pipe chase	PSS/SFS containment isolation valves
12356	Middle annulus	Class 1E electrical penetrations Various mechanical piping penetrations
12362	RNS HX room A	RNS pressure boundary
12363	RNS HX room B	RNS pressure boundary
12400	Control room vestibule	Control room access
12401	Main control room	Main control panels VBS HVAC dampers VES isolation valves Lights
12404	Lower MSIV compartment B	SGS containment isolation valves, instrumentation and controls
12405	VBS B and D equipment room	VWS/PXS/CAS containment isolation valves

Room No.	Room Name	Equipment Description
12406	Lower MSIV compartment A	SGS containment isolation valves, instrumentation and controls
12412	Electrical penetration room Division A	Divisional electrical penetrations
12421	RCC/non 1E penetration room	Divisional cabling
12422	Reactor trip switchgear II	Reactor trip switchgear
12423	Reactor trip switchgear I	Reactor trip switchgear
12452	VFS penetration room	VFS containment isolation valves, divisional cabling
12454	Rad pipe chase	SFS/PSS/VFS/CVS cont. isolation valves
12504	Upper MSIV compartment B	SGS CIVs, instrumentation and controls
12506	Upper MSIV compartment A	VWS/PXS/CAS containment isolation valves
12552	Personnel hatch	Personnel airlock (interlocks)
12553	Operating deck staging area	VES high pressure air bottles
12556	Upper annulus	PCS piping and cabling PCS air baffle
12561	Fuel handling area	Spent fuel storage racks
12701	PCS valve room	PCS isolation valves/instrumentation
	PCS Water storage tank	Level and temperature instrumentation

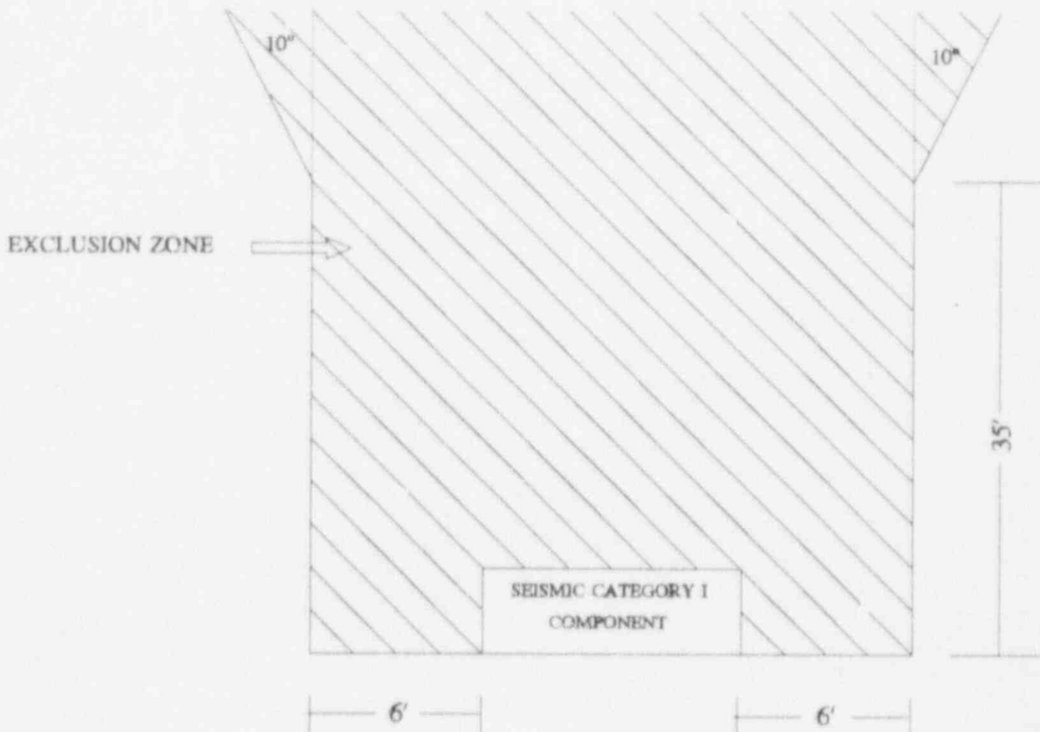


Figure 3.7.3-1
Impact Evaluation Zone

Question 230.63

The soil column properties for horizontal and vertical (P-wave) models are not consistent. Specifically, (a) the damping ratio for S & P wave motions are different, and (b) Poisson's ratio for soils above the ground water table appear to be too high. Provide an explanation to justify (a) the use of same properties for the horizontal and vertical models, and (b) the use of a high Poisson's ratio in the analysis.

Response: (Revision 1)

The strain-compatible soil/rock shear modulus and damping (for shear wave) were obtained from the average properties of two SHAKE analysis results of the respective soil/rock profile using H1 and H2 time histories as described in Section 2A.4 of the SSAR. These properties shown in Tables 2A-9 through 2A-12 were subsequently used in the SSI analysis. Along with these properties, the P-wave velocity and the damping associated for P-wave were obtained and used in the SSI analyses. The SSAR revision identified below provides the requested explanation.

a. ~~The generic strain dependent shear wave damping curves for soil/rock materials were obtained from a collection of laboratory test results as a function of shear strain amplitude. No comprehensive study to measure P-wave damping has been conducted. It is generally assumed that P-wave and S-wave damping are the same. This assumption was used for AP600. The following study was performed to evaluate the effect of P-wave damping.~~

~~The soft to medium soil profile with P-wave velocity of 5,000 ft/sec was analyzed using SHAKE in two cases. In Case 1, the P-wave damping was assumed to be the same as the S-wave damping (shown on Table 2A-10). In Case 2, the P-wave damping was computed using~~

$$\cancel{\beta_p = \frac{4}{3}\left(\frac{V_s}{V_p}\right) \beta_s}$$

~~The above relationship is based on zero dissipation for volumetric changes. The values for βs and~~ ~~V_s~~ ~~were obtained from Table 2A-10 and~~ ~~V_p~~ ~~was assumed to be 5,000 ft/sec. If~~ ~~β_p~~ ~~was found to be less than 0.1 percent, the minimum of 0.1 percent was used. The response spectra in the free-field at the depth of 40 ft corresponding to the basemat elevation are compared in Figure 230.63-1. The results are almost identical such that the curves for the two cases are essentially superimposed. The results are not affected by the P-wave damping ratio. This is due to the fact that the soil column frequency at 40 ft depth is larger than 31 Hertz and the input motion at ground surface is effectively retained at the basemat level in the free field. For this reason, use of smaller P-wave damping values is not expected to affect SSI responses significantly.~~

b. ~~The P-wave velocity for each layer in the soil/rock profile was obtained from the strain-compatible shear wave velocity and the Poisson's ratios shown in Section 2A.4. The calculated P-wave velocities are also shown in Tables 2A-9 through 2A-12. Depending on the SSI case and depth to the water table, the Poisson's ratios of the submerged layers were adjusted, if necessary, to maintain the P-wave velocity of the water (5,000 ft/sec). This adjustment of P-wave velocity and hence, the Poisson's ratio, is needed to reflect the P-wave propagation~~

~~speed in saturated media. The Poisson's ratio for layers above the water table are typical values appropriate for each respective soil profile. The SSI results, particularly the horizontal responses, are believed to be insensitive to the change of Poisson's ratio. On the other hand, the vertical responses for each soil/rock case were governed by the respective shallow water table case due to the fact that use of P wave velocity of water results in less attenuation of motion with depth, thus resulting in large effective foundation motion. The parametric SSI study on depth to the water table concluded that the water table at grade level is the governing condition for each respective generic soil profile analyzed. For these cases, the Poisson's ratio is assumed such that the P wave velocity of the water is maintained.~~

SSAR Revision: (Revision 1)

Add the following paragraphs in Section 2A.4 at the end of the subsection titled "Free-Field Analysis Cases":

The generic strain-dependent shear wave damping curves for soil/rock materials were obtained from a collection of laboratory test results as a function of shear strain amplitude. No comprehensive study to measure P-wave damping has been conducted. P-wave and S-wave damping are assumed to be the same. A study was performed to evaluate the effect of P-wave damping. The soft-to-medium soil profile with P-wave velocity of 5,000 ft/sec was analyzed using SHAKE in two cases. In Case 1, the P-wave damping was assumed to be the same as the S-wave damping (shown on Table 2A-10). In Case 2, the P-wave damping was computed using

$$\beta_P = \frac{4}{3}\left(\frac{V_S}{V_P}\right) \beta_S$$

This relationship is based on zero dissipation for volumetric changes. The values for β_s and V_s were obtained from Table 2A-10 and V_p was assumed to be 5,000 ft/sec. If β_p was found to be less than 0.1 percent, the minimum of 0.1 percent was used. The response spectra in the free-field at the depth of 40 ft corresponding to the basemat elevation are compared in Figure 2A-32. The results are almost identical such that the curves for the two cases are essentially superimposed. The results are not affected by the P-wave damping ratio. This is due to the fact that the soil column frequency at 40 ft depth is larger than 31 Hertz and the input motion at ground surface is effectively retained at the basemat level in the free-field. For this reason, use of smaller P-wave damping values is not expected to affect SSI responses significantly.

The P-wave velocity for each layer in the soil/rock profile was obtained from the strain-compatible shear wave velocity and the Poisson's ratios shown in Section 2A.4. The calculated P-wave velocities are also shown in Tables 2A-9 through 2A-12. Depending on the SSI case and depth to the water table, the Poisson's ratios of the submerged layers were adjusted, if necessary, to maintain the P-wave velocity of the water (5,000 ft/sec). This adjustment of P-wave velocity and hence, the Poisson's ratio, is needed to reflect the P-wave propagation speed in saturated media. The Poisson's ratio for layers above the water table are typical values appropriate for each respective soil profile. The SSI results, particularly the horizontal responses, are believed to be insensitive to the change of Poisson's ratio. On the other hand, the vertical responses for each soil/rock case were governed by the respective shallow water table case due to the fact that use of P-wave velocity of water results in less attenuation of motion with depth, thus resulting in large effective foundation motion. The parametric SSI study on depth to the water table concluded that the water table at grade level is the governing condition for each respective generic soil profile analyzed. For these cases, the Poisson's ratio is assumed such that the P-wave velocity of the water is maintained.

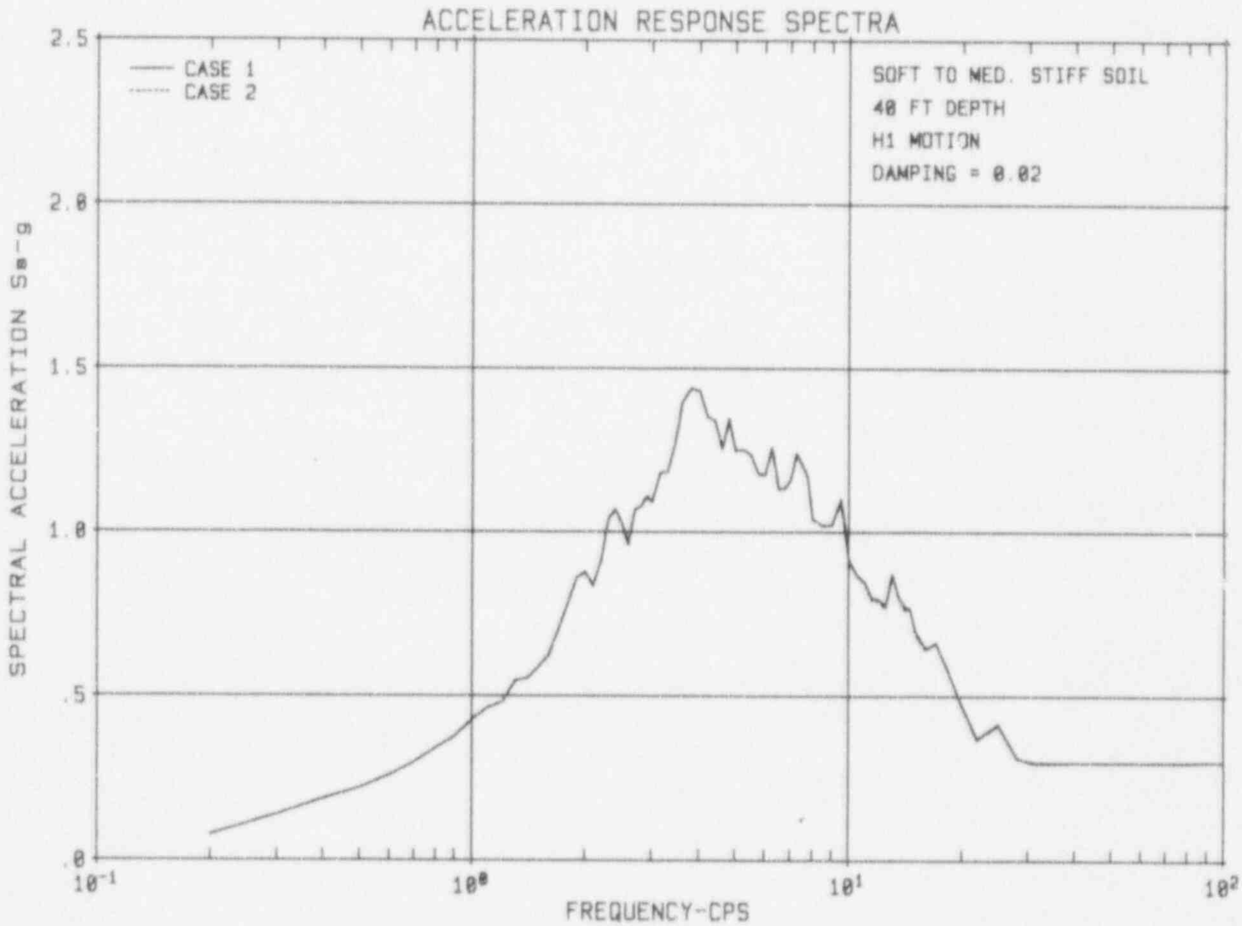


Figure 2A-32
Effect of P Wave Damping

Question 260.20

In response to Q260.12, Westinghouse states that Revision 1 to the standard safety evaluation report (SSAR) includes revisions to Table 16.2-1 to reflect the implementation of the process on the regulatory treatment of non-safety-related systems. Revision 1 deletes this table. Clarify this response.

