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NSRA-APSL-93-0078
Docket No.: STN-52-003

March 18, 1993

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:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of November 16, 1992 and January 26, 1993. This transmittal completes the responses to the November 16, 1992 letter. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A. Attachment B is a complete listing of the questions associated with the November 16, 1992 letter and the corresponding Westinghouse letters that provided our response.

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

Brian A. McIntyre /for
Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

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

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ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED MARCH 18, 1993

RAI No.	Issue
410.016	Reactor Coolant Leakage
410.017	Reg. Guide 1.45, Position C.9
410.018	Reg. Guide 1.45, Position C.8
410.019	Reg. Guide 1.45, Position C.7
410.020	Reg. Guide 1.45, Position C.6
410.023	First stage ADS hydrostatic loads
410.025	Reg. Guide 1.52
410.027	Equipment requiring protection from flooding
410.028	Potential sources of flooding
410.030	Maximun flood level
410.033	Flood protection
410.034	Flood protection
410.037	PXS equipment location
410.040	Multi-door passageways leakage prevention
410.043	CCW layout
410.044	Flood hazards
410.046	Break protection from open cycle systems
410.047	Water tight doors
410.048	SFP cooling pumps & heat exchangers flood prot.
410.049	Flood consequences
410.050	Flooding protection for remote shutdown panel
410.051	Equipment requiring missile protection
410.052	Turbine missiles
410.053	Secondary missiles
410.054	Equipment protection
410.059	Stored energy - nuts, bolts and studs

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ATTACHMENT A
AP600 RAI RESPONSES
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RAI No.	Issue
410.061	Sources of missiles
410.062	Missile protection of remote shutdown panel
410.063	Equipment requiring missile protection
410.064	Credible missile sources
410.066	Missiles inside containment
410.067	Source of missiles
410.068	Equipment requiring missile protection
410.069	Missile protection
410.070	Strike probability
410.071	Missile protection of remote shutdown panel
410.072	Protection from externally generated missiles
410.073	Protection of PCS from external missiles
410.074	Fuel storage pool
410.075	Missile protection of remote shutdown panel
410.076	Pipe break effects
410.080	Turbine stop and FW control valves
410.081	Pipewhip restraints
410.083	Protection from pipe breaks
410.084	Environmental/flooding effects
410.086	Loads
410.087	RV asymetric loads
410.088	Control room/MSIV compartment separation
410.089	Pipe breaks
410.091	Instrumentation protection
410.093	Classification of Structures, Systems, Components
410.094	HVAC system drawings

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ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED MARCH 18, 1993

RAI No.	Issue
450.006R01	Control room ventilation
460.008	Radwaste management system source terms
460.009	Liquid radwaste
460.010	Gaseous radwaste
460.011	Solid radwaste
460.013	Exhaust duct radiation monitoring
460.014	SG blowdown & CCW radiation monitors
460.015	Sampling provisions
460.016	Accident monitoring

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ATTACHMENT B

CROSS REFERENCE OF WESTINGHOUSE RAI RESPONSE TRANSMITTALS TO NRC LETTERS OF NOVEMBER 18, 1992 AND JANUARY 29,1993

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
100.007	Pre-application RAI's	11/16/92	02/09/93
410.016	Reactor Coolant Leakage	11/16/92	03/18/93
410.017	Reg. Guide 1.45, Position C.9	11/16/92	03/18/93
410.018	Reg. Guide 1.45, Position C.8	11/16/92	03/18/93
410.019	Reg. Guide 1.45, Position C.7	11/16/92	03/18/93
410.020	Reg. Guide 1.45, Position C.6	11/16/92	03/18/93
410.021	PZR safety valve discharge	11/16/92	02/25/93
410.022	Reference for PZR relief discharge system	11/16/92	02/25/93
410.023	First stage ADS hydrostatic loads	11/16/92	03/18/93
410.024	Instrumentation for PZR relief discharge system	11/16/92	02/25/93
410.025	Reg. Guide 1.52	11/16/92	03/18/93
410.026	Flooding	11/16/92	02/25/93
410.027	Equipment requiring protection from flooding	11/16/92	03/18/93
410.028	Potential sources of flooding	11/16/92	03/18/93
410.029	External flooding	11/16/92	02/25/93
410.030	Maximun flood level	11/16/92	03/18/93
410.031	Flood protection	11/16/92	01/22/93
410.032	Equipment protection	11/16/92	02/25/93
410.033	Flood protection	11/16/92	03/18/93
410.034	Flood protection	11/16/92	03/18/93
410.035	Ground water seepage	11/16/92	02/25/93
410.036	Flood protection	11/16/92	02/25/93
410.037	PXS equipment location	11/16/92	03/18/93
410.038	External flooding	11/16/92	02/25/93
410.039	Design criteria	11/16/92	02/25/93
410.040	Multi-door passageways leakage prevention	11/16/92	03/18/93
410.041	Waterproofing	11/16/92	02/25/93
410.042	ICRWST lines	11/16/92	01/14/93
410.043	CCW layout	11/16/92	03/18/93
410.044	Flood hazards	11/16/92	03/18/93
410.045	Flooding monitors	11/16/92	02/25/93
410.046	Break protection from open cycle systems	11/16/92	03/18/93
410.047	Water tight doors	11/16/92	03/18/93
410.048	SFP cooling pumps & heat exchangers flood prot	11/16/92	03/18/93
410.049	Flood consequences	11/16/92	03/18/93
410.050	Flooding protection for remote shutdown panel	11/16/92	03/18/93
410.051	Equipment requiring missile protection	11/16/92	03/18/93
410.052	Turbine missiles	11/16/92	03/18/93
410.053	Secondary missiles	11/16/92	03/18/93
410.054	Equipment protection	11/16/92	03/18/93
410.055	Missiles from rotating equipment	11/16/92	02/09/93
410.056	Missile prevention	11/16/92	02/09/93
410.057	Missile generation from non-high-energy systems	11/16/92	02/09/93
410.058	Valve and bonnet design	11/16/92	02/25/93
410.059	Stored energy - nuts, bolts and studs	11/16/92	03/18/93
410.060	Hydrogen bottle explosion	11/16/92	02/25/93
410.061	Sources of missiles	11/16/92	03/18/93
410.062	Missile protection of remote shutdown panel	11/16/92	03/18/93
410.063	Equipment requiring missile protection	11/16/92	03/18/93
410.064	Credible missile sources	11/16/92	03/18/93
410.065	Analysis for RCP missile containment	11/16/92	01/22/93
410.066	Missiles inside containment	11/16/92	03/18/93
410.067	Source of missiles	11/16/92	03/18/93
410.068	Equipment requiring missile protection	11/16/92	03/18/93
410.069	Missile protection	11/16/92	03/18/93
410.070	Strike probability	11/16/92	03/18/93
410.071	Missile protection of remote shutdown panel	11/16/92	03/18/93
410.072	Protection from externally generated missiles	11/16/92	03/18/93
410.073	Protection of PCS from external missiles	11/16/92	03/18/93
410.074	Fuel storage pool	11/16/92	03/18/93
410.075	Missile protection of remote shutdown panel	11/16/92	03/18/93
410.076	Pipe break effects	11/16/92	03/18/93
410.077	Containment penetrations	11/16/92	02/09/93

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
410.078	Leak cracks in pipes	11/16/92	02/09/93
410.079	Section 3.6.1.1.J of SSAR	11/16/92	02/09/93
410.080	Turbine stop and FW control valves	11/16/92	03/18/93
410.081	Pipewhip restraints	11/16/92	03/18/93
410.082	Line restrictions	11/16/92	02/25/93
410.083	Protection from pipe breaks	11/16/92	03/18/93
410.084	Environmental/flooding effects	11/16/92	03/18/93
410.085	Subcompartments	11/16/92	02/09/93
410.086	Loads	11/16/92	03/18/93
410.087	RV asymetric loads	11/16/92	03/18/93
410.088	Control room/MSIV compartment separation	11/16/92	03/18/93
410.089	Pipe breaks	11/16/92	03/18/93
410.090	Protection for RC loop	11/16/92	02/09/93
410.091	Instrumentation protection	11/16/92	03/18/93
410.092	Pipe failure protection	11/16/92	02/09/93
410.093	Classification of Structures, Systems, Components	01/26/93	03/18/93
410.094	HVAC system drawings	01/26/93	03/18/93
420.008	ITAAC - safety monitoring system	11/16/92	02/09/93
460.008	Radwaste management system source terms	11/16/92	03/18/93
460.009	Liquid radwaste	11/16/92	03/18/93
460.010	Gaseous radwaste	11/16/92	03/18/93
460.011	Solid radwaste	11/16/92	03/18/93
460.012	Exhaust monitoring	11/16/92	02/09/93
460.013	Exhaust duct radiation monitoring	11/16/92	03/18/93
460.014	SG blowdown & CCW radiation monitors	11/16/92	03/18/93
460.015	Sampling provisions	11/16/92	03/18/93
460.016	Accident monitoring	11/16/92	03/18/93

Records printed: 90

Question 410.16

Section 5.2.5.1.2 of the SSAR states that limits for reactor coolant leakage are identified in the technical specifications. Section 3.4.7 of Chapter 16 of the SSAR (Technical Specifications) specifies the limits of identified and unidentified leakage. Describe how the identified leakage can be quantitatively measured, and how operators can determine if the leakage limit is exceeded.

Response:

The amount of identified leakage from the reactor coolant system can be determined by adding up the amounts from the valve stem leakoff collection drain lines. The leakage is detected by flow sensors in the drain lines, and the sensed flow is totaled and averaged over time by the plant instrumentation system. The plant instrumentation system provides an alarm in the main control room if the average leak rate for a given measurement period (less than one hour) exceeds 10 gpm for identified leakage.

In addition, reactor head seal leakage is monitored by a temperature detector on the seal leakoff line, which detects if the temperature in the leakoff line exceeds a predetermined setpoint. This high temperature is alarmed in the main control room, alerting operators to the possibility of a head seal leak.

Leakage from the reactor coolant system to the component cooling water system is detected by the component cooling water radiation monitor, by an increase in surge tank level, by high flow downstream of selected components, or by some combination of these measurements. The measurements are monitored by the plant instrumentation system. If any of the variables should exceed a predetermined setpoint, the plant instrumentation system provides an alarm in the main control room. The operators would then trend the component cooling water surge tank for a change in level to calculate a leakage rate. Pressurizer level monitoring along with reactor coolant system mass balance can be used as a check to confirm that the reactor coolant leakage is completely accounted for.

SSAR Revision: NONE

Question 410.17

Position C.9 of RG 1.45 states that the technical specifications should address the availability of various types of instruments for RCPB leakage to ensure adequate coverage at all times. Describe how the AP600 design will meet this regulatory position (Section 5.2.5).

Response:

SSAR Chapter 16, Technical Specification 3.4.9, defines the operability requirements for RCS leakage detection instrumentation. In addition, instrumentation used to identify reactor coolant pressure boundary leakage is designed so that its operability may be determined at all times. Should a detector fail (signal outside its calibrated range or self-monitored trouble detected), the plant instrumentation system will alarm in the main control room that the specific leak detection monitor readout is questionable. The alarm prompts the operators to observe other sensors providing leak detection information. Technical Specification 3.4.9 allows leakage to be averaged over 24 hours; therefore, operators have sufficient time to determine if small leaks are from the reactor coolant system and to take corrective action in an orderly manner.