Response:

In Revision 1 of the SSAR the reference in Subsection 16.2.1.1 to Table 16.2-1 of the SSAR was replaced by a reference to WCAP-13856 (SSAR Reference 2, see Subsection 16.2.6). The response to RAI 260.12 will be revised accordingly. The response to RAI 100.11 provides additional information on defense-in-depth systems, systems found to be significant in the AP600 implementation of regulatory treatment of nonsafety systems (RTNSS), and application of Quality Group D guidelines.

SSAR Revision: NONE

Question 410.110

Provide additional information regarding the capability for detection, control, and isolation of radioactive leakage into and out of the SWS, and prevention of accidental release to the environment. The discussion should include both normal operation and post-accident operation.

Response:

The service water system interfaces with the potentially radioactive component cooling water system at the component cooling water system heat exchangers. As described in SSAR Subsection 9.2.2, the component cooling water system is a closed loop cooling system that provides a barrier to the release of radioactivity between the plant components being cooled that handle radioactive fluid and the environment. Component cooling water system fluid is normally not radioactive but may become contaminated if leakage occurs from some of its interfacing systems. As described in SSAR Subsection 9.2.2.4.5.2, the component cooling water system includes features to detect, control and isolate radioactive inleakage, including radiation monitoring. The component cooling water system also provides for level indication and alarm at the component cooling water system surge tank. As described in Subsection 9.2.2.4.5.3, excessive leakage from the component cooling water system causes the water level in the surge tank to drop and a low level alarm to actuate. This is one method for detecting possible leakage of component cooling water system fluid into the service water system.

Provisions in the service water system design for detecting inleakage of any radioactive fluid include a radiation monitor that monitors the service water system cooling tower blowdown effluent, and the provision for taking local service water system fluid samples both upstream and downstream of the component cooling water heat exchangers. A local sample can be taken of the tower blowdown effluent as well. Tower blowdown flow can be isolated by remote manual control. The tower blowdown valve fails closed upon loss of electrical power or instrument air. The service water system is a cooling water system that normally interfaces with the environment only through such means as the cooling tower, discharge of blowdown and backwashing of strainers. The service water system is not required to operate for post accident conditions. The provisions discussed above for detection, control and isolation of radioactive leakage during normal operation also apply to post-accident operation.

SSAR Revision: NONE

Question 410.112

Provide additional information in Section 9.2.1.5 of the SSAR on the instrumentation that will indicate the valve positions of the SWS.

Response:

Power-actuated valves in the service water system are provided with valve position indication instrumentation. Power-actuated valves that operate in an on/off manner include limit switches that provide end of travel indication. Power-actuated valves that can be operated in a modulating fashion are provided with variable feedback type devices that indicate actual valve position. SSAR Subsection 9.2.1.5 states that heat exchanger inlet block valves are provided with position indication. This statement will be revised to reflect that service water system power-actuated valves are provided with valve position indication instrumentation.

SSAR Revision:

Replace the next to last paragraph in SSAR Subsection 9.2.1.5 with the following:

~~Heat exchanger inlet block valve position indication is provided.~~Power actuated valves in the SWS are provided with valve position indication instrumentation.

Question 410.128

Section 9.2.8.4 of the SSAR indicates that the performance of system components is demonstrated by continuous operation. Section 9.2.8.5 of the SSAR describes all of the instrumentation for the turbine building closed cooling water system. The instrumentation described does not include any temperature or valve position indicators. The staff finds that there are temperature indicators shown in the P&ID. If this is the case, revise the instrumentation description in Section 9.2.8.4 of the SSAR. If not, explain how the system heat removal function can be demonstrated by continuous operation without any temperature indicators. In addition, explain how leakage in the system can be detected and isolated.

Response:

Temperature indication is provided at locations upstream and downstream of the turbine building closed cooling water system heat exchangers, and flow indication is provided for individual cooled components as well as for total system flow. Level indication and alarms are provided for the surge tank as described in Subsection 9.2.8.5 and provide the primary means of detecting leakage into or out of the system. The means of isolating a leak depends on the nature and location of the leak, but in general is accomplished by closing the appropriate component isolation valves. SSAR Figure 9.2.8-1 will be provided in revision 2 of the SSAR.

SSAR Revision:

Revise Subsection 9.2.8.5 as follows:

9.2.8.5 Instrument Applications

~~The pump control and the system flow indicators are located in the main control room.~~

~~Local indication of closed cooling water surge tank level is provided, as well as surge tank low- and high-level alarm in the main control room. Each pump discharge contains a pressure gauge.~~

~~Pressure indicator connections are provided where required for testing and balancing the system. Flow indicator taps are located at strategic points in the system for initial balancing of the flows and for verifying flows during plant operation.~~

Parameters important to system operation are monitored in the main control room. Flow indication is provided for individual cooled components as well as for the total system flow. A total system flow signal also automatically modulates open the system bypass valve, when required, to maintain pump minimum flow.

Temperature indication is provided at locations upstream and downstream of the turbine building closed cooling water system heat exchangers. High temperature of the cooling water supply alarms in the main control room. Temperature test points are provided at locations to facilitate thermal performance testing.

Pressure indication is provided upstream of the pumps and at each pump discharge. Also, low pressure at the discharge of an operating pump automatically starts the standby pump. Pressure instrumentation at the surge tank alarms in the main control room on high pressure in the tank.

Level instrumentation on the surge tank provides level indication and both low- and high-level alarm in the main control room. Also, at low tank level, a valve in the makeup water line automatically actuates to provide makeup flow from the demineralized water transfer and storage system.

Question 410.133

Table 9.2.8-1 of the SSAR lists all of the components cooled by the turbine building closed cooling water system. Figure 9.2.8.1 of the SSAR is the P&ID for the system. The staff cannot find the isophase bus cooling units and miscellaneous pump motors in Figure 9.2.8.1, although they are listed in the table. Clarify the discrepancy and explain the function of these isophase bus cooling units and miscellaneous pumps.

Response:

Cooling for the isolated phase (isophase) bus cooling unit and miscellaneous pump motors are no longer included in the design of the turbine building closed cooling water system. The isophase bus is self-cooled and no cooling water is required. There are no "miscellaneous pump motors" that are cooled by TCS. SSAR Figure 9.2.8-1 will be provided in revision 2 of the SSAR.

SSAR Revision:

Revise SSAR Table 9.2.8-1 as follows.

Table 9.2.8-1

Turbine Building Closed Cooling Water System
Normal Power Generation Operation
(Nominal Values)

Component	Approximate Total Flow (gpm)	Approximate Total Duty (Btu/hr x 10^6)
~~Main turbine lube oil coolers~~	~~3488~~	~~10.17~~
~~Steam generator feed pump motors~~	~~40~~	~~0.40~~
~~Air side seal oil cooler~~	~~260~~	~~1.46~~
~~Hydrogen side seal oil cooler~~	~~100~~	~~0.16~~
~~Exciter air coolers~~	~~280~~	~~0.89~~
~~Vacuum pump seal water cooler~~	~~757~~	~~1.14~~
~~Isophase bus cooling units~~	~~87~~	~~0.87~~
~~Generator hydrogen coolers~~	~~6500~~	~~33.86~~
~~EH control coolers~~	~~20~~	~~0.08~~
~~Condensate pump motor cooler~~	~~50~~	~~0.15~~
~~Turbine plant sampling system~~	~~437~~	~~1.56~~
~~Miscellaneous pump motors~~	~~100~~	~~1.00~~
~~Total~~	~~12,119~~	~~51.74~~
Main turbine lube oil coolers	3000	10.17
Main feedwater pump seal water coolers	40	0.40
Air side seal oil cooler	260	1.46
Hydrogen side seal oil cooler	100	0.16
Exciter air coolers	300	0.89
Generator hydrogen coolers	6510	33.86
EH control coolers	20	0.08
Condensate pump motor coolers	50	0.15
Secondary sampling system coolers	207	0.77
Total	10,487	47.94

Question 410.136

Section 10.1.2 of the SSAR lists six protective features for the steam and power conversion system. Are any of these protective features safety-related?

Response:

Overpressure protection for the main steam lines as described in Item B of SSAR Subsection 10.1.2 is a safety-related function. The other protective features in Subsection 10.1.2 are for investment protection and personnel safety only.

SSAR Revision:

Revise Subsection 10.1.2, Item B, first paragraph as follows:

B. Overpressure Protection

Spring-loaded safety valves are provided on both main steam lines, in accordance with the ASME Code, Section III. The pressure relief capacity of the safety valves is such that the energy generated at the high-flux reactor trip setting can be dissipated through this system. The design capacity of the main steam safety valves equals or exceeds 105 percent of the NSSS design steamflow at an accumulation pressure not exceeding 110 percent of the main steam system design pressure. **Overpressure protection for the main steam lines is a safety-related function.** The main steam safety valves are described in Subsection 10.3.2.

Question 410.144

Section 1.5.1 of Chapter 13 of the EPRI Utility Requirements Document (Volume III) specifies 22 key requirements (Sections 1.5.1.1 through 1.5.1.22) for design life, operability, reliability, accessibility, maintainability, and inspectability for the turbine-generator design for passive ALWRs. Describe how the turbine-generator design for the AP600 design meets each of these 22 requirements.

Response:

The Utility Requirements Document does not create licensing or regulatory requirements. Below is a discussion of how the AP600 turbine generator design meets the 22 requirements with respect to potential impacts on safety-related components and control systems.

1. Design Life

   The turbine-generator is designed for 60 years of service. Some major components, such as the rotors, may require changeout during the plant lifetime. The purpose of this requirement is to protect the owner's investment and to improve turbine-generator availability. Increasing the turbine generator service period to 60 years has no impact on safety-related components.

2. Availability

   The turbine-generator is designed for improved reliability in order to support the overall plant reliability goal. Design features to support this reliability goal include the 22 requirements discussed in this RAI response. By reducing the number of unit trips, the likelihood of challenging safety-related systems is reduced.

3. Design Margin for Abnormal Operating Conditions

   The turbine generator and its auxiliary systems are provided with design margins that will allow it to continue to operate at full or reduced power levels during anticipated secondary side off-design operating conditions. Anticipated off-design secondary side conditions include:

   - Operation at off-normal frequency deviations
   - Turbine operation at 70% power with one string of HP or LP feedwater heaters out of service
   - Operation with the MSR out of service
   - Operation of the generator rotor 2 times a year at a field voltage of 125% of rated load field voltage for one minute.

   Design margins increase unit availability and reduce the likelihood of challenging safety-related systems.

4. Proven Technology

Components of the AP600 turbine-generator are conventional and typical of those that have been used in other power plants. This requirement has no impact on safety-related components and control systems.

5. Technical Analysis

Criteria for stress analysis, fatigue, and brittle fracture analysis of the turbine-generator components are established. These technical analyses will be prepared for the AP600 and provide design basis information to the operator.

6. Reliability and Accessibility of Threaded Fasteners

Design requirements associated with the use of threaded fasteners and positive locking devices for threaded fasteners are established to reduce the potential for loss generation due to threaded fastener failure. Reducing the number of turbine trips reduces the likelihood of challenging safety-related systems. Fasteners should be readily accessible for removal/installation using manual or hydraulic wrenches to prevent delays during turbine generator maintenance. This requirement has no impact on safety-related components and control systems.

7. Maintenance and Inspection Enhancements

Recommendations are provided for enhancing maintenance and inspection of turbine generator equipment to increase worker productivity and minimize the impact of maintenance, inspection, and test activities on plant availability. Equipment access, equipment layout, special tools, and maintenance space requirements are established. This requirement has no impact on safety-related components and control systems.

8. Turbine Flow Margin

The turbine is designed with up to a 5% flow margin above the flow required to meet the maximum guaranteed output in order to provide flexibility in matching steam conditions from the steam generator. This is a standard design requirement for turbines. This requirement has no impact on safety-related components and control systems.

9. Turbine Loading Rates

The turbine is designed for a load step increase or decrease of 10% in 60 seconds when the plant power level is in the range of 50 to 100%. SSAR Subsection 7.7.2 provides information on specified reference transients that the plant control system is capable of maneuvering through without violating plant protection and component limitations. These reference transients include a specified 10 percent step load increase or decrease. Subsection 7.7.1.1 also describes the ramp rate capability without reactor trip or steam dump actuation. This requirement has no impact on safety-related components and control systems.

10. Load Following Operation

The AP600 is designed to allow daily load changes between 100 and 30 percent of rated power and is capable of transitioning between baseload and load follow operation. The turbine generator control system has the same loading and load following characteristics as the NSSS. This information is found in SSAR Subsection 10.2.7.

11. Turbine Control System Reliability

Provisions are included to increase the reliability of the control system including turbine overspeed protection, quick closing steam supply valves, redundant control system features, two-out-of-three voting logic on turbine governor speed signals, redundant instrumentation, and redundant methods of overspeed protection.

The AP600 provides quick closing main stop, control valves, intercept valves, and reheat stop valves to prevent turbine overspeed events. Redundant power sources are supplied to the turbine electrohydraulic control (EHC) system and redundant instrumentation is provided at critical components to increase control system reliability.

Two separate methods of turbine overspeed protection are provided in the AP600. The first is the normal overspeed control which is an integral portion of the speed control unit. The second is the emergency trip system which uses three separate speed signals and two-out-of-three logic for overspeed tripping.

Additional information on these AP600 systems is available in SSAR Section 10.2. Incorporation of these features into the AP600 design potentially reduces the number of turbine trips, thereby reducing the likelihood of challenging safety-related systems.

12. Stress Corrosion Cracking Resistant Materials

Turbine rotor and blade materials are selected and fabrication techniques are employed which have proven resistance to stress corrosion cracking (SCC). See SSAR Subsection 10.2.3.

13. Field Repairable Diaphragm Blade Materials

Provisions are provided for field repair of diaphragm blades in order to reduce the outage time required for diaphragm repair. This has no impact on safety-related components and control systems.

14. Torsional Shaft Vibrations

Specific provisions to preclude damage due to sub-and super-synchronous torsional shaft vibrations are provided to reduce the probability of LP turbine blade failures.

15. One-piece Rotor Designs

The AP600 turbine rotors are fully integral.

16. Moisture Erosion Protection

    Provisions are included for protection against moisture erosion of turbine steam path components in order to reduce the need for blade replacements which result in lengthened turbine outages, and therefore, reduced plant availability. This requirement has no impact on safety-related components and control systems.

17. Rotor Interchangeability

    LP and Hp rotor interchangeablity reduces outage time and spare parts inventory. This has no impact on safety-related components and control systems.

18. Turbine Valve Assembly

    Proven designs are incorporated into the AP600 turbine generator valves and design criteria have been established to control vibration induced failures.

19. Lube Oil System and Seal Oil System Cleanliness

    Full flow filters and corrosion resistant supply piping downstream of the full flow filters in the lube and seal oil systems is provided to reduce lube oil contaminants which are a major cause of turbine-generator bearing degradation and failures. This requirement has no impact on safety-related components and control systems.

20. On-line Diagnostic Monitoring

    On-line continuous monitoring and trending instrumentation for the turbine generator is provided to detect and locate problems before component failures occur. The AP600 turbine-generator supervisory instrumentation is listed in SSAR Subsection 10.2.5.

21. Generator Retaining Ring Material

    18Mn-18Cr material is used for the generator retaining rings to provide improved resistance to stress corrosion cracking. This requirement has no impact on safety-related components and control systems.

22. Stator End Winding Supports

    Design improvements have been implemented to address end winding vibration and loosening problems. These problems have caused insulation breakdown, ground failures, and phase lead breakage resulting in reduced plant availability. This requirement has no impact on safety-related components and control systems.

SSAR Revision: NONE

Question 410.150

Section 10.4.10.2.2 of the SSAR states that Section 3.2 provides the codes and standards for the auxiliary steam system. Section 3.2.4 states that Table 3.2-3 lists mechanical and fluid system component and its associated equipment class and seismic category as well as other related information. However, the staff cannot find the information on the code and standards of the auxiliary steam system in Table 3.2-3 or Section 3.2 of the SSAR. Provide the above information.

Response:

The auxiliary steam system is safety classification Class E. The auxiliary steam system is listed in Table 3.2-3 sheet 107 of 107 as one of the systems that contain no components that are Class A, B, C, or D. Class E components are designed to industrial standards as discussed in note 12 of Table 3.2-1.

SSAR Revision:

Revise the first sentence in SSAR Subsection 10.4.10.2.2 as follows.

10.4.10.2.2 Component Description

~~Codes and standards applicable to the auxiliary steam system are listed in Section 3.2.~~ Auxiliary steam system component classification is as described in Section 3.2.

Question 410.151

Section 10.4.10.4 of the SSAR states that the auxiliary steam system is tested prior to initial plant operation. How will the test be performed? Where is the test program? The staff cannot locate it in Chapter 14 or any other SSAR Chapter.

Response:

Testing procedures for the auxiliary steam system are not located in Chapter 14 of the SSAR. The scope of Chapter 14 is presented in Subsection 14.2.1. Chapter 14 contains only preoperational and startup tests for systems that: are relied on for safe shutdown and cooldown of the reactor; are relied on for establishing conformance with safety limits; are classified as engineered safety features actuation systems; are assumed to function during an accident; or are used to limit the release of radioactive material as described in Subsection 14.2.12 of Regulatory Guide 1.70. The auxiliary steam system performs none of these functions, therefore testing procedures are not included in Chapter 14. Testing procedures for the auxiliary steam system are included in the system specification and vendor equipment instruction manuals which are not part of the AP600 design certification review.

SSAR Revision: NONE

Question 410.153

Section 9.3.1 of the SSAR states that (a) the compressed and instrument air systems are free of all corrosive contaminants and hazardous gases, flammable or toxic, which may be drawn into the airstream and (b) the breathing air subsystem is free of radioactive contamination (See Information Notice 85-06). How will this be accomplished? Are the compressor intakes located in an area free of corrosive contaminants and hazardous gasses? Will regular periodic checks be made to assure high quality air?

Response:

a. The air compressors associated with the compressed and instrument air system (CAS) are located in the turbine building on elevation 135'. Carbon dioxide and nitrogen are stored in the turbine building at elevation 100'. These gasses are not corrosive or flammable. Due to the free volume of the turbine building and the distance between the gas storage site and the inlet to the breathing air compressor, failures of these bulk gas systems are unlikely to affect the breathing air subsystem.

   The hydrogen and oxygen bulk gas storage facilities are located outdoors approximately 650' from the air compressor intakes. Postulated failures of these systems would not affect the compressed and instrument air system (CAS).

b. No potential radioactive contamination sources are located near the breathing air system compressor intake. Air supplied to the breathing air subsystem compressor is taken from the turbine building. The AP600 turbine building has no significant sources of airborne radioactivity (refer to SSAR Subsection 12.3.3.4).

   The compressor intakes (instrument, service, and breathing air) are located in an area where corrosive contaminants, radiation hazards, and hazardous gasses are not normally found. The compressed air system (CAS) consists of three separate subsystems. The independence of the breathing air subsystem prevents the type of accidents resulting from cross connections as indicated in Information Notice 85-06. Additionally, provisions are made for sampling lines to determine air quality.

SSAR Revision: NONE

Question 410.158

Why does Figure 9.3.1-1 (Sheet 2 of 2) of the SSAR show (a) the instrument air dryers (1A and 1B) and (b) the breathing air emergency backup bottles with a dotted line? Does this mean they are not part of the system?

Response:

The dotted lines on the referenced drawing are a way of indicating a commodity or a packaged device that is procured directly from a vendor. The dotted line serves to delineate the boundary of the vendor equipment package. The instrument air dryers (1A and 1B) as well as the emergency backup air bottles are a part of the CAS.

SSAR Revision: NONE

Question 410.159

Revise Section 9.3.1.2.2 of the SSAR to provide the following information:

a. Because Section 9.3.1.2.2 addresses the quality of air in the compressed and instrument air system, the staff believes that it is more appropriate to reference ANSI/ISA-S7.3-1975 (R1981) on air quality instead of Regulatory Guide (RG) 1.68.3 (testing) in the last sentence in the second paragraph. Therefore, modify this sentence to state that "The test performance criteria shall be -28°F dewpoint at line pressure in accordance with ANSI/ISA-S7.3-1975 (R1981)."

b. Because Section 9.3.1.2.2 addresses the quality of air in the compressed and instrument air system, the staff believes that it is more appropriate to reference ANSI/ISA-S7.3-1975 (R1981) on air quality instead of Section 9.3.1 of the standard review plan in the last sentence in the third paragraph. Therefore, modify this sentence to state that "The afterfilters are a disposable cartridge filter capable of removing 98 percent of one micron and larger particulates and 100 percent of three micron and larger particulates in accordance with ANSI/ISA-S7.3-1975 (R1981)."

c. Provide a commitment to NUREG-1275 regarding air quality for the compressed and instrument air system by stating that the air quality (-28°F dewpoint and particulates ≤ 3 microns) meets the manufacturer's air supply requirements for all pneumatic equipment that is either safety-related or relied upon to perform a safety function.

Response:

We concur with the recommendations made on the substitution of ANSI/ISA-S7.3 - 1975 (R1981) for Regulatory Guide 1.68.3 in sections a and b.

a. The SSAR will be modified to state that " The test performance criteria shall be -28 °F dewpoint at line pressure in accordance with ANSI/ISA-S7.3-1975 (R1981)."in Subsection 9.3.1.2.2.

b. Subsection 9.3.1.2.2 will be changed to indicate that "The afterfilters are a disposable cartridge filter capable of removing 98 percent of 1 micron and larger particulates and 100 percent of 3 micron and larger particulates in accordance with ANSI/ISA-S7.3-1975 (R1981)."

c. This is not required in the AP600 SSAR because only the fourth stage automatic depressurization system (ADS) valve uses CAS for normal operation. A separate safety-related accumulator is provided to drive the valve to the fail-safe position. There are no devices in the AP600 plant that rely upon instrument air to perform a safety-related function.

SSAR Revision: See marked sections of the SSAR as attached to and referenced in the response to RAI 410.152.

The following are statements addressing this RAI and SSAR revisions:

a. Revise Subsection 9.3.1.2.2 as follows.

9.3.1.2.2 Component Description

Instrument Air Subsystem

The ~~three~~ two oil-free rotary compressors associated with the instrument air subsystem are each rated at 800 scfm with a discharge pressure of 125 psig. The compressors are driven by 200-hp motors.

Each of the two air dryers has a rated capacity of ~~1600~~ 800 scfm at -~~6~~40°F dewpoint at 120 psig. The test performance criteria shall be -28°F dewpoint at line pressure in accordance with ~~Regulatory Guide 1.68.3 recommendations~~ANSI/ISA-S7.3-1975 (R1981).

b. Revise Subsection 9.3.1.2.2 as follows.

Each of the two prefilters and afterfilters are sized for ~~1600~~ 800 scfm at 120 psig. The prefilters are disposable coalescing cartridge filters capable of removing 99.9 percent of all moisture larger than 0.3 micron and particulates greater than 15.0 microns. The afterfilters are a disposable cartridge filter capable of removing 98 percent of one micron and larger particulates and 100 percent of three micron and larger particulates in accordance with ~~NUREG-0800, Section 9.3.1~~ANSI/ISA-S7.3-1975 (R1981). .

c. No SSAR revision necessary.

Question 410.161

Because Section 9.3.1.4 of the SSAR addresses the testing of the compressed and instrument air system, the staff believes that it is more appropriate to reference RG 1.68.3 instead of ANSI/ISA-S7.3-1975 (R1981) on air quality in the third paragraph of the section. Therefore, modify the paragraph to state that "During the initial plant testing prior to reactor startup . . . upon a complete and sudden loss, a gradual loss, and an increase of compressed air pressure as described in RG 1.68.3." This information should also be provided in Section 14 of the SSAR.

Response:

The third paragraph of SSAR Subsection 9.3.1.4 will be deleted because testing of air operated safety-related valves is outside the scope of the CAS testing. There are no devices in the AP600 plant that rely upon instrument air to perform safety functions.

SSAR Revision: See marked section of the SSAR as attached and referenced in RAI 410.152.

The SSAR will be revised as follows:

The third paragraph of Section 9.3.1.4 will be deleted.

9.3.1.4 Tests and Inspections

The compressors, aftercoolers, receivers, prefilters, dryers, purification systems, afterfilters, and the control panels are inspected or tested prior to installation. The complete, installed compressed air system is inspected, tested, and then operated to verify it meets its performance requirements, including operational sequences and alarm functions.

Air compressors and associated components on standby are checked and operated periodically. Air filters are inspected for cleanliness.~~, and the desiccant~~ Dessicant in the air dryers is changed when it no longer performs according to the manufacturer's specifications.

~~During the initial plant testing prior to reactor startup, safety systems utilizing compressed air are tested to verify fail-safe operation of air operated valves upon loss of compressed air or reduction of air pressure as described in Reference 1. Section 1.9 summarizes conformance with Regulatory Guide 1.68.~~

Sample points are provided downstream of the air dryers in the instrument air subsystem and downstream of the purifiers in the breathing air subsystem. Each sample line includes a manual one inch globe valve with a reduced port.

The breathing air system is inspected, tested, and then operated to verify its performance requirements.

Question 410.163

Revise Figure 9.2.9-1 of the SSAR to agree with the system description in Section 9.2.9. For example, the turbine building drain tanks and pumps, referred to in the system description, are not given the same title in the figure.

Response:

Figure 9.2.9-1 has been revised and will be incorporated into the SSAR Revision 2.

SSAR Revision:

Revised Figure 9.2.9-1 will be provided in the SSAR Revision 2.

Question 410.176

Revise Figure 9.5.4-1 of the SSAR to show the whole system, including the day tank and the piping from the day tank to the diesel generator.

Response:

A revised Figure 9.5.4-1 will be provided in revision 2 of the SSAR.

SSAR Revision:

Figure 9.5.4-1 of the SSAR will be revised.

Question 410.185

Regarding waste water drainage in the plant, Section 9.2.9 of the SSAR states, in part, that level controls are provided for the building sumps, surge tank, and waste water retention basin to prevent overflow. The staff is concerned that the drainage system to the sump or surge tank may fail because of events such as an earthquake. Provide information on flood levels and the methods for draining out the water after a limiting pipe break, assuming a period of water leakage while the operator isolates the problem area. Also, identify any safety-related equipment in other plant areas that will be affected by such flooding due to pipe rupture.

Response:

Protection for internal flooding in areas of the plant containing safety-related systems or equipment is described in SSAR Subsection 3.4.1.2.2. The flooding evaluation is based on determining the postulated fluid system failures which could result in the most adverse internal flooding conditions. Flooding sources include:

- High energy piping (breaks and cracks)
- Moderate energy piping (through-wall cracks)
- Storage tank ruptures
- Actuation of fire suppression systems
- Flow from upper elevations and adjacent areas

The internal flooding analysis shows that safety-related systems, structures, and components are not prevented from performing their required safe shutdown function due to the effects of the most adverse fluid system failures.

The limiting pipe break in the turbine building is a postulated failure of the circulating water system (CWS) piping or expansion joint. No credit for turbine building drain lines, sump pumps, or level controls is taken for pipe breaks in the turbine building. As discussed in SSAR Subsection 10.4.5.2.3, this limiting break will not result in detrimental effects to safety-related equipment.

SSAR Revision: NONE

Question 410.187

Section 10.4.5.2.2 of the SSAR states, in part, that the CWS is designed to withstand the maximum operating discharge pressure of the CW pumps. However, flooding may occur in the turbine building if the CWS piping fails. Provide an analysis for the effects of a postulated failure of the CWS piping or expansion joints, and verify that any safety-related structures, systems, and components in the turbine building will be protected from the resulting flood water level.