SSAR Revision: NONE

Question 410.18

Position C.8 of RG 1.45 states that the leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation. Describe how the AP600 design will meet this regulatory position (Section 5.2.5).

Response:

The instrumentation for reactor coolant pressure boundary leak detection can be tested for operability during plant operation. The electronics for the containment atmosphere radioactivity monitors, containment sump level sensors, containment air cooler condensate flow monitor, and leakoff collection line flow monitors are located outside containment and are accessible for calibration during plant operation. The primary sensing elements for the level and flow monitors are calibrated during refueling outages.

SSAR Revision: NONE

Question 410.19

Position C.7 of RG 1.45 states that procedures for converting various leakage indication to a common leakage equivalent should be available to the operators. Describe how the AP600 design will implement this regulatory position (Section 5.2.5).

Response:

The plant instrumentation system is a microprocessor-based system, that accepts inputs from all reactor coolant pressure boundary leakage detection sensors and monitors. The plant instrumentation system accepts the sump level and drain line flow signals in the units in which they are measured and converts them into equivalent identified and unidentified leakage rates. The reactor coolant leakage rate data is then provided to the main control room operators.

SSAR Revision: NONE

Question 410.20

Position C.6 of RG 1.45 states that the leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. The airborne particulate radioactivity monitoring system should remain functional when subjected to the SSE. Describe how the AP600 design will meet this regulatory position (Section 5.2.5).

Response:

No credit is taken for immediate leak detection by the containment airborne particulate radioactivity monitor (See SSAR Subsection 5.2.5.3.1). Therefore, it is not necessary to qualify the monitor and the associated sample line to function subsequent to an SSE. Containment activity is monitored by the containment high-range radiation monitor, which is seismically qualified. See SSAR Subsection 11.5.2.3.2.

SSAR Revision: NONE

Question 410.23

Section 5.4.11.2 of the SSAR states that the discharge of water, steam, and gases from the first-stage automatic depressurization system valves when used to vent noncondensable gases does not result in pressure in excess of the in-containment refueling water storage tank (IRWST) design pressure.

When the high pressure discharge of steam and noncondensable gases, through piping system, injects into the in-containment water .torage tank, it will be condensed and mixed with the tank water. The IRWST, in this case, will function similarly to the suppression pool in the BWR plants. Provide an analysis to demonstrate that the hydrodynamic loads on the tank and piping have been adequately considered.

Response:

The analysis method for the hydrodynamic loads on the IRWST is described in SSAR Subsection 3.8.3.3.2. Dynamic loads on the IRWST due to the three stages of ADS operation are determined using the results from the ADS hydraulic test described in SSAR Section 1.5. This test, described in WCAP 13342, determines the dynamic effects on the test tank by simulating operation of the ADS and spargers. Loads on the IRWST boundary are calculated using the pressure source load obtained from the ADS test. Hydrodynamic finite element analyses are performed on both the test tank and the IRWST. The analyses on the test tank develop the source pressure load by comparison of pressure measurements at selected locations in the tank against results at these locations predicted by the analyses. The source pressure load is then used in the IRWST analysis to give the dynamic responses of the tank boundary (deflections, accelerations, and stresses).

IRWST pressurization is analyzed using a detailed WGOTHIC model of the IRWST. Mass and energy releases are taken from the NOTRUMP analyses. The resulting pressure time histories will be included in the structural assessment of the IRWST.

The hydrodynamic loads imposed by manual venting of non-condensable gases using the first stage ADS valves are bounded by the analysis performed during operation of the ADS for sequential operation of stages one to three which vent to the IRWST.

The first phase of the ADS tests has been completed and the test measurements are being reduced and documented. These results are being reviewed and compared against the hydrodynamic and pressurization analyses. These tests were conducted at volumetric flow rates approximately two times those predicted for the AP600 design. Preliminary review of these results show that the hydrodynamic pressures are within the range considered during the preliminary design of the tank. The hydrodynamic analyses of the test tank and of the IRWST, which were performed prior to availability of the test results, are now being being updated based on the test results.

The ADS tests have included strain measurements on the ADS sparger. These test results will be used to verify sparger loads. Piping loads are to be measured during the Phase B tests.

The results of the hydrodynamic and pressurization analyses, based on the first phase ADS tests, will be submitted to NRC by October 31, 1993.

SSAR Revision: None

Question 410.25

Section 9.4.1 of the SSAR states that a supplemental air filtration subsystem filters outside makeup air and pressurizes the MCR and TSC areas if high airborne radioactivity is detected in the MCR supply air duct and/or receipt of a containment isolation signal. Also, Note 2 in Figure 9.4.1-1 (sheet 1 of 6) of the SSAR states that the supplemental air filtration system may be deleted pending radiological analysis by Westinghouse and is shown for information only. Additionally, the referenced figure shows only one fresh air intake and it is not clear that any radiation monitors are provided.

Is Westinghouse going to delete the supplemental air filtration system? If so, describe in detail the justifications for doing so. To meet GDC 19, Westinghouse must demonstrate that the air filtration system is a safety-related ESF filtration system, and that it conforms with the guidelines provided in Sections 6.4, 6.5.1, and 9.4.1 of the SRP, Regulatory Guide 1.52, ASTM Standard D3803-1989, and ASME Standards N509-1989, N510-1989, and AG-1-1991, which includes appropriate provisions for single-failure criterion design, dual fresh air intakes, redundant radiation monitoring in each fresh air intake and associated testing requirements of the safety-grade components. Also, provide an updated flow diagram and general arrangement drawings demonstrating conformance with the guidance of the referenced SRP sections, regulatory guide, and standards.

Response:

The licensing design basis relating to the habitability of the main control room utilizes the main control room habitability system (VES) and is consistent with passive safety system design criteria and the intent of regulatory guidance and requirements. The AP600 design described in SSAR Section 6.4 provides a passive method of complying with General Design Criterion 19. The system design does not comply with the specific details of regulatory guidance or standards since that guidance was developed to address plant designs based on active safety systems.

The main control room dose analysis provided in SSAR Subsection 15.6.5 demonstrates that radiation exposure of MCR personnel during the postulated accidents does not exceed the limits set by 10 CFR 50, Appendix A, GDC 19. The design of the VES complies with the intent of Regulatory Guide 1.52, including tolerance for limiting single failures.

The nuclear island nonradioactive ventilation system (VBS) provides a capability similar to that of the ESF systems found in current plants, with respect to air filtration and adsorption capability. The VBS is not safety related and is not fully redundant (for example, a single air intake). The system does incorporate appropriate redundancy for powered components and filter units in order to maximize system availability consistent with defense-in-depth systems. The system is classified as a class D system; the additional requirements on procurement, inspection, and monitoring are described in SSAR Subsection 3.2.2.6. Further, the VBS system design basis is to limit MCR personnel doses assuming VBS operation under design basis source term conditions to the limits established under GDC 19.

The supplemental air filtration will not be eliminated from the nuclear island nonradioactive ventilation system, and the note on the P&ID will be deleted in a revision to the figure.

The redundant radiation monitors provided to detect elevated airborne radioactivity levels to the main control room are located in the VBS just inside the main control room envelope and are shown in Figure 9.4.1-1, Sheet 6 of 6, of the SSAR.

The SSAR will be revised as follows:

SSAR Revision:

SSAR Figure 9.4.1-1, Sheet 1, will be revised to delete note 4.

Question 410.27

Identify all safety-related equipment and equipment important-to-safety (i.e. non-safety related equipment whose failure could adversely affect the ability of safety related equipment to perform its safety function) requiring protection from internal and external flooding (Section 3.4.1).

Response:

AP600 uses safety-related equipment which functions during or following a design basis event to provide:

- integrity of the reactor coolant pressure boundary,
- capability to shut down the reactor and maintain it in a safe shutdown condition,
- capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

The Plant safety-related equipment are located on the nuclear island (containment building, shield building, auxiliary building) which are Seismic Category I structures designed to withstand the effects of external flooding. Equipment in these buildings are protected from the effects of external flooding.

The design basis for protection from internal floods is that for postulated flood events, safe shutdown can be achieved considering both the consequences of the flood and a single failure. Table I provides a listing of the plant active safety-related equipment requiring protection from internal flooding. Safety-related I&C equipment flooding protection is addressed in Q435.56. The table identifies the component's safety function, based on the criteria defined above, and the method applied for equipment flood protection. In the event of component flooding, one of the following criteria is applied for protection of equipment:

- Equipment will be qualified for submergence due to flooding.
- Equipment will be evaluated to show that failure of the equipment due to flooding is acceptable since its safety-related function is not required or has otherwise been accomplished.

Safety-related equipment located above the flooding level have been excluded from Table I. In addition, Table I does not include safety-related passive equipment (e.g., pipes, manual valves, tanks).

SSAR Subsections 3.4.1.2.2.1 and 3.4.1.2.2.2 provide additional detail on the impact of flooding on the active safe shutdown components.

SSAR Revision: NONE

TABLE I

AP600 Safety-Related Equipment Requiring Flood Protection

Tag Number	Component Description	Safety (1) Function	Plant Location	Protection (2) Method
CHEMICAL AND VOLUME CONTROL SYSTEM				
CVS PL V042	VALVE	CI	Containment	Note 2
CVS PL V045	VALVE	CI	Containment	Note 2
CVS PL V056	VALVE	CI	Containment	Note 2
PASSIVE CORE COOLING SYSTEM				
PXS PL V014A	VALVE	RCPB, SSD	Containment	Note 4
PXS PL V014B	VALVE	RCPB, SSD	Containment	Note 4
PXS PL V015A	VALVE	RCPB, SSD	Containment	Note 4
PXS PL V015B	VALVE	RCPB, SSD	Containment	Note 4
PXS PL V016A	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V016B	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V017A	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V017B	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V027A	VALVE	SSD	Containment	Note 2
PXS PL V027B	VALVE	SSD	Containment	Note 2
PXS PL V028A	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V028B	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V029A	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V029B	VALVE	RCPB, SSD	Containment	Note 1
PXS PL V117A	VALVE	SSD	Containment	Note 4
PXS PL V117B	VALVE	SSD	Containment	Note 4
PXS PL V118A	VALVE	SSD	Containment	Note 4
PXS PL V118B	VALVE	SSD	Containment	Note 4
PXS PL V119A	VALVE	SSD	Containment	Note 1
PXS PL V119B	VALVE	SSD	Containment	Note 1
PXS PL V120A	VALVE	SSD	Containment	Note 1
PXS PL V120B	VALVE	SSD	Containment	Note 1
PXS PL V121A	VALVE	SSD	Containment	Note 3
PXS PL V121B	VALVE	SSD	Containment	Note 3
PXS PL V122A	VALVE	SSD	Containment	Note 1
PXS PL V122B	VALVE	SSD	Containment	Note 1
PXS PL V123A	VALVE	SSD	Containment	Note 1
PXS PL V123B	VALVE	SSD	Containment	Note 1