Response:

Section 10.4.5.2.3 of the AP600 SSAR (Revision 1) provides an analysis for postulated failure of the CWS piping or expansion joints. The descriptive paragraph is as follows.

"The effects of flooding due to a circulating water system failure, such as rupture of an expansion joint will not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the turbine building and the base slab of the turbine building is located at grade elevation. Water from a system rupture will run out of the building through the doors before the level could rise high enough to cause damage. Site grading will carry the water away from safety-related buildings."

SSAR Revision: NONE

Question 410.190

Section 10.4.7.1.1 of the SSAR indicates that double valve startup feedwater isolation is provided by the startup feedwater control valve (SFCV) and the startup feedwater isolation valve (SFIV). The SFIV serves as a containment isolation valve and closes on a containment isolation signal or backflow in the line. Describe whether the SFCV will close on a containment isolation signal, and is subject to leak testing in accordance with Appendix J of 10 CFR Part 50. If not, what are the closure actuation and leakage test requirements for the SFCV?

Response:

SSAR Subsections 7.3.1.1.3.4 and 7.3.1.1.4.4 discuss the "Engineered Safety Features" isolation logic for the startup feedwater system. The isolation logic is illustrated on Figure 7.2-1 sheets 2, 10, and 11.

The containment isolation features for the feedwater line include the secondary side of the steam generator, the main feedwater, main steam and blowdown lines inside containment, and the feedwater and startup isolation valves outside containment. The startup feedwater control valves are not a part of the containment boundary and thus not subject to leakage testing under 10 CFR 50 Appendix J. The valves are designed and procured to comply with industry standards relative to seat leakage. As specified in the SSAR, the valve design is such that positive isolation is provided independent of the valve trim surfaces and therefore will provide reliable isolation.

SSAR Revision: NONE

Question 410.208

Section 7.4 of the SSAR identifies safety-related equipment located outside containment. This should be referenced in Section 3.5. There is no equipment important-to-safety whose failure could adversely affect safety-related equipment (see Q410.27). Clarify why this is the case. Further, Section 3.7.3.13 of the SSAR discusses methods of protecting safety-related SSCs from adverse interaction with non-safety-related SSCs. Section 3.7.3.13.1 says that physical separation is provided between safety-related and non-safety-related SSCs to the maximum extent possible. Clarify how safety-related SSCs are protected if the physical separation cannot be achieved. Any nonseismic component identified as a source is evaluated according to guidelines in Sections 3.7.3.13.1 through 3.7.3.13.3 and appropriate protection is provided. Section 3.5.1.1 of the SSAR should reference Section 3.7.3.13 for clarity.

Response:

A reference to Section 7.4 has been added in Section 3.5. This defines the safety-related equipment required for safe shutdown both inside and outside containment. The equipment required for safe shutdown is classified as safety-related. Thus failure to function of nonsafety-related equipment does not jeopardize safe shutdown. The failures of nonsafety-related equipment that could affect safe shutdown are structural or pressure boundary failures resulting in effects such as missiles, flooding, pipe whip and jets. The failures resulting in missiles are discussed in SSAR Section 3.5. Those due to flooding and pipe rupture are discussed in Sections 3.4 and 3.6. Gravitational missiles resulting from seismic failures are discussed in SSAR Subsection 3.7.3.13. This subsection is already referenced in Section 3.5. As described therein, the nonsafety-related component is upgraded to seismic Category II when other methods of protection cannot be achieved.

SSAR Revision:

Revise first paragraph of Section 3.5 as follows:

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures systems and components important to safety be protected from the effects of missiles. The AP600 criteria for protection from postulated missiles provide the capability to safely shut down the reactor and maintain it in a safe shutdown condition. The AP600 criteria also protect the integrity of the reactor coolant system pressure boundary and maintain offsite radiological dose/concentration levels within the limits defined in 10 CFR 100. Systems required for safe shutdown are identified in Section 7.4.

Question 410.213

Westinghouse states that the AP600 uses only safety-related systems and equipment to establish and maintain safe-shutdown conditions, and that there is no equipment important-to-safety (as defined in Q410.27) outside the containment that requires missile protection. Justify this statement. Is this statement also true for the defense-in-depth systems and equipment that are identified in Table 3.2-3 of the SSAR, and for the systems and equipment identified as important by the analysis to determine the need for the regulatory treatment of non-safety-related systems? If so, justify. If not, describe the systems and the protection provided.

Response:

The AP600 has no equipment important to safety as defined in RAI 410.27, that is, nonsafety-related equipment whose failure could adversely affect the ability of safety-related equipment to perform its safety function. Equipment required for safe shutdown is classified as safety-related. Thus failure of nonsafety-related equipment to function would not jeopardize safe shutdown. The potential failures resulting in missiles are discussed in SSAR Section 3.5. Gravitational missiles resulting from seismic failures are discussed in SSAR Subsection 3.7.3.13.

Defense-in-depth systems and equipment are not safety-related systems and equipment and are not required for safe shutdown.

The AP600 evaluation of the regulatory treatment of nonsafety-related systems (RTNSS) described in Reference 410.213-1 identifies a small number of nonsafety-related systems (as defined by RTNSS) that should be available during reduced reactor coolant system inventory conditions (midloop). The recommended regulatory oversight outlined in Reference 410.213-1 for these systems is that they be operable prior to entering reduced reactor coolant system inventory conditions. Missile protection is not required for these systems.

Reference:

410.213-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process Summary Report, September, 1993.

SSAR Revision: NONE

Question 410.219

The March 18, 1993, response to Q410.63 states that no safety-related equipment or equipment important-to-safety requires protection from internally-generated missiles because there are no credible missile sources. The staff believes that this is an incorrect characterization. There is safety-related equipment that requires missile protection. The means of providing protection is by ensuring that there are no credible missile sources. Section 3.5 of the SSAR should clearly state what safe shutdown structures, systems, and components must be protected from missiles [internally-generated (outside containment), internally-generated (inside containment), turbine generator, those generated by natural phenomena, and externally-generated]. If the same SSCs must be protected for all these missile hazards, it should be so stated in Section 3.5 of the SSAR. If different safety-related SSCs must be protected for different missile hazards, then the SSCs should be identified in the appropriate missile section of the SSAR.

More specifically, the staff needs to know what safe-shutdown equipment is located in the containment, what missile sources exist in the containment that could adversely affect this equipment, and how this equipment is protected from these missiles. Also, there is no discussion regarding separation of redundant divisions of safety-related systems. Is there physical separation between redundant divisions of safety-related systems inside the containment? If so, what is the nature of the separation (physical distance, enclosure in separate compartments, or the use of barriers)?

Response:

SSAR Subsection 3.5.2 identifies those structures, systems, and components that are required for safe shutdown of the plant and require protection from externally generated missiles and from internally generated missiles as well. The response to RAI 410.67 discusses the protection of safe shutdown equipment inside the containment from potential missile sources. As stated in the RAI response, no missile source exists in the containment that could adversely affect the safe-shutdown equipment inside containment. SSAR Appendix 9A 3.1.1 of the fire protection analysis provides a discussion on physical separation and redundant divisions of safe-shutdown systems inside containment. Responses to RAIs 410.64, 410.68, and 410.69 discuss how safe shutdown equipment inside containment is protected from the effects of externally generated missiles.

SSAR Revision:

Revise SSAR Section 3.5, page 3.5-2 after the sixth bullet as follows:

- Equipment required for safe shutdown is located in plant areas separate from potential missile sources wherever practical.

The AP600 passive design minimizes the number of safety-related structures, systems, and components required for safe shutdown. Systems required for safe shutdown are identified in Chapter 7. Safety class structures, systems and components, their location, seismic category, and quality group classifications are given in Section 3.2. General arrangement drawings showing locations of the structures, systems and components are given in Section 1.2. The areas required for safe shutdown, and the major systems and components housed therein that are required to be protected from internally and externally generated missiles for safe shutdown, are summarized below:

- The containment vessel, including the reactor coolant loop, and passive core cooling system inside containment
- The shield building, including the passive containment cooling system
- Containment penetration areas, including containment isolation valves and Class IE cables
- The control complex including the main control room, reactor protection system, batteries and dc switchgear
- The spent fuel pit.

Revise SSAR Section 3.5.2 as follows:

3.5.2 ~~Structures, Systems and Components to be Protected from~~ Externally Generated Missiles

~~The AP600 passive design minimizes the number of safety related structures, systems, and components required for safe shutdown. Systems required for safe shutdown are identified in Chapter 7. Safety class structures, systems and components, their location, seismic category, and quality group classifications are given in Section 3.2. General arrangement drawings showing locations of the structures, systems and components are given in Section 1.2. The areas required for safe shutdown, and the major systems and components housed therein that are required for safe shutdown, are summarized below:~~

- ~~The containment vessel, including the reactor coolant loop, and passive core cooling system inside containment~~
- ~~The shield building, including the passive containment cooling system~~
- ~~Containment penetration areas, including containment isolation valves and Class IE cables~~
- ~~The control complex including the main control room, reactor protection system, batteries and dc switchgear~~
- ~~The spent fuel pit.~~

Protection from external missiles is provided by the external walls and roof of the nuclear island structures. Openings through these walls are evaluated on a case-by-case basis to provide confidence that a missile passing through the opening would not prevent safe shutdown and would not result in an offsite release exceeding the limits defined in 10 CFR 100.

Question 410.243

Address the following concerns regarding the nuclear island non-radioactive ventilation system (VBS). The staff expects that under all postulated radiation conditions, the VBS will be able to continue to operate and protect the control room operators as long as there is power available. The following questions are based on this premise.

a. Charcoal adsorber efficiency for organic iodine removal should be 95 percent, not 90 percent, to be able to take credit for the VBS to function as a first line of defense under the "Defense-in-Depth" concept. In order to specify a 95-percent iodine removal efficiency, specify that an iodine penetration of ≤ 1 percent for a 4-inch depth of activated carbon cell when laboratory testing is performed at 30 °C and ≤ 70-percent relative humidity, or an iodine penetration of ≤ 0.7143 percent for a 4-inch depth of activated carbon cell when laboratory testing is performed at 30°C and ≤ 95% relative humidity, in accordance with ASTM D3803-89 standards. Revise the SSAR accordingly.

b. The November 16, 1993, response to RAI 100.10 indicates that the VBS is credited initially following a "HIGH" (not "HIGH HIGH") radiation signal in conjunction with the VES to meet GDC 19 dose limits. Therefore, the VBS filtration subsystem should be safety-related, and comply with RG 1.52 positions and Table 4-1 of ANSI/ASME N509-1989 for instrumentation and controls.

   If the VBS is not credited in conjunction with the VES to meet GDC 19 dose limits following a "HIGH" and/or "HIGH HIGH" radiation signal, then the VBS filtration subsystem is non-safety-related, and needs to conform only to the guidance of RG 1.140. However, detailed categorical conformance with RG 1.140 positions and Table 4-2 of ANSI/ASME N509-1989 should be provided for instrumentation and controls.

   Revise Section 9.4.1 of the SSAR and WCAP-13054 regarding conformance to Sections 6.4 and 9.4.1 of the SRP to show conformance with RG 1.52 or RG 1.140 (as applicable based upon the response to the above comments), ANSI/ASME standards N509-1989, N510-1989 and ASTM D3803-1989 and ASME code AG-1-1991.

c. Provide the rationale for installing the supply air filter fan upstream of the air filtration unit in contrast to the conventional design at current operating plants. The rationale should address continued fan operability during the accident conditions without clogging due to foreign debris, including radioactive debris, during accident conditions. Note that the conventional design of filtration units provides filtered inlet flow to the supply fan(s). Revise the SSAR accordingly.

d. The following concerns pertain to the VBS Flow Diagram Figure, VBS MS 006:

   1. For the "Normal Operation" mode:

   A. "Data Point 108" - The supply flow for the clean and reference material store area is 1340 SCFM and the return flow is 930 SCFM. Provide the rationale for this difference, and/or revise the figure accordingly.

   B. The supply flows to the MCR kitchen and toilet room are 300 and 100 SCFM, respectively. The discharge flows are 380 and 120 SCFM. Provide the rationale for these values, and/or revise the figure accordingly.

C. The supply flows to the technical support center men's room, women's room and kitchen are 350, 350, and 400 SCFM, respectively. The discharge flows are 400, 400, and 450 SCFM, respectively. Provide the rationale for these values, and/or revise the figure accordingly.

2. For the "Smoke Removal Mode" for the MCR:

The tagging room supply flow at data points 51 and 52 is 235 SCFM each, and the return flow at data point 106 is 700 SCFM. Provide the rationale for these values, and/or revise the figure accordingly.

3. For the "Smoke Removal Mode" for the TSC:

The supply flow to the offices at data point 93 is 1480 SCFM, and the return flow at data point 125 is 2080 SCFM. Provide the rationale for these values, and/or revise the figure accordingly.

e. Provide the following information on Sheets 3, 4, and 5 of 6 of Figure 9.4.1, and Table 9.4.1-1 of the SSAR:

1. Provide flow diagrams and corresponding data for the Division A and C, and B and D Class 1E electrical rooms HVAC subsystem and Division A and C, and B and D emergency battery rooms exhaust.

2. Provide the rationale for providing only 25-percent efficiency for Division A and C, and B and D air handling unit (AHU) prefilters.

3. Explain why Division B and D emergency battery room exhaust fans have 5 horse power (HP) motors verses Division A and C exhaust fans have 3 HP motors with identical flows of 2,400 SCFM per fan, as identified in Table 9.4.1-1 of the SSAR.

4. Clarify the system capacity for the Division B and D Class 1E electrical room HVAC subsystem, which is shown as "15,00 SCFM" on page 9.4-42 of Table 9.4.1-1 of the SSAR. Revise the SSAR accordingly.