TABLE I (Cont'd)

AP600 Safety-Related Equipment Requiring Flood Protection

Safety (1) Function	Tag Number	Component Description	Plant Location	Protection (2) Method
PASSIVE CORE COOLING SYSTEM (Cont'd)				
PXS PL V124A	VALVE	SSD	Containment	Note 1
PXS PL V124B	VALVE	SSD	Containment	Note 1
PXS PL V125A	VALVE	SSD	Containment	Note 1
PXS PL V125B	VALVE	SSD	Containment	Note 1
NORMAL RESIDUAL HEAT REMOVAL SYSTEM				
RNS PL V001A	VALVE	RCPB	Containment	Note 2
RNS PL V001B	VALVE	RCPB	Containment	Note 2
RNS PL V002A	VALVE	RCPB, CI	Containment	Note 2
RNS PL V002B	VALVE	RCPB, CI	Containment	Note 2
RNS PL V013	VALVE	CI	Containment	Note 1
RNS PL V015A	VALVE	RCPB	Containment	Note 1
RNS PL V015B	VALVE	RCPB	Containment	Note 1
RNS PL V017A	VALVE	RCPB	Containment	Note 1
RNS PL V017B	VALVE	RCPB	Containment	Note 1
RNS PL V021	VALVE	CI	Containment	Note 2
RNS PL V023	VALVE	CI	Containment	Note 2
SPENT FUEL PIT COOLING SYSTEM				
SFS PL V034	VALVE	CI	Containment	Note 2
SFS PL V048	VALVE	CI	Containment	Note 1

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TABLE I (Cont'd)

AP600 Safety-Related Equipment Requiring Flood Protection

NOTES to Table I:

(1) SAFETY FUNCTIONS:
CI = Containment Isolation
RCPB = Reactor Coolant Pressure Boundary
SSD = Safe Shutdown

(2) FLOOD PROTECTION METHOD:

Note 1: Equipment is designed for submergence or is normally submerged.

Note 2: This is a normally closed valve which is not required to operate during safe shutdown operation. The valve is required to remain closed following component flooding.

Note 3: Valve is normally open and does not have to be repositioned during a safe shutdown and a coincident flooding event. The valve is required to remain open following component flooding.

Note 4: Redundant equipment located in separate compartments is provided. In the unlikely event that one of the two compartments is flooded, the redundant equipment located in the unflooded compartment is available to ensure safe shutdown capability with a single failure.

Question 410.28

Identify potential sources of internal flooding on a floor-by-floor basis in all buildings containing safety-related equipment. How will safety related equipment and equipment important-to-safety be protected from flooding from these sources (Section 3.4.1)?

Response:

The unique compartmentalization of the AP600 plant and the physical separation and isolation of the compartments and equipment areas from each other precludes flooding in one compartment or area from propagating to an adjoining compartment or area. Because of these plant arrangement features, the flooding evaluation is based on the specific boundaries of the compartments or areas rather than on a floor-by-floor identification of potential sources of internal flooding for a specific building.

SSAR Subsection 3.4.1 provides an evaluation of the flooding events in both the containment building and the auxiliary building and the consequences of compartment or area flooding for various component or piping failures. The internal flooding analysis shows that systems, structures, and components are not prevented from performing their required safe shutdown functions because of the effects of each postulated failure.

Subsection 3.4.1.2.2.1, Containment Flooding Events, addresses the impact of flooding on the safe shutdown systems and components inside containment. This subsection identifies the safe shutdown components below the maximum flood level (elevation 108'-2") and specifies the subcompartments containing these components.

The AP600 containment contains four major, physically separated, floodable subcompartments that are partially or completely below the maximum flood level. These four subcompartments are the RCS compartment, the PXS-A compartment, the PXS-B compartment, and the CVCS compartment.

The RCS compartment consists of the reactor vessel cavity, two steam generator compartments, and an interconnecting vertical access tunnel. The RCS compartment therefore represents one large, floodable volume that is designed to be flooded up to an elevation of 108'-2".

Any leakage occurring within containment drains by gravity to the containment sump, which is in the RCS compartment. The maximum level that could occur in the RCS compartment from all the water available in containment is elevation 108'-2". Since the RCS compartment contains no safe shutdown components below the maximum flood-up level, the flooding of this compartment has no impact on safe shutdown capability. Reverse flow from the RCS compartment to the two PXS compartments and the CVCS compartment, via the floor drains from those compartments, is prevented by redundant backflow preventers.

The PXS-A, PXS-B, and CVCS compartments are physically separated and isolated from each other by structural walls so that flooding in any one of these compartments or in the RCS compartment cannot cause flooding in any of the other compartments. The only means for flooding to occur in one of these three compartments is for a component or pipe failure to occur within that specific compartment. In the unlikely event that such a failure did

occur and result in the flooding of one of these compartments, the compartment would eventually flood up to the maximum flood level at elevation 108'-2" and then overflow to the maintenance floor level at elevation 107'-2" which drains to the RCS compartment.

The PXS-A and PXS-B compartments primarily contain components associated with the PXS. The safe shutdown-related components of the PXS located in these two compartments are redundant and essentially identical. In the unlikely event that one of the two PXS compartments is flooded because of a failure within that compartment, the PXS is not prevented from performing its safe shutdown function.

Subsection 3.4.1.2.2.2, Auxiliary Building Flooding Events, addresses the impact of flooding on the safety equipment in the auxiliary building. The AP600 auxiliary building contains radiologically controlled areas (RCA) and nonradiologically controlled areas, which are separated by 2- and 3-foot-thick structural walls and floor slabs. These structural barriers are designed to prevent flooding propagation across the boundary between these areas.

The nonradiologically controlled areas of the auxiliary building are designed to provide maximum separation between the mechanical and electrical equipment areas. This separation prevents the leakage from spreading from the piping areas and the mechanical equipment areas to the Class 1E electrical and Class 1E I&C equipment rooms. Any leakage from postulated pipe or component failures in the mechanical equipment compartments will drain from the auxiliary building to the turbine building.

The Class 1E and non-Class 1E electrical equipment areas are on floor levels 1, 2, 3, and 4 in the nonradiologically controlled area of the auxiliary building. The primary line of defense against flooding in the electrical equipment areas is to exclude fluid systems and their associated piping from these areas. The only fluid systems piping in the general electrical equipment areas is the piping associated with the demineralized water and fire protection systems.

There is no fluid system piping in the specific rooms containing Class 1E electrical equipment. The potential for flooding in the electrical equipment area is limited to the fire-fighting activities. The seismically qualified fire protection piping is routed only in the corridors and adjacent to the stairwells on levels 1, 2, 3, and 4.

There is a limited supply of water (approximately 18,000 gallons) provided to the fire protection system standpipe fire hose stations adjacent to each stairwell in the electrical equipment areas. Fire-fighting activities on levels 1, 2, 3, or 4 would contribute to flooding on level 1. The drain lines, stairwells, and elevator shaft would direct the water from the fire-fighting area to the sump pumps on elevation 66'-6" (level 1). The maximum water depth on level 1 would have no impact on the Class 1E batteries performing their safe shutdown function.

The demineralized water system piping in the electrical equipment area is in the corridors adjacent to the battery rooms on levels 1 and 2 only. The maximum nominal diameter of the demineralized water system piping is 1 inch; therefore, it is not considered as a source of flooding.

The safe shutdown components in the radiologically controlled areas are primarily containment isolation valves, which are located near the containment vessel and above elevation 82'-6". These containment isolation valves are located above the maximum flood level for this area. Flooding in the RCA would have no impact on the AP600 safe shutdown systems from performing their functions.

SSAR Revision: NONE

Question 410.30

Describe if the maximum flood level in Table 2.0-1 of the SSAR takes into account probable maximum floods (PMFs) generated by a combination of probable maximum precipitation (PMP) or other combinations of less severe environmental and man-made events along with seismic and wind effects.

Response:

The maximum flood level in Table 2.0-1 of the SSAR takes into account probable maximum floods (PMFs) generated by a combination of probable maximum precipitation (PMP) or other combinations of less severe environmental and man-made events along with seismic and wind effects. As described in SSAR Section 2.4, the combined license applicant will evaluate all events leading to potential flooding to demonstrate that the site meets the interface requirements. Events to be considered are those identified in Standard Review Plan Section 2.4.2.

SSAR Revision: NONE

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Question 410.33

Discuss the ability of safety-related equipment to perform its safety function while fully flooded, partially flooded, or wet (e.g. from spray). Particular attention should be given to the five containment isolation valves that are below the internal flood level, and are, therefore, subject to flooding (Section 3.4.1).

Response:

See the response to Q410.5.

SSAR Revision: NONE

Question 410.34

Section 3.4.1.2.2.1 of the SSAR states that the PXS-A, PXS-B, and CVCS compartments are physically separated and isolated from each other by structural walls such that flooding in any of these compartments or in the RCS compartment cannot cause flooding in any of the other compartments. This appears to contradict another statement in this section which says that, because the floor drains for these compartments are routed to the containment sump, flooding in one compartment could cause flooding in another compartment. The staff recognizes that "backflow preventers" are located in each line to prevent reverse flow into other compartments but insufficient detail has been provided on the design and operations of these components (see Q410.26). Clarify these statements.

Response:

See the response to Q410.6.

SSAR Revision: NONE.

Question 410.37

Section 3.4.1.2.2.1 of the SSAR states that the safe shutdown components located in PXS-A and PXS-B are redundant and "essentially identical." Clarify what is meant by "essentially identical."

Response:

See the response to Q410.8.

SSAR Revision: NONE

Question 410.40

Are any external or internal doorways or passageways too large to close with a single door? If so, how will leakage be prevented (Section 3.4.1)?

Response:

Some of the doorways between the safety-related auxiliary building and the adjacent turbine, annex I, annex II, and radwaste buildings are double doors. These doorways are located at or above grade elevation, and they are not watertight. As described in SSAR Subsection 3.4.1, the probable maximum external flood elevation is less than the plant grade elevation. In addition, the design of the buildings is such that water from internal flooding in areas adjacent to the auxiliary building is directed away from or prevented from entering the auxiliary building.

SSAR Revision: NONE

Question 410.43

Identify the component cooling water on the building layout drawings (Section 3.4.1).

Response:

See the response to Q410.12.

SSAR Revision: NONE

Question 410.44

Discuss possible flood hazards resulting from below grade tunnels between buildings (Section 3.4.1).

Response:

Safety-related equipment is located exclusively in the containment and auxiliary buildings. These buildings have a common basemat, and there are no below-grade tunnels between these buildings and any other buildings. Thus, there are no potential flooding hazards that could affect safety-related equipment associated with below-grade tunnels on the AP600.

SSAR Revision: NONE

Question 410.46

Do any open-cycle systems enter any buildings housing safety-related equipment and equipment important-to-safety? If so, how will this equipment be protected from the effects of a break in that part of the open-cycle system within th: building (Section 3.4.1)?

Response:

The AP600 relies only on safety-related equipment for safe shutdown following a flooding event. Safety-related equipment is located exclusively in the containment and auxiliary buildings.