Response:

a. The nuclear island non-radioactive ventilation system (VBS) supplemental air filtration subsystem operating during abnormal modes is a defense-in-depth function when VBS is operable and an ac power source is available. There is no credit taken for VBS filtration operation in the main control room habitability analysis to meet General Design Criteria 19 limits under accident conditions. The main control room operator habitability requirements under accident conditions are provided by the main control room habitability system (VES) which is designed to satisfy nuclear safety-related system design and seismic Category I requirements and satisfies Regulatory Guide 1.52 positions.

Since the VBS is nonsafety-related and is not required to be operable after a design basis accident, a 95% charcoal adsorber efficiency in conjunction with a 1% penetration should not be applied to design and surveillance requirements for the supplemental air filtration unit charcoal adsorber and a charcoal adsorber efficiency of 90% is adequate for iodine removal based on non-engineered safety features filtration systems designed in accordance with Regulatory Guide 1.140. The main control room habitability/radiological analysis for the VBS supplemental air filtration unit in a defense-in-depth function is described in the response to RAI 450.010. This analysis was performed using the 90% charcoal adsorber efficiency and concluded that General Design Criteria 19 dose limits are not exceeded under accident conditions.

As described in SSAR Subsection 9.4.1, the VBS supplemental air filtration subsystem is designed, constructed and tested in accordance with ASME N509-1989 and ASME N510-1989 to satisfy the guidelines of Regulatory Guide 1.140. ASME N509-1989, Section 5.2.3 requires that charcoal adsorbent media used in non-engineered safety features adsorbers meet the laboratory testing requirements of ASME AG-1-1988, Section FF.

b. Initiation of VBS supplemental air filtration operation during abnormal postulated radiation conditions is a nonsafety-related, defense-in-depth function following receipt of a "High" radiation signal. Upon receipt of a "High-High" radiation signal, which is an indicator that VBS supplemental air filtration mode is not available or not functioning properly, the VBS safety-related main control room pressure boundary isolation dampers are automatically closed and the safety-related VES is automatically initiated. Only the safety-related main control room pressure boundary HVAC isolation dampers in conjunction with VES are credited in meeting General Design Criteria 19 dose limits following a "High-High" radiation signal.

In addition, ASME AG-1 Code provides design, construction, performance, and testing requirements for the nuclear safety-related air and gas treatment system only, and ASME N509-1989 is applicable to both nuclear safety-related and nonsafety-related air and gas treatment systems. Therefore, Regulatory Guide 1.52 and ASME AG-1-1991 design criteria do not apply to VBS and application of Regulatory Guide 1.140 and ASME N509-1989 for normal air filtration system design is appropriate. Conformance to Regulatory Guide 1.140 positions is described in Appendix 1A of the SSAR. Additional information regarding conformance to Regulatory Guide 1.140 is provided in the response to RAI 410.240 and 410.241. Conformance to the instrumentation and controls requirements in Table 4-2 of ASME N509-1989 is described in the response to RAI 410.240.

c. The VBS supplemental air filtration unit supply air fan is located upstream of the air filtration unit in order to meet the requirements of Section 4.7.2, "Habitability Systems" of ASME N509-1989. Subsection 4.7.2.(c) states that "The makeup air fan shall be located upstream of the air-cleaning unit if the air-cleaning unit is in a contaminated space". Subsection 4.7.2.(e) states that "Recirculating system housing should be kept at a positive pressure if located outside the habitable boundary in a contaminated space or interspace". The interspace refers to all other space - contaminated or clean - where the nuclear air treatment system or its parts may be located. VBS supplemental air filtration units are located in the equipment room at elevation 135'-6' of the auxiliary building which could be a contaminated space after a radioactivity release event. Therefore, installing the supply air filter fan upstream of the supplemental air filtration unit and designing the unit housing for positive pressure are consistent with the design philosophy specified by ASME N509-1989. The positive pressure unit housing design enhance constructability, reduces cost, minimizes the unit housing and filter bypass

leakage concern compared to a negative pressure unit housing design; therefore, VBS supplemental air filtration unit operability and maintainability is improved.

d. 1. For the "Normal Operation" mode:

A. "Data Point 108" - The difference between the supply flow (data point 53) and the return flow (data point 108) for the clean and reference material storage area is 100 SCFM of transfer air flow into the toilet room (data point 146) and the kitchen area (data point 147) in order to ensure directional flow in the toilet room and kitchen area. The balance of 100 SCFM transfer air flow is described in item d.1.B.

B. Main control room toilet room has supply flow of 100 SCFM (data point 45) and exhaust flow of 120 SCFM (data point 146) and kitchen area has supply flow of 300 SCFM (data point 46) and exhaust flow of 380 SCFM (data point 147). The balance of 20 SCFM is transferred from the clean and reference material storage area to the toilet room and the balance of 80 SCFM is transferred from the clean and reference material storage area to the kitchen area.

C. Technical support center men's room has supply flow of 350 SCFM (data point 75) and exhaust flow of 400 SCFM (data point 141), women's room has supply flow of 350 SCFM (data point 74) and exhaust flow of 400 SCFM (data point 142), and dining/kitchen area has supply flow of 400 SCFM (data point 65) and exhaust flow of 450 SCFM (data point 144). The technical support center corridor has supply flow of 300 SCFM (data point 69) and return flow of 150 SCFM (data point 111) with the balance of 150 SCFM extra air transferred to the men's room (50 SCFM), the women's room (50 SCFM), and the dining/kitchen area (50 SCFM).

2. For the "Smoke Removal Mode" for the main control room:

The tagging room supply flow (data points 51 and 52) is 235 SCFM each, and the return flow (data point 106) is 700 SCFM. The main control room/technical support center toilet exhaust fan is isolated during the smoke purge mode. The portion of supply air to the toilet and kitchen area is exhausted through the tagging room (data point 106).

3. For the "Smoke Removal Mode" for the technical support center:

The supply flow to the offices (data point 93) is 1,480 SCFM, and the return flow (data point 125) is 2,080 SCFM. The main control room/technical support center toilet exhaust fan is isolated during the smoke purge mode. The portion of supply air to the corridor, toilet and kitchen area is exhausted through the offices (data point 125).

e. 1. Relevant system process data information (flow rates, temperatures, and pressures) are contained in various engineering documents and are available for review at the design agent offices as discussed in response to RAI 410.238.

2. Division "A and C" and "B and D" air handling units (AHU) are designed with 25% efficiency prefilters and 80% efficiency high efficiency filters as indicated in Subsection 9.4.1.2.1, Table 9.4.1-1 (Sheets 4 and

5) and Figure 9.4.1-1 (Sheets 3 and 4) of the SSAR. The prefilters function is to extend the life of high efficiency filters.

3. The Division "A and C" and "B and D" emergency battery room exhaust fans should have 5 horse power (HP) motors. The SSAR will be revised accordingly.

4. The design air flow for the Division "B and D" Class 1E electrical room HVAC subsystem air handling units should be read as "15,000 SCFM". The SSAR will be revised accordingly.

SSAR Revision: SSAR Table 9.4.1-1 shall be revised as follows:

Table 9.4.1-1 (Sheet 4 of 6)

Component Data - Nuclear Island Nonradioactive Ventilation System

(Nominal Values)

Class 1E Battery Room Exhaust Subsystem

Exhaust Fan Data

Quantity per electrical division	2
System capacity per fan (%)	100
Design air flow rate (scfm)	2,400
Fan static pressure (in. WG)	3.5
Motor nameplate horsepower	~~3.0~~ 5.0

Table 9.4.1-1 (Sheet 5 of 6)

Component Data - Nuclear Island Nonradioactive Ventilation System

(Nominal Values)

D. Division "B" and "D" Class 1E Electrical Room HVAC Subsystem

Supply Air Handling Units

Quantity	2
System capacity per unit (%)	100
Design air flow (scfm)	~~15.00~~ 15,000
Fan static pressure (in. WG)	6.5
Motor nameplate horsepower	30

Question 435.75

The AP600 RTNSS report (WCAP-13856) identifies three screening criteria that are used to determine if nonsafety SSCs involved in the calculation of initiating event frequencies are important enough to be considered for regulatory oversight. The last two criteria appear to be inconsistent with the RTNSS process identified in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. These criteria eliminate nonsafety SSCs (that prevent occurrence of initiating events) from regulatory oversight based on a relative measure of their importance; whereas the Commission paper states that "nonsafety SSCs used to prevent the occurrence of initiating events will be subject to regulatory oversight applied commensurate with their R/A missions for prevention." The criteria are especially inappropriate for evaluating the offsite power systems in the AP600 design because application of the criteria has resulted in elimination of the offsite power systems from regulatory oversight during power operating modes, although they are key to the establishment of loss of offsite power frequency and have always received regulatory attention. The staff believes that the availability of the offsite power sources need to be controlled by technical specifications during all plant modes. Address this concern.

Response:

The regulatory oversight for offsite power systems during power o[illegible] provided in the technical specification requirements for the operability of offsite (and onsite) electrical power sources for existing plants, is based on meeting the technical specification screening criteria in reference 435.75-1. For existing plant technical specifications, the safety-related ac power is required to provide the capability to mitigate the consequences of an event (Criterion 3 of reference 435.75-1). These ac power sources are needed for operation of the various pumps that directly provide safety injection and core cooling, along with the cooling water for the various support system pumps. The technical specification screening criteria do not identify components for inclusion in technical specifications because they can help to preclude the initiation of the event.

For the AP600, no ac power sources are required to support the operation of the safety-related passive systems in the performance of their mitigation functions following an event. The results of the focused PRA described in reference 435.75-2 confirms that the nonsafety-related ac power systems, including the offsite power sources, are not significant to plant safety, and therefore, are not identified by Criterion 4 of reference 435.75-1. The application of the screening criteria in reference 435.75-1 to the AP600 SSCs does not identify any required ac power sources. The technical specifications in SSAR Subsection 3.8 of Section 16.1 provide the required electrical power system LCOs which are identified by the screening criteria. The offsite power systems and the associated onsite power systems, such as the transmission switchyard and plant ac distribution equipment, are not required to be included in the regulatory oversight provided by technical specifications.

The RTNSS process described in reference 435.75-2 includes a comprehensive evaluation of nonsafety-related SSCs that prevent the occurrence of initiating events. Consistent with reference 435.75-3 (approved by the Commission), the last two criteria provide a specific means to evaluate the importance of nonsafety-related systems for prevention of specific initiating events considered in the AP600 PRA. Reference 435.75-3 requires an evaluation of the impact and significance of the reliability/availability missions for prevention of the initiating events on plant safety. The regulatory oversight identified for the nonsafety-related SSCs should be based on the significance of the mission(s) they perform in preventing an initiating event. The criteria in reference 435.75-2 were specifically developed to quantitatively assess the significance of these missions.

For AP600, the safety-related passive systems do not require ac power sources to perform their accident mitigation functions. Therefore, the availability of ac power following an event will not prevent these systems from performing their safety-related functions. The results of the PRA evaluation confirms that the loss of offsite power is not important from the perspective of at-power risk.

Using this process, an RTNSS-significant mission is identified for ac power sources and the associated electrical distribution equipment in reference 435.75-2. The mission identified is to prevent the loss of offsite power during shutdown with reduced RCS inventory conditions. As discussed in reference 435.75-2, the need for additional regulatory oversight during reduced inventory, shutdown conditions, was identified and short-term availability controls are provided in Table 10.3-3.

References:

435.75-1 Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, Federal Register, Volume 178, Number 139, July 22, 1993.

435.75-2 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

435.75-3 SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 1994.

SSAR Revision: NONE

Question 440.54

The staff addressed concerns relating to rapid boron dilution during a PWR startup raised by the French regulatory authority in NUREG/CR-5819. The French authority postulated a scenario that starts with the highly borated reactor being deborated as part of a startup procedure. The reactor is at hot condition with the reactor coolant pumps (RCPs) running and the shutdown banks removed. Unborated or diluted water is being pumped by charging pumps from the volume control tank into the cold leg.

The initiating event is a loss of offsite power that results in tripping the RCPs and charging pump, and scramming the shutdown rods. The charging pump comes back on line quickly when the diesel generators start up. Charging continues until the volume control tank is empty. This diluted water is assumed to accumulate in the lower plenum. It is then assumed that the offsite power is recovered and the RCPs are restarted. The RCP restart causes the slug of diluted water to rapidly pass through the core and results in a potential to cause a power excursion sufficiently large to damage the core. Another variation to this scenario includes an event having the slug of deborated water through the core by inadvertent blowdown of an accumulator. In light of these potential rapid boron dilution scenarios, show the adequacy of the AP600 design by demonstrating that the rapid boron dilution events are incredible, the results are not serious if they occur, or proposing protective measures.

Response:

The chemical and volume control system is designed to address a potential rapid boron dilution scenario in the event of a loss of power to the two CVS remotely-operated demineralized water system isolation valves. When power is interrupted the chemical and volume control system makeup pumps stop and two safety-related motor operated gate valves, in series, from the demineralized water system automatically close to isolate the unborated water source. In addition, the three-way chemical and volume control system makeup pump suction valve is automatically aligned to the boric acid tank. The chemical and volume control system makeup pumps are sequenced onto the diesel generator but will not restart unless actuated by a low pressurizer level signal or the makeup control system. In the event that the pumps are actuated, the system is aligned to take suction from the borated water source and the unborated water source is isolated. Restoration of power to the isolation valves does not result in the valves opening. The AP600 chemical and volume control system does not have a volume control tank. Loss of power does not result in a dilution event.

Refer to the response to RAI 440.122 for a related discussion.

SSAR Revision: NONE

Question 440.58

Describe what changes have been incorporated into Chapter 16, "Technical Specifications," of the SSAR for AP600 to deal with shutdown operations. Identify any deviations from the guidance specified in NUREG-1449 (Sections 6.5 and 7.3.2) for shutdown Technical Specifications and justify the deviations with appropriate technical bases (see also Q440.53, Q440.55, Q440.56, Q440.71, and Q440.72).

Response:

The specific technical specification guidance provided in Subsections 6.5 and 7.3.2 of NUREG-1449 relates to concerns with the shutdown operation of existing plants. Existing plants have substantially different system designs and are operating with technical specifications based on NUREG-0452 or earlier standards. However, the underlying concerns relating to causes of events and recovery from those events during shutdown operations are applicable to AP600. The objective of the NUREG-1449 guidance for improvements to existing technical specifications is to reduce the likelihood that an event will occur during shutdown conditions and to help assure that systems are available to mitigate the consequences of an event that occurred during shutdown.

The NUREG-1449 position, that additional requirements for decay heat removal capability are needed during shutdown, is based on reducing core damage frequency. The core damage frequency is both a function of the frequency of initiating events and of the effectiveness of mitigating systems. For existing plants, an acceptable core damage frequency is heavily dependent on reduction in the frequency of initiating events. However, for the AP600, the focused PRA sensitivity study performed for the RTNSS evaluation and documented in Reference 440.58-1, shows that core damage frequency goals can be met assuming no credit for mitigation functions of nonsafety-related systems such as the active decay heat removal systems and their supporting systems. Additionally, the normal residual heat remvoal system does not meet Criterion 4 of Reference 440.58-2. Therefore, due to the availability of passive safety-related systems that have specific requirements for operability during shutdown conditions, there are no LCO requirements for the normal residual heat removal system during shutdown or refueling for AP600.

Table 440.58-1 summarizes the technical specification LCO requirements for safety injection and core cooling using the safety-related, passive systems during shutdown conditions in Modes 5 and 6. These requirements help to assure the availability of the required systems and equipment needed in the event that the normal residual heat removal capability is lost. The LCO requirements for the safety-related, passive systems are based on meeting Criterion 3 of the Reference 440.58-2 and are consistent with the guidance provided in NUREG-1449.

The electrical system LCOs for shutdown operations are consistent with NUREG-1431. These LCOs require that electrical sources and distribution trains be OPERABLE as required to support systems and equipment required to be OPERABLE by other LCOs.

The technical specification LCOs for the associated instrumentation and control systems will be revised to support detection and mitigation of shutdown events. The revised LCOs will be submitted with revision 2 of the AP600 SSAR.

References:

440.58-1 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

440.58-2 Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, Federal Register, Volume 178, Number 139, July 22, 1993.

SSAR Revision:

The associated technical specifications, along with revised bases for each of the revised technical specifications will be included in revision 2 of the AP600 SSAR.

Table 440.58-1

MODE	Automatic Depressurization System	Core Makeup Tank	Passive RHR	IRWST	Containment	Containment Cooling
MODE 5 RCS pressure boundary closed	9 of 10 paths OPERABLE All paths closed	One CMT OPERABLE	One heat exchanger OPERABLE	One injection flow path and one recirculation sump flow path OPERABLE	None	None
MODE 5 RCS pressure boundary open	Stages 1, 2, and 3 open	None	None	One injection flow path and one recirculation sump flow path OPERABLE	Closure capability	Two water flow paths OPERABLE
MODE 5 RCS pressure boundary open, reduced RCS inventory	Stages 1, 2, and 3 open	None	None	Two injection flow paths and two recirculation sump flow paths OPERABLE[1]	Closure capability	Two water flow paths OPERABLE
MODE 6 Reactor internals in place, refueling cavity not full	Stages 1, 2, and 3 open	None	None	Two injection flow paths and two recirculation sump flow paths OPERABLE[1]	Closure capability	Two water flow paths OPERABLE
MODE 6 Reactor internals removed, refueling cavity full	None	None	None	None	None	None

Note 1: The IRWST injection flow path isolation valves are closed in this mode.

Question 440.78

The NRHRS relief valve, which provides for the low temperature over-pressure protection (LTOP), is sized to prevent over-pressure of those credible events with a water-solid pressurizer. Section 5.2.2.1 of the SSAR states that the makeup/letdown flow mismatch and the inadvertent start of an active reactor coolant pump events, respectively, are the limiting mass and heat input conditions to size the relief valve. However, no analysis is provided in the Chapter 15 safety analysis regarding LTOP.

a. Provide the safety analyses and results for both the mass and energy input overpressurizing design basis transients that demonstrate proper sizing of the NRHRS relief valve for LTOP. The results should include transient curves that demonstrate that the peak RCS pressures are within the design pressure-temperature (P/T) limits determined for the AP600.

b. Provide the limiting single active failures and instrumentation uncertainties assumed in these analyses and the bases for these assumptions.

c. Provide the basis for the assumption made for nergy input transient that the water in the secondary side of steam generator is 50°F hotter than the , ary side.

d. The nil-ductility reference temperature of the reactor vessel material increases as exposure to neutron fluence increases due to neutron embrittlement effect. Therefore, the operating P/T limit curves need to be periodically adjusted to accommodate the actual shift in the nil-ductility temperature, and the LTOP system must be re-evaluated to ensure that its functional requirements can still be met using the NRHRS suction relief valve. Does the relief valve sizing analysis take this into consideration? Are the sizing and setpoint of the relief valve based on bounding P/T curves that are applicable to the life of the plant?

e. The NRHRS is not designed to be a safety-related system. Because the NRHRS relief valve is required for LTOP, confirm that the relief valve as well as piping in the NRHRS are designed to meet safety-related criteria.

Response:

a. The normal residual heat removal system (RNS) relief valve, which mitigates the low temperature overpressure transients is sized to prevent the RCS pressure from exceeding the applicable pressure-temperature (P/T) limit. The limiting mass and energy input transients assumed for the sizing analysis are as follows:

- <u>Mass Input:</u> Maximum makeup water flow to the RCS assuming both CVS makeup pumps are in operation and letdown is isolated. Figure 440.78-1 shows the mass flow rate assumed in the analysis. This transient is postulated to occur over a range of reactor coolant temperatures between 100°F and 350°F.

- <u>Energy Input:</u> Restart of one reactor coolant pump with water in the steam generator secondary side 50°F hotter than the primary side water, and the RCS water-solid. This transient is postulated to occur over a range of reactor coolant temperatures between 100°F and 200°F.

The minimum RNS relief valve capacity has been calculated at an RCS pressure equivalent to the valve setpoint of 563 psig plus 10% accumulation (619 psig). With this setpoint, the relief valve would mitigate the limiting LTOP transient while maintaining the RCS pressure less than the P/T limit. The nominal steady-state P/T limits applicable up to 54 effective full power years (EFPY) as shown in Figure 440.78-2 is assumed in the analysis. Since the relief valve does not have a variable P/T lift setpoint, the setpoint plus 10% accumulation must be less than the bounding P/T limit. From Figure 2, the bounding P/T limit is 621 psig at the flange.

The results of the analysis show that the mass input transient is limiting up to 205°F. Since, the energy input transient is considered not credible above 200°F, the mass input transient is limiting for the entire range of the LTOP operation. With RCS in water-solid condition, the minimum RNS relief valve capacity required is 555 gpm which is the maximum makeup water flow at 619 psig RCS pressure.

A transient curve that illustrates that the peak RCS pressures are within the design P/T limits has not been generated in the analysis since the transient response is a constant RCS pressure at 619 psig for all RCS temperatures less than 350°F.

b. Single active failure is not considered for passive valves such as the RNS self-actuated spring relief valve. As such, the analysis does not consider a single failure of this valve. Also, no single active failure can occur in the RNS that could prevent the RNS suction relief valve from performing its functions.

The 10% setpoint accumulation includes a 3% setpoint uncertainty. No other uncertainties are explicitly modeled in the analysis.

c. The 50°F for the energy input transient was derived from an evaluation of typical administrative controls utilized during heatup and cooldown operations of standard Westinghouse plants. At least two reactor coolant pumps are maintained in operation whenever the RCS temperature is greater than approximately 160°F, which is sufficient to establish isothermal conditions on the primary side. The steam generator secondary side water immediately surrounding the tubes will also remain at a temperature near that of the circulating reactor coolant on the primary side.

During cooldown operations, when the reactor coolant temperature has been decreased to approximately 160°F, the reactor coolant pumps are stopped. Cooldown of the RCS continues via circulation through the RNS heat exchangers. However, RNS operation effectively bypasses the steam generators, and both the primary and secondary sides of the steam generators could remain at a relatively constant temperature greater than the RCS temperature. Cooldown of the RCS continues from 160°F at approximately 24 hours after shutdown, to 120°F at approximately 96 hours after cooldown. At this temperature, the reactor vessel head bolts can be loosened and refueling operations can begin. Therefore, a maximum ΔT of 40°F between the RCS and the steam generators could develop while the RCS was intact. The LTOP analysis performed used a 50°F ΔT as the initial condition for the energy input transient to conservatively bound the cooldown scenario described above.

d. The normal pressure-temperature heatup and cooldown curves for the AP600 plant are developed for a 60 year design life with 90 percent availability. This equates to 54 effective full power years (EFPY). The current

analysis for the relief valve sizing is based on the 54 EFPY bounding P/T limit curves. For EFPY greater than 54, or if the P/T curve is revised, the relief valve setpoint must be re-evaluated. The current relief valve setpoint of 563 psig is based on the bounding P/T limit of 621 psig.

e. The RNS relief valve and associated piping are safety-related. See the responses to RAI's 210.37, 210.061, 440.084 and 440.127 for further information addressing the safety-related design criteria applied to the normal residual heat removal system piping and components.

SSAR Revision: NONE

Figure 440.078-1

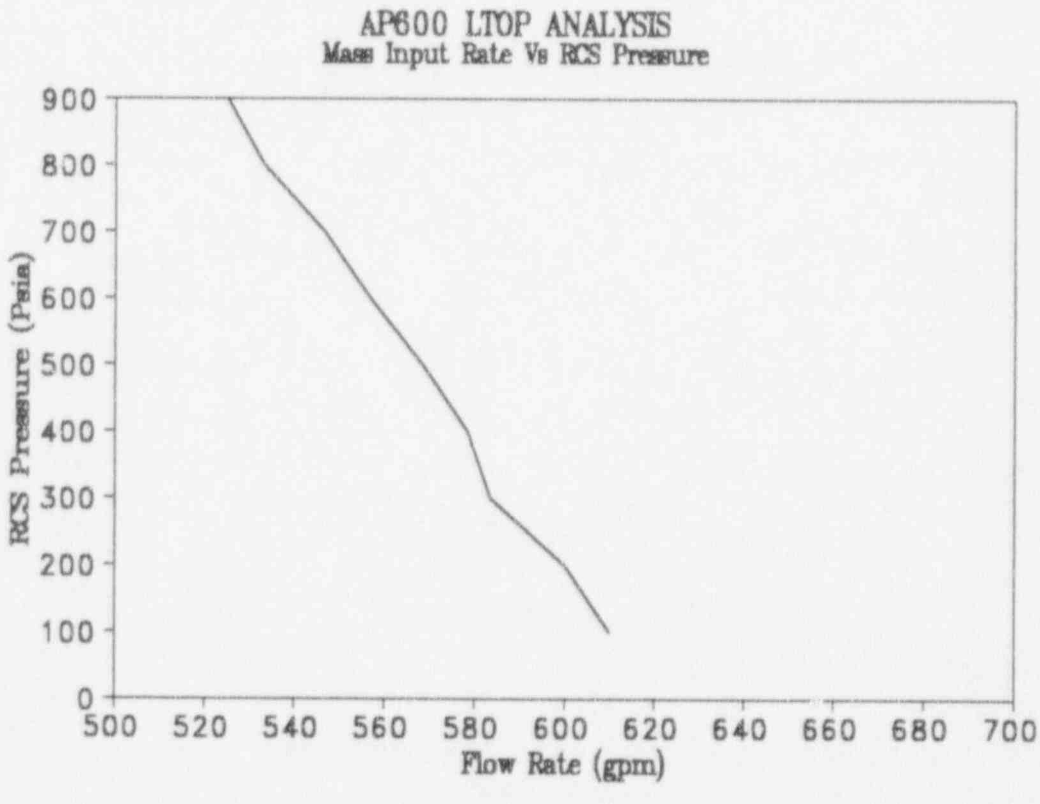
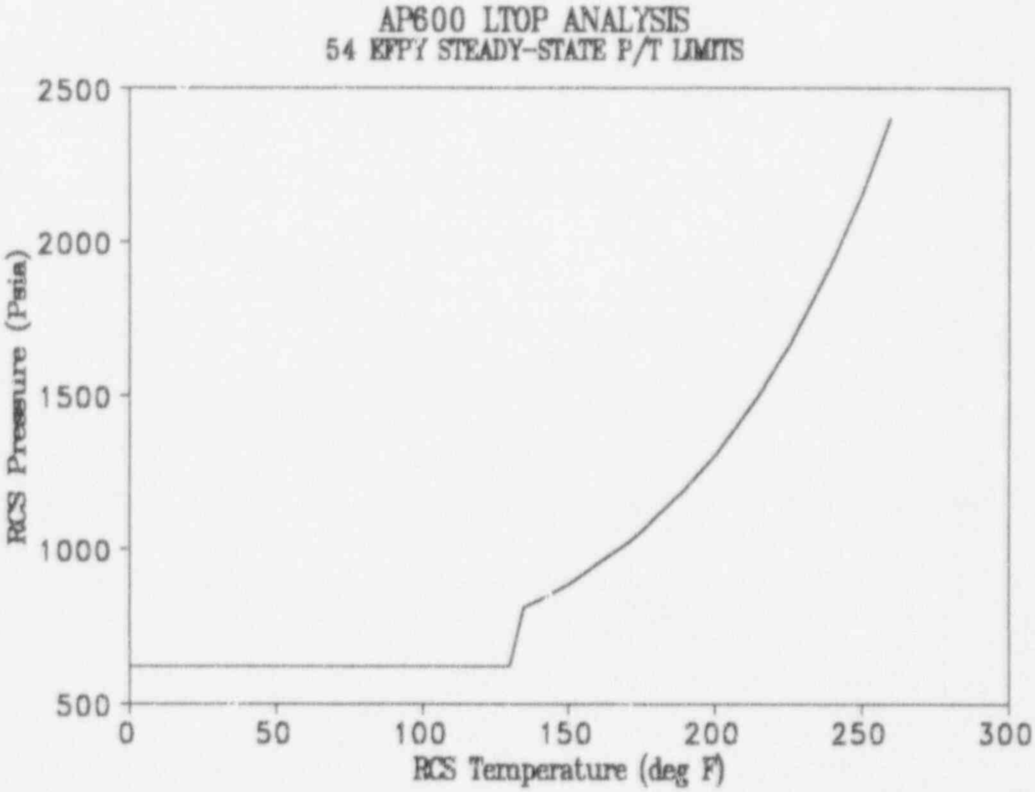


Figure 440.078-2



Question 440.122

Section 9.3.6.1.1 of the SSAR states that the safety functions provided by the chemical and volume control systems (CVCS) are limited to containment isolation of the CVCS lines penetrating the containment, termination of inadvertent reactor coolant system boron dilution, isolation of makeup on a steam generator or pressurizer high level signal, and preservation of the RCS pressure boundary. For termination of inadvertent boron dilution, Section 9.3.6.4.5.1 of the SSAR states that following a reactor trip signal, the demineralized water system (DWS) line is isolated by closing two remotely-op[illegible]ted DWS isolation valves, and the three-way pump suction control valve aligns to take suction from the boric [illegible]k. In Technical Specification (TS) Table 3.3.2-1, "Engineered Safeguards Actuation System Instrumentati[illegible]apter 16 of the SSAR, it is not clear how and what actuation logic or signal is used to accomplish the DWS makeup isolation in an inadvertent boron dilution event occurring during various modes of plant operation. Provide this information.