The fire protection system and the demineralized water transfer and storage system are open-cycle systems that enter the containment. During plant operation, the containment piping for these systems is isolated by containment isolation valves and is not a potential flooding source.

As described in SSAR Subsection 3.4.1.2.2.2, the nonradiologically controlled area of the auxiliary building is designed to provide maximum separation between the mechanical and electrical equipment areas. The fire protection system is an open-cycle system that enters the mechanical equipment area. As described in SSAR Subsection 3.4.1.2.2.2, leakage from fire protection piping in the mechanical equipment area drains to the turbine building or the annex 1 building without affecting safe shutdown capability. The fire protection system (supplied from the seismic water source) and the demineralized water transfer and storage system are open-cycle systems that enter the electrical equipment area. The maximum nominal diameter of the demineralized water piping is 1 inch; therefore, it is not considered as a source of flooding. As described in SSAR Subsection 3.4.1.2.2.2, the design of the fire protection system limits the volume of water available to cause flooding in the electrical equipment area to an amount that does not affect safe shutdown capability.

The fire protection system, the demineralized water transfer and storage system, and the chemical and volume control system are open-cycle systems that enter the radiologically controlled area of the auxiliary building. Of these three systems, the fire protection system has the largest system volume. Severe flooding can be tolerated in this area because there is no equipment required for safe shutdown on the lowest elevation. Safety-related valves in the radiologically controlled area are located above elevation 82'-6". If a leak from a moderate-energy crack in the fire protection piping remained undetected for several days and if the water in both fire water storage tanks were emptied into the building, flooding would not reach the safety-related valves.

SSAR Revision: NONE

Question 410.47

Identify all watertight doors and hatches on the general arrangement drawings, not just 3-hour fire doors (Section 3.4.1).

Response:

There are no watertight doors or hatches in the AP600 plant. Inside containment four maintenance hatches, located on the elevation 107'-2" floor level, provide access to the equipment in the CVS compartment. These maintenance hatches are designed to prevent gross in-leakage into the CVS compartment from water on the elevation 107'-2" floor.

SSAR Revision: NONE

Question 410.48

It appears from Figure 1.2-5 of the SSAR that both divisions of the spent fuel pool cooling pumps and heat exchangers are susceptible to flooding. How will these components be protected?

Response:

As described in SSAR Subsection 9.1.3, the spent fuel pool cooling system (SFS) serves no safety-related function; therefore, this system contains no safety-related equipment other than containment isolation valves. Since this system serves no safety-related function, the SFS pumps and heat exchangers are not required to be protected from flooding (The plant arrangement and the location of the equipment, reduce the susceptibility for flooding this equipment).

In the event of a postulated pipe break in the vicinity of the SFS equipment, the water released from that break would drain to elevation 66'-6" floor level via several pathways, which include floor drains, vertical pipe chases, the stairwell, and the elevator shaft.

SSAR Revision: NONE

Question 410.49

Identify the potential flooding consequences if the IRWST or the PRHR heat exchangers were to fail. Identify the protective features used to protect safety-related equipment and equipment important-to-safety from the resulting flooding (Section 3.4.1).

Response:

If a leak were to occur in the IRWST, the leaking water would flow to the elevation 107'-2" maintenance floor and into the RCS compartment. The containment sump level monitoring system would detect the leakage and alert the operator to the need for corrective action. If the leak could not be readily repaired, the RCS compartment would begin to flood. The consequences of such flooding are described in SSAR Subsection 3.4.1.2.2.1. The RCS compartment is designed to be flooded as discussed in SSAR Section 3.4.

The consequences of a PRHR heat exchanger tube failure are described in SSAR Subsection 6.3.3.3.3. IRWST temperature instrumentation or other available instrumentation would detect the leaking tube and alert the operator to the need for corrective action. If the leak were not isolated, the gradual addition of reactor coolant to the IRWST would cause the tank to overflow into the refueling cavity. The refueling cavity contains no safe shutdown components. The refueling cavity would eventually fill and overflow to the RCS compartment as in the preceding case.

SSAR Revision: NONE

Question 410.50

How will the remote shutdown panel be protected from external and internal flooding (Section 3.4.1)?

Response:

The remote shutdown workstation is located in the nonradiologically controlled area of the auxiliary building, at grade elevation (100'-0"). As described in SSAR Subsection 3.4.1.1.1, seismic Category I structures such as the auxiliary building are protected from external flooding by site grading and waterproofing of building surfaces below grade.

As described in SSAR Subsection 3.4.1.2.2.2, the sources of internal flooding in the portion of the auxiliary building containing the remote shutdown workstation is limited to fire-fighting activities or failure of the fire protection system piping. The fire protection system piping is located outside the remote shutdown workstation room, and the system design limits the volume of water available to flood this area. Fire protection water flows from elevation 100'-0" to level 1 via floor drains, stairwells, and the elevator shaft. These drain paths protect the remote shutdown workstation from any flooding effects of fire protection.

SSAR Revision: NONE

Question 410.51

Identify all safety-related equipment and equipment important-to-safety (i.e. non-safety related equipment whose failure could adversely affect the ability of safety-related equipment to perform its safety function) that require protection from internally-generated missiles (outside containment) (Section 3.5.1.1).

Response:

There is no safety-related equipment which requires protection from internally generated missiles (outside containment), since the AP600 Plant design has no credible missile sources. See the response to Q410.61.

SSAR Revision: NONE

Question 410.52

How will safety-related equipment and equipment important-to-safety be protected from turbine missiles (Section 3.5.1.1)?

Response:

Protection against turbine missiles is described in SSAR Subsection 3.5.1.3. Protection is provided by the orientation of the turbine-generator and by the use of fully integral low-pressure turbine rotors. Analyses of the probability of the generation of missiles have been submitted to the NRC staff. The report for rotors with shrunk-on discs was approved by the NRC staff (see Reference 2 of SSAR Subsection 3.5.1.3). The methodology for fully integral rotors was submitted in Reference 1 of Subsection 3.5.1.3. Preliminary staff review (see Reference 3 of SSAR Subsection 3.5.1.3) agreed that the fully integral low-pressure rotors may be less susceptible to stress corrosion cracking than the shrunk-on discs. In the meeting on November 5, 1992 between the NRC staff, EPRI, and turbine vendors, it was concluded that the turbine failures were not a safety issue and that the inspections recommended by the turbine vendors to ensure availability and reliability were more than sufficient to ensure an acceptably low probability of missile generation.

SSAR Subsections 3.5.1.3, 3.5.4, 3.5.5, and 10.2.3.6 and Table 1.8.1 (Sheet 2 of 8) will be revised as follows.

SSAR Revision:

3.5.1.3 Turbine Missiles

The turbine generator is located north of the nuclear island with its shaft oriented north-south. In this orientation, the potential for damage from turbine missiles is negligible. Safety-related structures, systems and components are located outside the high-velocity, low-trajectory missile strike zone, as defined by Regulatory Guide 1.115. Thus, postulated low-trajectory missiles cannot directly strike safety-related areas. Credible high-trajectory missiles that could reach safety-related areas do not have sufficient kinetic energy to penetrate the tornado missile resistant concrete and steel structures housing safety-related equipment. -

The turbine and disc design is described in Section 10.2. ~~The turbine inspection program to be implemented on AP600 will use a probabilistic approach for scheduling the inspection of the fully integral low-pressure turbine rotors. The turbine inspection program is based on the Westinghouse report WSTG-4-P (References 1 and 2).~~

3.5.4 Missile Protection Interface Requirements

The combined license applicant must demonstrate that the site satisfies the interface requirements provided in Section 2.2. This requires an evaluation for those external events that produce missiles that are more energetic than the tornado missiles postulated for design of the AP600, or additional analyses of the AP600 capability to handle the specific hazard.

~~The applicant must implement a turbine inspection program satisfying the requirements of References 1 and 2.~~

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~~3.5.5 References~~

1. ~~WSTG-4-P, Proprietary and WSTG-4-NP, Non-Proprietary, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," October 1984.~~

2. ~~NRC Safety Evaluation Report, letter from B. D. Liaw to J. A. Martin dated December 27, 1984.~~

10.2.3.6 Inservice Inspection

Inspections for the turbine assembly and valves are performed in accordance with the turbine vendor's recommendations. They include the following:

- Disassembly of the turbine is conducted during plant shutdown. Inspection of parts that are normally inaccessible when the turbine is assembled for operation (couplings, coupling bolts, turbine rotors, and low-pressure turbine blades) is conducted. ~~The turbine is inspected in sections so that over a 10-year period, the entire turbine is inspected once.~~

  This inspection consists of visual, surface, and volumetric examinations as indicated below:

  1. Each rotor is inspected visually and by magnetic particle testing on accessible surfaces. Ultrasonic inspection of the side-entry blade grooves is conducted. This inspection is conducted at intervals of about 6 years.

  2. A 100 percent surface examination of couplings and coupling bolts is performed.

Westinghouse

Table 1.8-1 (Sheet 2 of 8)

Summary of AP600 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Sub-section
2.10	Flood and ground water elevations	Site Interface	Site specific parameters	2.4
2.11	Hydrostatic loads on systems, components, and structures	Site Interface	Site specific parameters	2.4
2.12	Seismic parameters peak ground acceleration response spectra shear wave velocity	Site Interface	Site specific parameters	 2.5 2.5 2.5
2.13	Required bearing capacity of foundation materials	Site Interface	Site specific parameters	2.5
~~3.1~~	~~Turbine inspection requirements~~	~~Requirement of AP600~~	~~Combined License applicant program~~	~~3.5.4~~
3.2	Operating procedures to minimize water hammer	Requirement of AP600	Combined License applicant procedure	3.6, 10
3.3	Site seismic sensor location and "trigger" value	Requirement of AP600	On site implementation	3.7.4
3.4	Depth of overburden	Requirement of AP600	On site implementation	3.8
3.5	Depth of embedment	Requirement of AP600	On site implementation	3.8
3.6	Specific depth of waterproofing	Requirement of AP600	On site implementation	3.8.5

Question 410.53

How will safety-related equipment and equipment important-to-safety outside containment be protected from credible secondary missiles (e.g., concrete fragments) (Section 3.5.1.1)?

Response:

The AP600 uses only safety-related systems and equipment to establish and maintain safe shutdown conditions. There is no equipment important to safety (as defined in Q410.27) that requires protection from missiles. Safety-related systems and components are located within the seismic Category I nuclear island structures, as described in SSAR Subsection 3.5.2, and are protected from missiles, as described in SSAR Subsection 3.5.3. As stated therein, secondary missiles such as concrete fragments are considered. The consequences of scabbing are evaluated if the thickness is less than the minimum thickness to preclude scabbing. No credible internally generated missiles (outside containment) have been identified for which scabbing of the concrete would be predicted.

SSAR Revision: NONE

Question 410.54

Identify safety-related equipment and equipment important-to-safety that are subject to missiles from non-seismic Category 1 structures, systems, and components, and discuss how this equipment will be protected from such missiles (Section 3.5.1.1).