Response:

SSAR Subsection 9.3.6.4.5.1 describes the CVS [illegible]on for termination of inadvertent boron dilution events. SSAR Subsection 15.4.6 discusses the boron dil[illegible]yses and summarizes the protection provided during all plant operating modes. Boron dilution events are pr[illegible]ented during refueling (mode 6) by administrative controls that isolate the RCS from the potential source of unborated water by locking closed, two CVS demineralized water system isolation valves and aligning the three-way CVS makeup pump suction valve to the borated water source (boric acid tank). Inadvertent dilution events are automatically terminated during cold shutdown (mode 5), hot shutdown (mode 4), and hot standby (mode 3) modes upon a reactor trip signal, source range flux-doubling signal, safety injection signal, or by a loss of power to the two CVS remotely-operated demineralized water system isolation valves. Inadvertent boron dilution events during start-up (mode 2) or power operation (mode 1) result in an automatic reactor trip. Following reactor trip, au[illegible]atic termination of the dilution occurs, by isolating the unborated water source and realigning the three-way CV[illegible] up pump suction valve, and any post-trip return to criticality is prevented.

SSAR Section 16.1, Table 3.3.2-1, items 11.a and 11.c provide the actuation signal for boron dilution protection from the ESFAC logic and the source range instrumentation. The table will be modified to include item 11.d, the isolation signal from the P-4 reactor trip and item 11.e, the isolation signal generated as a result of a loss of power to the CVS demineralized water system isolation valves. The functional requirements and the functional diagram 7.2-1 (sheets 2, 3) will also be modified to include this actuation signal.

SSAR Revision:

The SSAR will be revised as indicated in the response to RAI 420.92 to include the reactor trip signal isolation of unborated water from the demineralized water system.

The protection system functional diagrams will be revised to include the signal that isolates the demineralized water system on a loss of power to the two CVS remotely-operated demineralized water system isolation valves. This information will be included in Revision 2 of the SSAR. The appropriate subsections within SSAR Sections 9.3.6, 15.4.6, 16.1, and 7.2 will be revised to incorporate this signal.

Question 440.128

Section 5.4.7.1.2.1 of the SSAR states that the RNS is designed to successfully reduce the RCS temperature from 350°F to 120°F within 96 hours after shutdown, and maintain the RCS temperature at or below 120°F for the entire plant shutdown with both subsystems of RNS pumps and heat exchangers available, and that a failure of an active component during normal cooldown will not preclude the ability to cooldown, but will only lengthen the time required to reach 120°F.

a. Has a single failure analysis (such as a failure modes and effects analysis) been performed to determine the limiting single failure?

b. How long will it take to cooldown to 120°F if one of the two subsystems is not available?

Response:

a. A failure modes and effects analysis was not performed for the operating modes of the normal residual heat removal system during plant cooldown since these analyses are only required for safety-related systems.

Redundancy is provided in the normal residual heat removal system (RNS) so that, once the system is aligned for decay heat removal, the capability to remove reactor decay heat is maintained if one RNS pump cannot be started (or stops during the decay heat removal operations), if cooling is lost to an RNS heat exchanger, or if the normal electrical supply to a pump is lost. Additional reactor decay heat removal redundancy is provided by the nonsafety-related startup feedwater system, the nonsafety-related spent fuel pit cooling system, the safety-related passive core cooling system, or the safety-related level in the refueling cavity/spent fuel pit--depending on the plant operating mode.

b. One RNS subsystem (one RNS pump and RNS heat exchanger combination) is capable of cooling the RCS from 350°F to 200°F (Cold Shutdown) in 35 hours after reactor trip and from 200oF to 120oF in 11 days. One RNS subsystem cools the RCS from 350°F to 120°F in a total of 12.5 days after reactor trip. This cooldown performance assumes that the RNS system is initiated at 4 hours after reactor trip.

SSAR Revision: NONE

Question 440.131

Appendix 1A of the SSAR indicates that Regulatory Guide (RG) 1.1 is not applicable to the AP600 RNS, because it is not a safety system and does not control or mitigate the consequences of an accident in the licensing basis accident analyses. RG 1.1 requires that emergency core cooling and containment heat removal systems be designed to provide adequate NPSH to the system pumps assuming maximum expected temperatures of pumped fluids. Since the RNS is a safety-significant system based on the analysis of regulatory treatment of non-safety systems, what are the bases for not designing the system to meet this requirement?

Response:

As documented in Reference 440.131-1, the normal residual heat removal system is identified as an RTNSS-significant nonsafety-related system during reduced reactor coolant system inventory midloop operations. To support the AP600 PRA initiating event frequency for the shutdown loss of decay heat removal event, this system is required to be in continuous operation to support shutdown core decay heat removal during reduced reactor coolant system inventory conditions. To support this function, Reference 440.131-1 provides short-term availability recommendations specifying that the normal residual heat removal system be available prior to initiating reduced reactor coolant system inventory operations during a plant shutdown. In addition, Reference 440.131-1 indicates that planned maintenance on the normal residual heat removal system will normally be scheduled during Mode 1. These short-term availability recommendations are provided in Reference 440.131-1 as the additional regulatory oversight for the normal residual heat removal system consistent with the process documented in Reference 440.131-2.

As defined in Reference 440.131-2, the RTNSS process provides a mechanism for identifying significant nonsafety-related systems and developing corresponding appropriate regulatory oversight. The application of guidance provided in Regulatory Guide 1.1 relative to providing adequate NPSH to the normal residual heat removal system pumps assuming maximum expected temperatures of pumped fluids is not necessary for the normal residual heat removal system to accomplish its RTNSS-significant function.

References:

440.131-1 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

440.131-2 SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 1994.

SSAR Revision: NONE

Question 440.136

Section 5.4.7.4.4 of the SSAR states that for accident recovery operations, upon actuation of automatic depressurization, the RNS can be employed to provide low pressure RCS makeup, and that operation of the RNS will not prevent the passive core cooling system (PXS) from performing its safety functions. What tests or analyses have been made to demonstrate proper PXS performance upon RNS operation?

Response:

Integral systems tests are in progress at the full height, full pressure SPES test facility in which small break LOCA transient tests will be performed. Tests to be performed include a small LOCA without the normal residual heat removal system available, and a repeat test with the normal residual heat removal system available.

In addition to the SPES tests, there are also tests which have been performed in the low pressure, reduced height, integral test facility at Oregon State University for a small-break LOCA to examine interactions of the nonsafety-related systems and the passive safety-related systems. A two-inch small break LOCA experiment was conducted with only the passive systems available. A repeat experiment was performed with both the passive systems and the normal residual heat removal system being available.

These experiments provide the necessary data to validate the AP600 safety analysis computer codes for passive safety-related and active nonsafety-related system interactions.

SSAR Revision: NONE

Question 440.181

It appears that, given the failure of the CMTs, it is assumed that operator action to initiate the ADS occurs at 10 minutes. Has an analysis been performed to establish the realistic amount of time available for the operator to initiate the ADS for the spectrum of initiators and sequences where manual action is credited? Provide an analysis to establish these time frames. Are these actions credited in the September 24, 1993 focused PRA (see also Q440.177)?

Response:

The manner in which operator action time frames were defined is discussed in the response to RAI 720.276, item b. These actions are credited in the analysis for the September 24, 1993 focused PRA documented in Reference 440.181-1.

Reference:

440.181-1 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.190

In several portions of the Design Change Description Report (February 15, 1994), it is stated that PRA modelling is not affected by the design changes. For example, the following statements are made in the report:

P2-8. The change in the CVS control logic improves the expected operation of the plant during non-LOCA events by minimizing the potential of automatic actuation of ADS. This change does not affect how the CVS is modeled in the PRA.

P2-11: The modification of the ADS Stage 1 setpoint increases the margin to automatic ADS actuation. This change does not affect the success criteria used in the PRA.

P2-13: Actuation of ADS Stages 2 and 3 on times instead of CMT level will not affect the ADS reliability.

P2-16: The change in the CMT by adding an inlet diffuser will have no impact on the PRA since it does not affect how the systems are modeled and has no effect on success criteria.

Does this mean that the safety benefit is so small as to be unquantifiable, or that modelling uncertainties in the PRA overwhelm the expected benefits?

Response:

When a design change is proposed for AP600, its effect on the PRA model is assessed by identifying if the change affects the following:

- Success criteria
- Comp. nent (or other) failure modes
- Probab lity of a failure
- Consequences of accidents.

In many cases, the effect is deemed to not be significant. This may be because of the nature of the PRA models, where basic events can only be modeled at "high levels" and do not contain detailed quantification of the failure modes.

Based on this assessment, the risk benefits involved in the quoted design changes are deemed to be unquantifiable with the current PRA models, implying that the expected risk benefits are "small".

PRA Revision: NONE

SSAR Revision: NONE

Question 440.196

In Figure F-11 of the PRA for an MSLB upstream of MSIV, there are no safety grade success paths following the PRHRS failure (PRT) if the MSL isolation fails (CIA). Why not?

Response:

In Figure F-11, events subsequent to main steam line isolation and passive RHR failure where gravity injection is successful lead to core damage category 2. Category 2 is defined as a containment isolation failure with successful emergency core cooling, including depressurization with gravity injection. The PRA report assumed that category 2 sequences would lead to core damage because the containment was not isolated and the loss of water inventory as steam to the environment would eventually uncover the core and lead to core damage. MAAP4 runs made since the PRA report submittal indicate that core uncovery is not expected until much later (beyond 72 hours). Revision 2 of the PRA report will reclassify these sequences as successful accident mitigation.

SSAR Revision: NONE

PRA Revision:

Revision 2 of the PRA, scheduled for December 31, 1994 will include a reclassification of these sequences as discussed above.

Question 440.198

Is PRHR "control" (e.g., path 2 on T_{SOV} tree of the PRA) credited for purposes of the RTNSS evaluation?

Response:

Following a main steam line break or a main steam line stuck open safety valve (T_{sov}) event, the RTNSS evaluation documented in Reference 440.198-1 assumes credit for PRHR control. PRHR control is modeled in the event trees as the CM (passive RHR or CVCS makeup) top event. For the RTNSS evaluation, the failure probability for top event CM is set to the probability that the operator fails to stop and start the PRHR system. The RTNSS evaluation assumes no credit for makeup provided by the operation of the chemical and volume control system.

Reference:

440.198-1 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

SSAR Revision: NONE

Question 440.199

Why isn't recirculation queried on the S1S tree of the PRA?

Response:

The query on recirculation of water to the reactor pressure vessel from the containment sump after successful gravity injection is not used because it is assumed that this function is accomplished directly through the direct vessel injection line break.

SSAR Revision: NONE

PRA Revision: NONE

Question 480.55

Sheet 2 of Table 6.2.3-1 of the SSAR lists two unidentified normal residual heat removal system penetrations. One contains a 3-inch in-containment gate valve, the other a ⅜-inch in-containment globe valve. Identify these penetrations by service and PI&D figure number.

Response:

The 3 inch connection is depicted on Figure 5.4-7 in the RHR suction line and identified as "CVS REGEN HX". The line serves as a low pressure letdown source during normal RHR operation. The subject 3 inch gate valve is depicted on Figure 9.3.6-2 sheet 1 as a 3 inch locked closed manual gate valve. The connection to the RNS from the CVS is identified as "RNS PUMP SUC".

The ⅜-inch globe valve is in the 1 inch line on Figure 5.4-7 in the RHR suction line identified as "PXS TEST HDR". The line serves as a leakage detection function for the RNS suction isolation valves. The ⅜-inch globe valve is depicted on Figure 6.3-3 as a ⅜-inch locked closed manual globe valve. The connection to the RNS from the PXS is identified as "RNS NRHR SUC".

SSAR Revision:

SSAR Table 6.2.3-1 has been revised to better depict the subject valves as an integral part of the RNS penetration design. Revised table is provided in response to RAI 480.61

Question 480.76

During the March 22, 1994 meeting, Westinghouse indicated that one of the containment isolation design features of the AP600 is the reduction in the number of penetrations (40 vs. 100). The staff believes that the bulk of this reduction has been achieved by ganging more lines together per penetration. This, in turn, may imply larger penetrations. Are any of the AP600 penetrations now so large that they are beyond "state-of-the-art?" If so, demonstrate that these lines are as safe as existing designs.