Response:

SSAR Section 7.4 identifies safety-related equipment located outside containment. As discussed in Section 7.4, the AP600 design uses only safety-related systems to establish and maintain safe shutdown conditions. Non-safety-related systems are not required for safe shutdown of the plant. There is no equipment important to safety (as defined in Q410.27, that is, non-safety-related equipment whose failure could adversely affect the ability of safety-related equipment to perform its safety function) requiring protection from missiles generated from non-seismic Category 1 structures, systems, and components. SSAR Subsection 3.7.3.13 discusses the methods of protecting safety-related structures, systems, and components from adverse interaction of nonseismic structures, systems, and components. As discussed in Subsection 3.7.3.13.1, the general plant arrangement provides physical separation between the safety-related equipment and nonseismic structures, systems, and components to the maximum extent practicable. Any nonseismic component identified as a source to specific safety-related equipment during the design process is evaluated in accordance with the criteria and guidelines specified in Subsections 3.7.3.13.1, 3.7.3.13.2, and 3.7.3.13.3; and appropriate protection is provided.

SSAR Revision: NONE

Question 410.59

Provide sample analyses to demonstrate that the stored energy for nuts, bolts, nut and bolt, and nut and stud combinations is not enough to generate credible missiles (Section 3.5.1.1).

Response:

Bolts and nuts do not become missiles unless they break. Installed bolts clamping two or more members together, (for example, flange to flange) are under static tension. They will not break in normal service. The level of tensile stress may increase because of externally applied force, for example, due to increased internal pressure of piping or pressure vessels. Because of differences in stiffness between the bolts and the members under clamping, however, the increase in bolt force is only a fraction of the external tension load applied on the bolt. Also, the increase in bolt stresses is self-limiting in flange joints because the stretch in the bolt would result in leaks at flange faces, relieving the internal pressure. This is why bolts and nuts installed in static systems such as piping or pressure vessels seldom break.

Installed bolts (and studs) under preload can break in tension if the external force continues to rise. Most structural bolts deform before breaking. Low- to medium-strength bolts (such as A36 and B7 bolts) have a low potential of becoming missiles because the stored energy available at fracture is relatively low ($<$ 200 lbf-in/in^3). During tensile tests, low- to medium-strength bolt specimens travel only a short distance upon failure.

To demonstrate that the stored energy for nuts, bolts, and combinations is not sufficient to generate a credible missile, a sample analysis was performed. A 1/2-inch-diameter B7 bolt and nut assembly was assumed in this analysis, and the broken fragment was assumed to be two times the diameter of the bolt in length. It was assumed that the nut weighed the same as the broken fragment. The velocity of the fractured bolt can be estimated by converting the stored strain energy into kinetic energy. The stored energy in a B7 bolt is about 184-lbf-in./in.^{3}. This stored energy would provide a maximum instantaneous velocity at fracture of 30 ft/sec, which would propel the bolt fragment upwards to a height of about 14 feet. If the bolt fragment were propelled in the downward direction, its velocity would continue to increase with distance. After 10 feet of travel, the velocity will reach approximately 40 ft/sec. A bolt fragment of this weight (0.11 lb) traveling at this velocity cannot credibly cause damage to even the smallest of components. In addition these numbers are very conservative because losses in the strain energy due to bolt deformation before the fracture and friction losses were not taken into account.

Bolts (and studs) can break while under stress because of metal fatigue and stress corrosion cracking. These failure mechanisms involve a crack initiation, growth, and final fracture. During the crack growth stage, the initial stresses in the bolt will be relieved. Thus any fragments of bolts as a result of failures due to fatigue and stress corrosion cracking do not become missiles because the initial pre-tension is relieved while fatigue cracks and stress corrosion cracks grow. High-strength, low-alloy steel bolting materials with yield strength greater than 180 ksi may be an exception. They can break because of stress corrosion cracking or hydrogen embrittlement cracking under pre-tension with a stored energy (600 lbf-in./in.^{3}) which is considered low compared with other large missiles. However, these bolting materials are not used either in equipment or in the structural members on the AP600.

SSAR Revision: NONE

Question 410.61

Identify the sources of missiles that meet the criteria in Section 3.5.1.1.2.3 of the SSAR.

Response:

The criteria in SSAR Subsection 3.5.1.1.2.3 refer to potential missiles outside containment. The only sources included in these categories for which impacts on safety-related structures, systems, or components may be considered are a limited number of non-safety-related pumps and fans inside the auxiliary building. The safety-related systems and components needed to bring the plant to a safe shutdown are located inside the containment shield building and auxiliary building, both of which have thick, structural concrete exterior walls that provide protection from missiles generated in other portions of the plant. These walls are evaluated for tornado missiles that bound the energy available in internally generated missiles. Potential missiles from the turbine-generator are discussed in SSAR Subsection 3.5.1.3 and the response to Q410.52.

The pressurized components in the high-energy portions of the high-energy systems inside the auxiliary building are constructed to the ASME Code, Section III. The high-pressure gas storage cylinders inside the auxiliary building are the air storage bottles for the main control room habitability system (VES). The air storage bottles and attached piping and valves are constructed to the ASME Code, Section III, requirements and are supported and designed for seismic loads and are therefore not credible sources of missiles.

The rotating equipment in the auxiliary building is eliminated as a missile source for one or more of the following reasons. Equipment in use less than 2 percent of the time is not considered a missile source. This includes pumps that operate less than 2 percent of the time and motors for valve operators and mechanical handling equipment and pumps. Pumps and fans in compartments surrounded by structural concrete walls with no safety-related systems or components are not considered missile sources. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. Non-safety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features have design requirements for a housing or an enclosure to ·tain fragments from postulated failures of rotating elements.

SSAR Subsection 3.5.1.1.2.4 will be added as follows.

SSAR Revision:

3.5.1.1.2.4 Credible Sources of Internally Generated Missiles (Outside Containment)

The consideration of missile sources outside containment that can adversely affect safety-related structures, systems, or components is limited to a few rotating components inside the auxiliary building. The safety-related systems and components needed to bring the plant to a safe shutdown are located inside the containment shield building and auxiliary building, both of which have thick structural concrete exterior walls that provide protection from missiles

generated in other portions of the plant. Protection against potential missiles from the turbine-generator is discussed in SSAR Subsection 3.5.1.3.

Rotating components inside the auxiliary building that are either safety related or are constructed as canned motor pumps would contain fragments from a postulated fracture of the rotating elements and are excluded from evaluation as missile sources. Rotating components used less than 2 percent of the time are also excluded from evaluation as missile sources. Non-safety-related rotating equipment in compartments surrounded by structural concrete walls with no safety-related systems or components inside the compartment is not considered a missile source. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. For one or more of these reasons the non-safety-related rotating equipment inside the auxiliary building is considered not to be a credible missile source. Non-safety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features have design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

The high-energy systems inside the auxiliary building do not include any pressurized components in the high-energy portions that are constructed to standards other than the ASME Code, Section III. See Table 3.6-1 for a list of the high-energy systems. The outlet pipes and valves for the air storage bottles for the main control room are constructed to the ASME Code, Section III, requirements and are designed for seismic loads. The attached pipes and valve are not credible missile sources due to an accidental impact.

Question 410.62

How will the remote shutdown panel be protected from missiles generated outside containment (Section 3.5.1.1)?

Response:

The remote shutdown workstation is located in the nonradiologically controlled area of the auxiliary building at grade elevation (100'-0") in its own compartment. It is separated from rotating equipment or pressurized components by the compartment walls and auxiliary building outside wall. SSAR Subsection 3.7.3.13 discusses the potential missiles due to failures of nonseismic items. Should a nonseismic component be capable of being dislodged from its supports in the compartment, the component will be separated, segregated, or supported as seismic Category II. There is no credible missile threat to the remote shutdown workstation.

SSAR Revision: NONE

Question 410.63

Identify all safety-related equipment and equipment important-to-safety which require protection from internally-generated missiles (inside containment) (Section 3.5.1.2).

Response:

There is no safety-related equipment which requires protection from internally generated missiles (inside containment), since the AP600 Plant design has no credible missile sources. See the responses to Q410.64 and Q410.67.

SSAR Revision: NONE

Question 410.64

Identify all sources of primary and credible secondary missiles (concrete fragments) that could adversely impact safety-related equipment and equipment important-to-safety inside containment (Section 3.5.1.2).

Response:

Inside containment only safety-related structures, systems, or components are relied on to provide safe shutdown or to mitigate postulated accidents.

No sources of primary and credible secondary missiles from which safety-related equipment inside containment must be protected have been identified. A limited number of fans may require design provisions to confirm that they are not missile sources. See the response to Q410.67 for a discussion of the potential for internally generated missiles. See the response to Q410.52 for a discussion of potential missiles from the turbine-generator. See the response to Q410.69 for a discussion of protection of safety-related structures, systems, and components from missiles generated by natural phenomena, including a design basis tornado.

SSAR Revision: NONE

Question 410.66

Provide additional detailed information on how components inside containment are prevented from producing credible missiles (Section 3.5.1.2.1.1).

Response:

As stated in Subsection 3.5.1.2, the considerations regarding missile prevention discussed in Subsection 3.5.1.1 are also applicable in the selection of equipment inside containment.

SSAR Revision: NONE

Question 410.67

Identify the sources of missiles meeting the criteria in Section 3.5.1.2.1.3 of the SSAR.

Response:

The criteria in SSAR Subsection 3.5.1.2.1.3 refer to potential missiles inside containment. The only sources included in these categories for which impacts on safety-related structures, systems, or components must be considered are a few non-safety-related fans inside containment. The safety-related systems and components needed to bring the plant to a safe shutdown are located inside the containment shield building and auxiliary building both of which have thick structural concrete exterior walls that provide protection from missiles generated in other portions of the plant. These walls are evaluated for tornado missiles that bound the energy available in internally generated missiles. Potential missiles from the turbine-generator are discussed in SSAR Subsection 3.5.1.3 and in the response to Q410.52.

The pressurized components in the high-energy portions of the high-energy systems inside containment are constructed to the ASME Code, Section III. There are no high-pressure gas storage cylinders inside containment.

The rotating equipment in containment is eliminated as a missile source for one or more of the following reasons. Equipment in use less than 2 percent of the time is not considered a missile source. This includes the reactor coolant drain pumps, the containment sump pumps and motors for valve operators, and mechanical handling equipment and pumps. Pumps and fans, such as the reactor cavity supply fans, located in compartments surrounded by structural concrete walls and containing no safety-related systems or components, are not considered missile sources. Rotating equipment with a housing or an enclosure that would contain the fragments of a postulated impeller failure is not considered a credible source of missiles. Non-safety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features has design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

SSAR Subsection 3.5.1.2.1.4 will be added as follows:

SSAR Revision:

3.5.1.2.1.4 Evaluation of Internally Generated Missiles (Inside Containment)

The consideration of credible missile sources inside containment that can adversely affect safety-related structures, systems, or components is limited to a few rotating components. The safety-related systems and components needed to bring the plant to a safe shutdown are inside the containment shield building and auxiliary building both of which have thick structural concrete exterior walls that provide protection from missiles generated in other portions of the plant.