Response:

The containment penetrations are identified in Table 6.2.3-1. The systems details from the table demonstrate that the AP600 penetrations are of equivalent size as existing designs. For example:

- The largest fluid penetrations are the main steam and feedwater lines at 32 and 16 inches respectively; lines of this size are typical for operating PWR plants.

- The next largest lines penetrating containment are the containment purge lines. The containment isolation valves are slightly larger than a typical minimum purge line but dramatically smaller than the typical 36 to 54 inch normal purge line isolation valves.

- The chilled water, component cooling, and normal residual heat removal systems are the remaining larger lines penetrating containment and are, on the average, slightly smaller and fewer in number than on existing designs.

- The remainder of the penetrating fluid lines as well as the fuel transfer canal isolation provisions, personnel airlocks and equipment hatches are typical in size to operating PWR's.

The reduction in the number of penetrations has been accomplished primarily by

- application of passive safety systems located within containment requires significantly fewer penetrations than existing active safety related systems,

- design refinements directed at limiting the number of containment penetrations via either eliminating the need for the system inside containment or combining functions served by a given penetration, e.g. the normal residual heat removal penetration combines the ability to remove decay heat, provide for IRWST cooling and low pressure letdown, and

- elimination of reactor coolant pumps requiring seal injection support.

SSAR Revision: NONE

Question 720.262

Provide a detailed explanation of the spectrum shape used in the AP600 seismic margins analysis. If the risk-based seismic analysis in the AP600 PRA does not bound the site-specific parameters of the actual site chosen, an applicant for a combined construction/ operating license will have to provide a new, site-specific risk- based seismic analysis.

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.263

Provide a list of structure, system, and component fragilities and HCLPFs. The list should include the median capacity, BC, and HCLPF, as discussed below:

a. Provide the mathematical definition of HCLPF.

b. Provide fragility/HCLPF information for plant essential structures (e.g., containment and auxiliary buildings) that house safety-related systems and functions credited in the seismic analysis (e.g., passive RHR and DC power), including passive and active systems.

c. Provide the fragilities/HCLPFs for all systems (passive and active) evaluated in the AP600 PRA seismic analysis, including RCS primary equipment and supports.

d. Provide the component fragilities/HCLPFs for the individual components modelled in the AP600 seismic analysis.

e. For each of the above, (1) indicate if the fragility estimate is based on a design-specific Westinghouse analysis, or if Westinghouse used a generic fragility, and (2) where generic fragilities were used, provide a basis for their use in the AP600 design with special attention provided to unique design components in the AP600 design (such as the core, check valves, and core makeup tanks).

Response: (Revision 1)

~~The response to the subject questions are provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.264

Provide the AP600 plant HCLPF based on those sequences leading to core damage.

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.265

Provide a reference in the AP600 PRA to the ITAAC requirement that failure of non-seismically qualified structures, systems, and components will not physically damage or inhibit the operation of seismically qualified equipment.

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.266

Provide an evaluation in the AP600 risk-based margins analysis of the effect of seismic failure of non-seismic equipment that interfaces with Seismic Category I equipment (e.g., mainsteam line rupture).

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.268

The description of the methodology for the seismic margins analysis does not specify how to treat seismic events during shutdown. During shutdown, the safety systems may not be able to function following a seismic event (e.g., due to maintenance), depending on the status of the plant. In addition, the non-safety systems may have been disabled by the same seismic event. Provide a risk-based evaluation of the plant HCLPF during shutdown, including the use of non-safety grade equipment for prevention and mitigation of core damage, containment failure, or offsite releases.

Response:

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

At least one passive safety-related heat removal feature is available during all modes of shutdown. With the reactor coolant system pressure boundary intact, the passive residual heat removal system heat exchanger is available. With the reactor coolant system pressure boundary open, automatic depressurization system venting and in-containment refueling water storage tank injection are available. With the reactor coolant system in the refueling mode, the refueling cavity inventory provides decay heat removal. These systems provide safety-related single failure tolerant means of removing decay heat should the nonsafety-related normal cooling system fail.

A qualitative assessment is used to show that the seismic margins defined for mode 1, bound those during shutdown. The qualitative assessment is provided as Attachment 1 to this RAI response.

Based on this assessment, the vulnerable shutdown state is during mid-loop operations. With a loss of offsite power and no ac power available, the normal residual heat removal system does not function. During mid-loop conditions the passive residual heat removal heat exchanger is ineffective and the accumulators and core makeup tanks are not required to be operable. The technical specifications require that the automatic depressurization stages 1/2/3 be open and the in-containment refueling water storage tank injection paths be operable in this condition. Containment closure capability is also required. This results in passive safety-related protection against a seismic event in this shutdown state.

SSAR Revision: NONE

PRA Revision: NONE

ATTACHMENT 1 to RAI 720.268(R1)

SEISMIC EVENT DURING SHUTDOWN

The plant can be shutdown for a variety of conditions: following reactor trip; to perform required (or emergency) maintenance as defined by tech specs; or for refueling. The state of the plant for each of these conditions may change and the time spent in each state may change, but the plant shutdown can be characterized by several parameters: reactor coolant system temperature and pressure; whether the reactor coolant system is closed or vented; how much water is in the vessel; what equipment can function; and what equipment is available. The traditional shutdown modes can be subdivided based on these parameters.

Plant Mode	Shutdown State	Plant Status
1	- -	The safety-related systems are available for automatic actuation.
2	Same as 3A	Same as 3A described below, except that boron dilution to allow a return to criticality may also be in progress.
3	3A	This is mode 3 with reactor coolant system pressure above ~ 1000 psig. The safety-related systems are required to be operable, including the accumulators. The startup feedwater system is operating for decay heat removal. Passive containment cooling is available and containment integrity is required.
3	3NA	This is the same as 3A, except that the reactor coolant system pressure is below 1000 psig and the accumulators are isolated.
4	4	In this state, the reactor coolant system temperature and pressure have been reduced and the normal residual heat removal system is in operation. The safety-related systems except the accumulators are required to be operable.
5	5I	This state is cold shutdown with the reactor coolant system intact and decay heat removal by the normal residual heat removal system. The pressurizer has a visible water level. The passive residual heat removal heat exchanger is available. One core makeup tank and one in-containment refueling water storage tank injection/containment recirculation line is available. Automatic depressurization system is available. The accumulators are isolated. Passive containment cooling is available if the containment is closed. Containment integrity/closure is not required.

5	5F	This state is cold shutdown with the reactor coolant system open and decay heat removal by the normal residual heat removal system. The pressurizer has a visible water level. Automatic depressurization system stages 1/2/3 paths are open. One in-containment refueling water storage tank injection/containment recirculation line is available. The core makeup tanks and accumulators are isolated. The passive residual heat removal heat exchanger and the steam generators are not effective for decay heat removal. Passive containment cooling is available. Containment closure is available; equipment hatches are closed and air locks may be open if operable. Maintenance cables and pipes use permanent or maintenance penetrations.
5	5D	This state is the same as 5F except that the reactor coolant system has been drained such that there isn't a visible level in the pressurizer. The reactor coolant system level may be reduced to mid-loop and steam generator equipment hatches removed. Nozzle dams may have been installed in the steam generators. The passive system availability is the same as 5F except for the in-containment refueling water storage tank. In this state the in-containment refueling water storage tank injection lines are isolated. However, they both must be operable.
6	6U	This state is the refueling mode where the reactor vessel head may have been loosened or removed, but the upper internals are in place and the refueling cavity is not fully flooded. Systems operation and availability are the same as in state 5D.
6	6R	This state is the refueling mode where the reactor vessel head and the upper internals have been removed and the refueling cavity is fully flooded. None of the passive injection or heat removal systems are required. Passive containment cooling and containment integrity/closure are not required. In this mode the water in the refueling cavity provides the safety-related decay heat removal, providing at least 6 hours heating before boiling and at least 72 hours boiling before fuel uncovery.

Returning from refueling would generally reverse the above steps. A maintenance outage could proceed through state 5D if maintenance is required on reactor coolant system components.

SEISMIC EVALUATION

Since there are passive safety-related features available in each of the shutdown modes, there is no need for nonsafety-related systems to function during a seismic event. The only differences between the at-power seismic margin analysis and a shutdown seismic margins analysis would be:

- The nonsafety-related residual heat removal system is connected to the reactor coolant system. The normal residual heat removal system has an ASME class 3, seismic category I pressure boundary designed for 900 psig.

- The HCLPFs calculated for safety-related equipment in mode 1 are representative of the seismic margins during shutdown conditions.

The initiating events during shutdown were reviewed. The major failures addressed in a mode 1 seismic event also apply to shutdown conditions (seismic failures of buildings, reactor coolant system components and piping, reactor vessel). If the buildings, vessel and reactor coolant system remain intact, the initiating events would be loss of offsite power and loss of the operating decay heat removal system because of loss of all ac power (station blackout) combined with a possible loss of coolant event.

The nonsafety-related systems would not be operable, so it is unlikely that a boron dilution event could occur The makeup pumps and reactor coolant pumps cannot operate and the accumulators are isolated below ~1000 psig Therefore, mass or energy addition events leading to overpressure events would not occur. In addition, the isolation valves close on the chemical and volume control system potential boron dilution paths.

Mode 2, States 3A and 3NA (Mode 3) and State 4SF (Mode 4)

If the plant is in either mode 2 or mode 3 (states 3A or 3NA), the startup feedwater system would be feeding the steam generators. If the postulated seismic event occurred, the startup feedwater system would be lost, but the passive safety features would all be available and would be automatically actuated by the safeguards signal. Therefore, the impact of the seismic event would also be the same as discussed in the level 1 seismic margins analysis.

If a break should occur, the core makeup tanks and accumulators are available for injection, the automatic depressurization system would activate to allow in-containment refueling water storage tank injection and recirculation.

States 4 and 5I

When mode 4 is entered, accumulator injection is isolated by closing motor-operated valves. When the reactor coolant temperature and pressure reach approximately 350°F and 400 psig, the normal residual heat removal system is placed in service. If the postulated seismic event were to occur, the normal residual heat removal system would fail to remove heat, but would not rupture since it is ASME class 3, seismic category I. The reactor coolant system is pressurized so that the passive residual heat removal system could be manually activated by the operators. If the operators do not activate the system, and either boiling starts or inventory is lost through the normal residual heat removal relief valve, a safety injection signal is available on low pressurizer level to activate the core makeup tanks and passive residual heat removal heat exchangers. As a result, this state has automatic safety-related decay heat removal. Also, note that the valves in the passive residual heat removal system would open on loss of instrument air.

State 5F

This state is entered when the automatic depressurization stages 1/2/3 are opened. If the postulated seismic event were to occur, the normal residual heat removal system would fail to remove heat, but would not rupture since it is ASME class 3, seismic category I. The passive residual heat removal system would not be effective for decay heat removal. The core makeup tanks and the accumulators are isolated. If boiling reduces inventory through the open automatic depressurization system valves, in-containment refueling water storage tank injection will be initiated when there is sufficient head to open the in-containment refueling water storage check valves.

States 5D and 6U

When level is reduced, the in-containment refueling water storage tank water injection lines are isolated. The automatic depressurization system valves are open so that the reactor coolant system is vented. The safety injection signal is disabled and the hot leg level signal is used to monitor and control the reactor coolant system. If the postulated seismic event were to occur, the normal residual heat removal system would fail to remove heat, but would not rupture since it is ASME class 3, Seismic category I. The passive residual heat removal system would not be effective for decay heat removal. The core makeup tanks and the accumulators are isolated. If either a loss of coolant or boiling occurs, a low hot leg level signal would automatically actuate to open the in-containment refueling water storage tank motor-operated valves to allow injection and recirculation. If the hot leg level signal is not available, the operators would open the valves.

State 6R

After the upper internals are removed and the refueling cavity is flooded, the safety-related systems are not required. The water would heat up and begin to boil after the loss of normal residual heat removal. If the postulated seismic event were to occur, the normal residual heat removal system would fail to remove heat, but would not rupture since it is ASME class 3, seismic category I. However, boil down to the top of the fuel would not occur for three days with the containment open. Procedures will be in place to instruct the plant personnel to close the containment once ac power is available and provide alternate means to add water to the containment.

CONCLUSION

Essentially the same safety-related equipment would be included in the seismic margins at shutdown analysis as in the level 1 analysis. The HCLPF for this equipment during shutdown is equal to or greater than the 0.5g level. Since the HCLPFs are above 0.5 g, no critical sequences would be identified. The vulnerable shutdown state is mid-loop. However, there is passive safety-related protection against a seismic event in this shutdown state.

Question 720.269

Revise the seismic margins methodology to include seismically-induced ATWS events.

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.270

Provide diagrams of the systems modelled in the seismic margins analysis that show what is and is not seismic Category I (e.g., piping, isolation valves, etc.). These should be included in the analysis.

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE

Question 720.271

Expand the seismic margins analysis to include initiating events that are greater than 0.5g (e.g., up to 0.75g). One of the purposes of performing a risk assessment of the AP600 design is to develop a better understanding of the response of the plant to severe accidents and any potential weak links in the design. Core damage sequences with HCLPFs greater than 0.5g will not contain any vulnerabilities, but these sequences may provide important information about the balance of prevention and mitigation in the design and may provide vital information about SSCs that should be included in the RAP or ITAAC. An extreme example of a potential sequence of interest is represented by the following:

(Initiator: 0.55g HCLPF) * (Injection: 0.2g HCLPF) *
(Depressurization: 0.35g HCLPF)

Although failure of injection or depressurization would occur at a low HCLPF value, the initiator's HCLPF is so high that the sequence would not constitute a vulnerability. In this case, the designer and the NRC must ensure that the initiator HCLPF was 0.5g or higher when an AP600 plant is completed, and must ensure that this information is maintained for use by a future COL applicant so that they would not modify the plant design in a manner that lowers the HCLPF of this initiator in the as-built plant.

Response: (Revision 1)

~~The response to the subject question is provided in the seismic margin report. This report will be submitted to the NRC by June 30, 1994.~~

The response to this RAI is provided in the response to 720.158 Rev.2.

SSAR Revision: NONE

PRA Revision: NONE