Rotating components inside containment that are either safety-related or are constructed as canned motor pumps would contain fragments from a postulated fracture of the rotating elements and are excluded from evaluation as

missile sources. Rotating components in use less than 2 percent of the time are also excluded from evaluation as missile sources. Non-safety-related rotating equipment in compartments surrounded by structural concrete walls with no safety-related systems or components inside the compartment is not considered a missile source. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. For one or more of these reasons the non-safety related rotating equipment inside containment is considered not to be a credible missile source. Non-safety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features has design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

The high-energy portions of high-energy systems inside the containment shield building are constructed to the requirements of the ASME Code, Section III. See Table 3.6-1 for a list of the high-energy systems. There are no high-pressure gas storage cylinders inside the containment shield building.

Question 410.68

Identify all safety-related equipment and equipment important-to-safety that require protection from missiles generated by natural phenomena, including the design basis tornado (DBT) (Section 3.5.1.4).

Response:

AP600 uses safety-related equipment which functions during or following a design basis event to provide the following:

- integrity of the reactor coolant pressure boundary,
- capability to shut down the reactor and maintain it in a safe shutdown condition,
- capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

The plant safety-related equipment are located on the nuclear island (containment building, shield building, auxiliary building) which protects systems and equipment from the effects of externally generated missiles. See response to Q410.69.

SSAR Revision: NONE

Question 410.69

Discuss how safety related structures, systems, and components (SSCs) that are important-to-safety will be protected from missiles generated by natural phenomena, including the DBT (Section 3.5.1.4).

Response:

Safety-related systems and components are located within the seismic Category 1 nuclear island structures, as described in SSAR Subsection 3.5.2. The thicknesses of the exterior walls and roof of the nuclear island structures are sufficient to prevent missile perforation and scabbing by the missiles specified in SSAR Subsection 3.5.1.4 and provide protection for the safety-related systems and components from missiles generated by natural phenomena.

SSAR Revision: NONE

Question 410.70

Provide an estimate of the strike probability per year for the plant (Section 3.5.1.4).

Response:

It is estimated that the probability of wind speeds greater than the 300-mph design basis tornado is between 10^{-6} and 10^{-7} per year for an AP600 at a worst location anywhere within the contiguous United States. The probability of a missile strike by a missile similar to those represented by the design basis tornado missiles given in SSAR Subsection 3.5.1.4 would be lower than the probability of the tornado occurrence.

SSAR Revision: NONE

Question 410.71

How will the remote shutdown panel be protected from missiles generated by natural phenomena (Section 3.5.1.4)?

Response:

The remote shutdown workstation is located at elevation 100' within the nuclear island, as shown in SSAR Figure 1.2-7. Protection against missiles generated by natural phenomena is provided by the exterior walls and roof of the nuclear island structures, as discussed in the response to Q410.69.

SSAR Revision: NONE

Question 410.72

Identify all equipment important-to-safety that require protection from externally-generated missiles (Section 3.5.2).

Response:

See the responses to Q410.68 and Q410.69.

SSAR Revision: NONE

Question 410.73

How will the passive containment cooling system (including tank, valves, piping) be protected from external missiles and their effects (Section 3.5.2)?

Response:

Safety-related portions of the passive containment cooling system are located within the seismic Category I shield building except as discussed below. Protection from external missiles is provided by the external walls and roof of the shield building. This protects the tank, valve room, and most of the air baffle.

The air inlets at the top of the shield building are small and include louvers. Missiles passing through the air inlets could perforate the air baffle but would be stopped by the containment vessel. Such perforation of the air baffle would not prevent function of the air baffle or the passive containment cooling system.

Piping and cables are routed in the stair tower on the outside of the shield building. This stair tower is designed for the tornado wind speed but is not designed as a tornado missile barrier. Thus, piping and cables routed in the tower are postulated to fail in a tornado. Rupture of the piping would not prevent function of the passive containment cooling system, nor would it drain the tank below its minimum required water level. Loss of the cables or air supply would cause the PCS actuation valves to fail open and initiate water flow.

SSAR Revision: NONE

Question 410.74

How will the fuel storage pool and the fuel within the pool be protected from external missiles and their effects (Section 3.5.2)?

Response:

The fuel storage pool and the fuel within the pool are located within the nuclear island structures, as shown in SSAR Figures 1.2-8 and 1.2-9. They are protected from external missiles and their effects by the exterior walls and roof of the nuclear island as discussed in the response to Q410.69.

SSAR Revision: NONE

Question 410.75

How will the remote shutdown panel be protected from external missiles (Section 3.5.2)?

Response:

The remote shutdown workstation is located at elevation 100' within the nuclear island, as shown in SSAR Figure 1.2-7. Protection against external missiles is provided by the exterior walls and roof of the nuclear island structures, as discussed in the responses to Q410.69 and Q410.71.

SSAR Revision: NONE

Question 410.76

Provide a pipe break effects analysis (Section 3.6.1), including:

a. postulated piping failures (including those identified in Section 3.6.1.1.F of the SSAR),

b. pipe failure locations (circumferential and longitudinal break, leakage cracks, and through-wall cracks),

c. piping that meets the leak-before-break (LBB) criteria, and

d. protective structures and other features used to mitigate the consequences of the piping failures.

Response:

This is an interim response Q410.76.

See SSAR Subsection 3.4.1 and the responses to Q410.5, Q435.56, Q410.11, Q410.28, Q410.33, Q410.44, Q410.45, Q410.46, Q410.48, Q410.050, Q410.84, and Q435.56 for the effects of moderate-energy pipe breaks.

A pipe break effects analysis will be performed for high-energy piping outside containment to include the following:

- Postulated piping failures
- Pipe failure locations
- Protective structures and other features used to mitigate the consequences of the piping failures

The response to Q210.6 identified the high-energy lines that meet the leak-before-break criteria.

A final response to the question will be submitted on August 15, 1993.

SSAR Revision: NONE

Question 410.80

Describe why the turbine stop valves and feedwater control valves are credited in the single failure analysis to limit a break of the main steam or feedwater lines inside containment (Section 3.6.1).

Response:

The main steam system and feedwater system are in compliance with Standard Review Plan 10.3 and the design alternatives identified in NUREG-0138 relative to utilizing the turbine stop valves and feedwater control valves to provide redundancy for safety-related equipment.

NUREG-0800 Section 10.3 acceptance criteria specify that a design is acceptable if the integrated design is in accordance with specified criteria, including NUREG-0138. Specifically it is acceptable to take credit for all valves downstream of the main steam isolation valves (MSIV) to limit blowdown of a second steam generator. Based on NUREG-0138, the SRP indicates that the turbine stop valves and control valves are to be considered functional in order to demonstrate that the design will preclude the blowdown of more than one steam generator, assuming a concurrent single active failure.

The NRR's discussion on Issue No. 1 within NUREG-0168 concludes that in accidents involving spontaneous failures of secondary system piping not a part of the primary system boundary (where the potential consequences are significantly lower), reliance on non-safety-grade valves in the postulated accident evaluation is permitted based on the reliability of these valves.

The AP600 design has incorporated the following features into the main steam and feedwater systems designs to provide highly reliable steam and feedwater isolation.

Main Steam Isolation Valve. The main steam isolation valve automatically closes upon receipt of either of two main steam isolation signals associated with independent Class 1E electrical divisions. The valve is a gate valve controlled by a pneumatic/hydraulic operator. The energy required to close the valve is stored in the form of compressed nitrogen in one end of the actuator cylinder. The valve is maintained open by high-pressure hydraulic fluid. For emergency closure, redundant Class 1E solenoids are energized, causing the high-pressure hydraulic fluid to be dumped to a fluid reservoir and the valve to close.

The backup isolation valves (such as the turbine stop valves) receive signals derived from the protection and safety monitoring system (PMS) to actuate the valves.

Main Feedwater Isolation Valves. The main feedwater isolation valves are the same valve type and actuator design as the main steam stop valves. The valves incorporate the same type of hydraulic/pneumatic operators with redundant actuation signals and solenoids to ensure closure when actuated by a PMS feedwater isolation signal.

Main Feedwater Control Valve. In addition to the design provisions for the feedwater isolation valve, the design of the backup isolation valve (the feedwater control valve) includes provisions to provide confidence in the closure.

The main feedwater control valve is an air-operated modulating valve that fails in the closed position upon loss of instrument air or electrical power. The feedwater isolation signal provided to the valves is derived from the PMS feedwater isolation signals. The valve is procured as an active safety class C (ASME class 3) component.

To provide additional confidence of feedwater isolation, the feedwater pumps are tripped upon receipt of a feedwater isolation signal derived from the ESF feedwater isolation signal.

With the preceding design provisions, the AP600 main steam system and feedwater system provide adequate protection against design basis events requiring either feedwater or main steam line isolation.

SSAR Revision: NONE

Question 410.81

Identify the locations of all pipe whip restraints (Section 3.6.1).

Response:

This is an interim response to Q410.81.

The results of the analysis to be provided in response to Q410.76 will determine the need for pipe whip restraints.

A final response to the question will be submitted on August 15, 1993.

SSAR Revision: NONE

Question 410.83

How is the remote shutdown panel protected from the effects of pipe failures (Section 3.6.1)?

Response:

There is no high energy piping in the remote shutdown workstation room or in proximity. The only moderate-energy piping close to the remote shutdown workstation room is the fire protection system piping, located in the adjacent corridor.

The effects of moderate-energy pipe failures in the auxiliary building or adjacent buildings on the remote shutdown workstation have been addressed in the responses to Q410.28, Q410.40, Q410.46, and Q410.50.

The areas in proximity to the remote shutdown workstation containing high-energy lines are the valve/piping penetration compartment, MSIV compartment, and the turbine building. The remote shutdown workstation area is separated from those other areas by solid structural walls designed for pressurization effects. No HVAC or other penetrations provide communication between the remote shutdown workstation areas and those other areas.

If a pipe failure should occur in the valve/piping penetration compartment or MSIV compartment, steam and water are directed to the turbine building or the environment through doors, drains, and blowout panels. Any further pressure buildup in the turbine building is relieved through the existing turbine building HVAC system openings and the metal siding of the turbine building. Similarly, a high-energy piping failure that occurs in the turbine building is relieved through the existing turbine building HVAC system openings and the metal siding of the turbine building.

SSAR Revision: NONE

Question 410.84

Provide examples of equipment that are subject to environmental and flooding effects due to venting from an adjoining subcompartment (Section 3.6.1).

Response:

The subcompartments in the containment vent into the free space in the containment above the operating deck or into the space above the maintenance floor that freely communicates with the space above the operating deck. These subcompartments do not vent into adjoining subcompartments. See the responses to Q410.1, Q410.34, and Q410.39 for additional information on isolation of subcompartments. The only subcompartment subject to flooding due to flooding in another subcompartment is the RCS compartment, which drains the PXS-A, PXS-B, and CVS subcompartments. See the response to Q410.10 for additional information on floodup of the PXS subcompartments. The subcompartments in the auxiliary building containing high-energy lines vent and drain into the turbine building or other locations that do not contain safety-related equipment. See SSAR Section 3.4 for additional information on the evaluation of flood design. See the response to Q410.3 for additional information on the flooding of safety-related components. See the response to Q410.28 for additional information on internal sources of flooding.

Only safety-related equipment is considered for evaluation of flooding effects and environmental qualification. Valves are the only active mechanical equipment considered for evaluation, since there are no safety-related pumps required to operate for safe shutdown following a postulated pipe break. Instrumentation required for accident mitigation and used for postaccident monitoring is also evaluated. The evaluations of the valves and valve operators are performed to show that the valves provide the required safety-related function. Safety-related equipment either inside containment or in auxiliary building subcompartments with high-energy lines is evaluated for environmental qualification. See SSAR Section 3.11 for additional information on environmental qualification. See the responses to Q410.5 and Q435.56 for additional information on the response of safety-related equipment when flooded or wet from spray.

The CVS, PXS-A, PXS-B, and RCS compartments have a vent path to the maintenance floor. The containment isolation valves and passive core cooling system valves above the elevation 107'-2" maintenance floor are examples of equipment that experiences the environmental effects from postulated piping failures.

The containment sump level monitoring system and the containment floodup level instrumentation in the RCS compartment that collects leakage from piping failures anywhere in the containment are examples of equipment subject to flooding effects due to piping failures.

SSAR Revision: NONE

Question 410.86

Provide more detailed information regarding pressurization loads for the IRWST and the reactor vessel annulus. What is the basis for a 5 gpm leakage rate in the primary loop piping (Section 3.6.1)?

Response:

Pressurization loads for the IRWST are discussed in the response to Q410.23.

The reactor vessel annulus is evaluated for subcompartment pressurization to cover: postulated leakage cracks in piping shown to meet the leak-before-break criteria in SSAR Subsection 3.6.3 and postulated pipe breaks in piping systems larger than a 1-inch diameter that do not satisfy the leak-before-break criteria. Pipe breaks are not postulated in piping with a diameter less than or equal to 1 inch. The only piping in the reactor vessel annulus that may have a postulated pipe break is the 8-inch passive core cooling piping. However, this piping is expected to be qualified for leak-before-break. Therefore, the postulated failure for the subcompartment pressurization analysis is the 5-gpm leakage crack in the hot or cold leg piping. The 5-gpm leakage rate is the reference value used in the leak-before-break evaluation of the primary loop piping. This is ten times the leak detection capability described in SSAR Subsection 5.2.5. See Subsection 3B.5.3 of SSAR Appendix 3B for details.

SSAR Revision: NONE

Question 410.87

Which pipe(s) are postulated to fail to provide the internal reactor pressure vessel asymmetric pressurization loads (Section 3.6.1)?

Response:

Reactor coolant system piping and attachments larger than a 3-inch diameter are qualified to the leak before break (LBB) criteria. The limiting 3-inch cold and hot leg breaks, which are postulated to fail to provide the internal reactor pressure vessel asymmetric pressurization loads, where identified in the spray line and in the surge line.

SSAR Revision: NONE

Question 410.88

Provide the dimensions for the reinforced concrete walls separating the control room from the MSIV compartment. Include wall dimensions on the plant arrangement drawings (Section 3.6.1).

Response:

Wall thicknesses can be obtained by scaling off the dimensions from the full-size general arrangement drawings provided in response to Q471.1. For information, the reinforced concrete wall on column line L is two feet thick and the wall on column line K is two feet thick.

SSAR Revision: NONE

Question 410.89

Provide the results of an analysis of pipe breaks in the turbine building and their effect on the control room and remote shutdown panel (Section 3.6.1).

Response:

The effects of pipe breaks in the turbine building on the remote shutdown workstation have been addressed in the response to Q410.83. Direct access to the main control room is protected by the boundary of the access corridor to the main control room in the turbine building. The access corridor walls are designed for the pressurization effects from a high-energy pipe break in the turbine building. Further, a high-energy piping failure in the turbine building is relieved through the existing turbine building HVAC system openings and the metal siding of the turbine building.

SSAR Revision: NONE

Question 410.91

Describe how safety-related instrumentation is protected from pipe failures and their effects (Section 3.6.1).

Response:

SSAR Section 3.11 identifies the safety-related instrumentation outside containment and discusses the environmental qualification of the safety-related equipment. As discussed in the response to Q435.56, in the event of potential flooding/wetting from a moderate-energy line break, design criteria are applied for the protection of safety-related instrumentation outside containment.

In the event of a high-energy line break outside containment, the only safety-related instrumentation that could be affected is the pressure and flow instrumentation in the MSIV compartment. The pressure instrumentation is used to detect a leakage crack in the exclusion zone of the main steamlines and to isolate the MSIVs. The flow instrumentation is used for startup feedwater flow control. This instrumentation will be qualified for the environmental conditions resulting from a leakage crack in the MSIV compartment as required in order to perform its safety functions.

SSAR Revision: NONE

Question 410.93

Table 3.2-3 of the SSAR, "Classification of Components and Systems," does not list Classification of Structures. Provide the classification of structures housing all corresponding specific subsystem(s) and component(s) identified in Sections 9.4.1-9.4.3 and 9.4.6-9.4.11 of the SSAR.

Also, Table 3.2-3 does not address the containment recirculation cooling system (VCS) and the turbine building ventilation system. Provide corresponding information in Table 3.2-3 for these subsystems.

Verify that all systems and components listed in Table 3.2-3 are complete for Sections 9.4.1-9.4.3 and 9.4.6-9.4.11 of the SSAR, including system ducting data.

In addition, the titles for the HVAC system in Sections 9.4.8 and 9.4.11 differ from those in Table 3.2-3 of the SSAR for VRS- and VHS- designated systems, respectively. Clarify the discrepancies.

Response:

Table 3.2-3 of the SSAR lists mechanical and fluid system components and associated equipment class and seismic category as well as other related information. Table 3.2-3 does not provide classification of structures. Classification of structures is given in Table 3.2-2.

The containment recirculation cooling system and the turbine building ventilation system do not have any components in Class A, B, C, or D and are not included in Table 3.2-3. Only systems having components in Class A, B, C, or D are listed in the table.

Table 3.2-3 contains a complete list of Classes A, B, C, and D components for Subsections 9.4.1 through 9.4.3 and Subsections 9.4.6 through 9.4.11. Table 3.2-3 provides generic design criteria only for equipment, valves, and dampers. It does not include ducts or piping.

The correct titles for the VHS and VRS systems are:

VHS Health Physics and Hot Machine Shop HVAC System
VRS Radwaste Building HVAC System

System titles on page 3.2-10 of SSAR Section 3.2 will be revised as follows:

SSAR Revision:

VHS Health Physics~~/Control Access Area~~ and Hot Machine Shop HVAC System
VRS ~~Solid~~ Radwaste Building ~~Ventilation~~ HVAC System

Question 410.94

Provide up-to-date, half-size drawings for all HVAC systems identified in Section 9.4 of the SSAR, "Air Conditioning, Heating, Cooling, and Ventilation System," (see Sections 9.4.1-9.4.3 and 9.4.6-9.4.11 of the SSAR), including the turbine building ventilation system that was not provided previously. These drawings should include piping and instrumentation diagrams (P&IDs), orthographic HVAC systems drawings, general arrangement drawings, and system flow diagrams showing pressure, temperature and flow data for all modes of operations, including normal, abnormal and emergency modes, as applicable.

Response:

HVAC system P&IDs are included in SSAR Section 9.4, Figures 9.4.1-1 through 9.4.11-1. The half-size drawings are provided via letter ET-NRC-93-3837 (N. J. Liparulo to R. W. Borchardt, dated March 16, 1993). General arrangement drawings are included in the proprietary Section 1.2, Figures 1.2-4 through 1.2-34. All HVAC systems described in SSAR Section 9.4 are non-safety-related systems. The nuclear island nonradioactive ventilation system (VBS) is the only HVAC system with a defense-in-depth function. The flow diagram for this system is provided in the response to the Q450.6. The turbine building ventilation system serves a non-safety-related, nonseismic, nonradioactive building and does not have any safety-related or defense-in-depth function, so its P&ID is not included in the SSAR. Orthographic HVAC systems drawings and flow diagrams are not included in the SSAR and are beyond the level of information detail normally provided in the SSAR. HVAC systems do not serve any safety-related functions except containment isolation and MCR isolation.

SSAR Revision: NONE

Question 450.6

Provide flow diagrams showing normal, abnormal, smoke removal and purge, and emergency (radiation and toxic release) modes of operation flow data (i.e., cfm, temperature, and pressure) for VES and VBS habitability systems (Section 6.4).

Response (Revision 1):

Process flow diagrams VBS M5 001, VBS M5 002, VBS M5 006, VBS M5 007 and VBS M5 008 (all Revision 0) are provided via letter ET-NRC-93-3837 (N. J. Liparulo to R. W. Borchardt, dated March 16, 1993). These drawings show flow data for VBS normal, abnormal (recirculation mode - smoke in outside air intake or toxic gas release), smoke removal and purge, and emergency (high radiation modes of operation. These operating modes are described in SSAR Subsection 9.4.1.

The main control room emergency habitability system is made up of two redundant trains of emergency air storage tanks. Each train is sized to deliver approximately 20 cfm of air to the MCR to meet the ventilation and pressurization requirements for 72 hours. A connection for refilling operations is provided for each train to allow for operation beyond 72 hours.

SSAR Revision: NONE

Question 460.8

Provide the following information regarding the source terms for evaluating the expected performance of radwaste management systems (Sections 11.1 and 11.2):

a. Table 11.2-6 states that the shim bleed rate is 737 gallons per day (gpd). The staff has determined that this rate is 288 gpd using the value of the reactor coolant letdown flow given in Table 11.1-7 of the SSAR and 657.5 gpd calculated from the yearly average of the CVCS letdown given in Table 11.2-1 of the SSAR. The staff has verified that a shim bleed rate of 288 gpd gives the same reactor coolant activity (RCA) values for noble gas radionuclides given in Table 11.1-8 of the SSAR. Resolve the inconsistencies among the tables.

b. The staff has determined that a primary-to-secondary leak rate of 75 lb/day (NUREG-0017, Rev. 1 value) rather than the 100 lb/day given in Table 11.1-7 of the SSAR results in the steam generator steam activity values given in Table 11.1-8 of the SSAR for noble gases. Correct the inconsistency.

c. Tables 11.1-7 and 11.2-6 of the SSAR give the total steam flow rate as $8.4x10^6$ lb/hr and $8.4x10^5$ lb/hr, respectively. The staff believes that the first value is correct. Correct the inconsistency.

d. The staff concludes that the steam generator liquid activities and, consequently, the steam generator steam activities of halogens, Cesium (Cs), Rubidium (Rb), and other nuclides listed in Table 11.1-8 of the SSAR are not consistent with those that would be calculated using the method given in Revision 1 of NUREG-0017. Describe how the activities were calculated, and correct them, if appropriate.

Response:

a. The calculation of the coolant source terms in Table 11.1-8 used a shim bleed of 288 gallons per day based on the assumption of continued operation with no forced cold shutdowns. The Table 11.2-1 shim bleed value of 240,000 gallons per year (658 gallons per day) is based on one forced shutdown per year. The use of the 288 gallon per day value tends to maximize the source terms but the assumption also reduces the annual release of activity from this pathway. Tables 11.1-7 and 11.1-8 have been corrected to utilize a shim bleed flow of 658 gallons per day (230 lb/hr) so that the defined source term is consistent with the determination of annual releases.

   The use of 737 gallons per day in Table 11.2-6 for the shim bleed rate has been revised to 658 gallons per day and the calculated releases in Table 11.2-7 have been revised to reflect this change. There is no significant impact on the total calculated releases.

b. Table 11.1-7 has been corrected to list the primary-to-secondary leak rate as 75 lb/day instead of the 100 lb/day previously listed. The determination of source terms correctly used the 75 lb/day value.

c. Table 11.2-6 has been corrected to list the total steam flow rate as $8.4x10^6$ lb/hr (instead of the previously listed value of $8.4x10^5$ lb/hr). The determination of anticipated releases of activity to the environment correctly used the $8.4x10^6$ lb/hr value.

d. The activities in the steam generator liquid and steam that were presented in Table 11.1-8 were based on the equations provided in ANSI/ANS 18.1-1984. These equations are the same as those used in NUREG-0017, Revision 1. To assure that there is no discrepancy with NUREG-0017, the values for primary and secondary coolant activities in Table 11.1-8 have been replaced by the values calculated using the GALE code. The steam generator steam activities are not provided in the GALE code output so these have been calculated using the equations from NUREG-0017, Revision 1.

SSAR Tables 11.1-7 and 11.2-6, Sheet 1, will be revised to reflect the changes made in the analytical assumptions. Tables 11.1-8, 11.2-7, and 11.2-8 will be revised to reflect the results of the analyses performed. Sections 11.1 and 11.1.3 will be revised to reflect the fact that the realistic primary and secondary source terms are calculated using the PWR-GALE code. Subsection 11.1.6 will be revised to include a reference for the PWR-GALE code and to respecify the reference to ANSI-18.1 as Reference 3. Subsections 11.2.3.3 ans 11.2.3.4 will be revised to reflect the impact of the larger total waste flow (increase from 1742 gpd to 2150 gpd).

SSAR Revision:

(Section 11.1, third and fourth paragraphs)

The second source term is a realistic model. This source term represents the expected average concentrations of radionuclides in the primary and the secondary coolant. These values are determined ~~based on ANSI-18.1~~ using the PWR-GALE code (Reference 1) which provides the bases for estimating typical concentrations of the principal radionuclides that are expected to occur. This source term model reflects the industry experience at a large number of operating PWR plants.

The source terms presented here do not include all radionuclides that will be present. The radionuclides included are those that are significant relative to radiation shielding design as well as those listed in ~~ANSI-18.1~~ the PWR-GALE code output and used in evaluating of radwaste systems operation.

(Subsection 11.1.3)

The realistic source terms for both the reactor coolant and the secondary coolant are determined using the ~~modeling in ANSI-18.1~~ PWR-GALE code. The reference plant values provided in ANSI-18.1 (Reference 3) were adjusted to be consistent with the AP600 parameters listed in Table 11.1-7. The adjustment factors are applied to the fission products. ~~The nuclides that are corrosion products are the same as those for the design basis source term.~~ The realistic source terms are listed in Table 11.1-8.

(Subsection 11.1.6)

11.1.6 References

~~1. ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors."~~

1. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," NUREG-0017, Revision 1, March 1985.

2. M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, Oak Ridge National Laboratory, May 1973.

3. ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors."

(Subsection 11.2.3.3, second and third paragraphs)

The dilution factor required to meet the 10 CFR 20 maximum permissible concentrations is ~~320~~ 400. The required dilution flow is dependent on the liquid waste discharge rate and, while the monitor tank pumps have a design flow rate of 100 gpm, the discharge flow is controlled to be compatible with the available dilution flow. With the average liquid waste release of ~~1742~~ 2150 gallons per day, the required dilution flow is:

Release Period	Required Dilution Flow
8 hours	~~1150~~ 1200 gpm
6 hours	~~1530~~ 1600 gpm
4 hours	~~2290~~ 2400 gpm
2 hours	~~3790~~ 4800 gpm

With the nominal circulating water blowdown flow of 3500 gpm, there is sufficient dilution flow to meet the maximum permissible concentrations with a waste discharge duration of ~~2.2~~ 2.8 hours or longer. Actual plant operation is dependent on the waste liquid activity level and the available dilution flow.

(Subsection 11.2.3.4)

11.2.3.4 Release Concentrations

The annual release data provided in Table 11.2-7 represent expected releases from the plant. To demonstrate compliance with the Reference 1 maximum permissible concentration limits for effluent, the discharge concentrations have been evaluated for the release of the average daily liquid waste volume of ~~1742~~ 2150 gallons over a period of 8 hours and using the nominal circulating water blowdown flow of 3500 gpm. Table 11.2-8 lists the nuclide release concentrations and the fraction of maximum permissible concentration for effluent.

Table 11.1-7

Parameters Used to Describe Normal Operation Sources

Parameter	Symbol	Units	AP600 Value	~~ANSI Standard~~ Nominal Value
Thermal power	P	MWt	1933	3400
Steam flow rate	FS	lb/hr	8.4×10^6	1.5×10^7
Weight of water in reactor coolant system	WP	lb	3.4×10^5	5.5×10^5
Weight of water in all steam generators	WS	lb	2.1×10^5	4.5×10^5
Reactor coolant letdown flow (purification)	FD	lb/hr	5.0×10^4	3.7×10^4
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	$\cancel{1.0 \times 10^2}$ 2.3×10^2	5.0×10^2
Steam generator blowdown flow (total)	FBD	lb/hr	4.2×10^4	7.5×10^4
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	NBD	-	0.0	1.0
Flow through the purification system cation demineralizer	FA	lb/hr	5.0×10^3	3.7×10^3
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	-	0.33	0.0
Fraction of the noble gas activity in the letdown stream which is not returned to the reactor coolant system	Y	-	0.0	0.0
Primary-to-secondary leakage	FL	lb/day	~~100~~ 75	~~100~~ 75

Table 11.1-8 (Sheet 1 of 4)

Normal Plant Operation Source Terms

~~Group I~~ Noble Gases

Nuclide	Reactor Coolant Activity ($\mu Ci/g$)	Steam Generator Steam Activity ($\mu Ci/g$)
Kr-85m	0.15	5.5 ~~5.6~~ x 10^{-8}
Kr-85	0.53 ~~1.1~~	2.0 ~~4.0~~ x 10^{-7}
Kr-87	0.14	4.8 ~~5.0~~ x 10^{-8}
Kr-88	0.25 ~~0.26~~	9.5 ~~9.8~~ x 10^{-8}
Xe-131m	0.71 ~~0.81~~	2.6 ~~3.0~~ x 10^{-7}
Xe-133m	0.064 ~~0.067~~	2.5 ~~2.6~~ x 10^{-8}
Xe-133	2.4 ~~2.6~~	9.1 ~~9.8~~ x 10^{-7}
Xe-135m	0.12	4.4 ~~4.5~~ x 10^{-8}
Xe-135	0.77 ~~0.79~~	2.9 ~~3.0~~ x 10^{-7}
Xe-137	0.031	1.1 ~~1.2~~ x 10^{-8}
Xe-138	0.11	4.0 ~~4.1~~ x 10^{-8}

~~Group II~~ Halogens

Nuclide	Reactor Coolant Activity ($\mu Ci/g$)	Steam Generator Liquid Activity ($\mu Ci/g$)	Steam Generator Steam Activity ($\mu Ci/g$)
Br-84	0.014	1.4 x 10^{-7}	1.4 x 10^{-9}
I-131	0.02	2.6 ~~2.4~~ x 10^{-6}	2.6 ~~2.4~~ x 10^{-8}
I-132	0.16	5.6 ~~5.5~~ x 10^{-6}	5.6 ~~5.5~~ x 10^{-8}
I-133	0.072 ~~0.073~~	7.6 ~~7.0~~ x 10^{-6}	7.6 ~~7.0~~ x 10^{-8}
I-134	0.28 ~~0.29~~	4.5 x 10^{-6}	4.5 x 10^{-8}
I-135	0.16 ~~0.17~~	1.1 x 10^{-5}	1.1 x 10^{-7}

Table 11.1-8 (Sheet 2 of 4)

Normal Plant Operation Source Terms

~~Groups III, IV and V~~

~~Group III~~ – Rubidium, Cesium

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
Rb-88	0.17	1.0 x 10^{-6}	5.2 ~~5.1~~ x 10^{-9}
Cs-134	3.0 ~~2.90~~ x 10^{-3}	1.5 ~~1.3~~ x 10^{-6}	7.4 ~~6.7~~ x 10^{-9}
Cs-136	3.8 ~~3.70~~ x 10^{-4}	1.8 ~~1.6~~ x 10^{-7}	8.8 ~~7.8~~ x 10^{-10}
Cs-137	4.0 ~~3.90~~ x 10^{-3}	2.0 ~~1.7~~ x 10^{-6}	1.0 ~~8.7~~ x $10^{-8\ 9}$

~~Group IV – Nitrogen-16~~

~~Nuclide~~	~~Reactor Coolant Activity (μCi/g)~~	~~Steam Generator Liquid Activity (μCi/g)~~	~~Steam Generator Steam Activity (μCi/g)~~
~~N-16~~	~~40~~	~~2.10 x 10-6~~	~~2.10 x 10-7~~

~~Group V~~ – Tritium

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
H-3	1	6.0 x 10^{-5} ~~0.001~~	6.0 x 10^{-5} ~~0.001~~

Table 11.1-8 (Sheet 3 of 4)

Normal Plant Operation Source Terms

Group VI - Miscellaneous Nuclides

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
Na-24	2.5×10^{-2}	$3.7\ \cancel{3.4} \times 10^{-6}$	$1.8\ \cancel{1.7} \times 10^{-8}$
Cr-51	1.3×10^{-3}	$3.7\ \cancel{3.2} \times 10^{-7}$	$1.8\ \cancel{1.5} \times 10^{-9}$
Mn-54	$6.7\ \cancel{6.6} \times 10^{-4}$	$1.8\ \cancel{1.6} \times 10^{-7}$	$9.2\ \cancel{8.1} \times 10^{-10}$
Fe-55	5.0×10^{-4}	$1.4\ \cancel{1.2} \times 10^{-7}$	$7.0\ \cancel{6.2} \times 10^{-10}$
Fe-59	$1.3\ \cancel{1.2} \times 10^{-4}$	$3.4\ \cancel{3.0} \times 10^{-8}$	$1.7\ \cancel{1.5} \times 10^{-10}$
Co-58	1.9×10^{-3}	$5.4\ \cancel{4.7} \times 10^{-7}$	$2.7\ \cancel{2.3} \times 10^{-9}$
Co-60	2.2×10^{-4}	$6.2\ \cancel{5.4} \times 10^{-8}$	$3.1\ \cancel{2.7} \times 10^{-10}$
Zn-65	2.1×10^{-4}	$6.0\ \cancel{5.2} \times 10^{-8}$	$3.0\ \cancel{2.5} \times 10^{-10}$
Sr-89	$5.9\ \cancel{5.8} \times 10^{-5}$	$1.6\ \cancel{1.4} \times 10^{-8}$	$8.0\ \cancel{7.1} \times 10^{-11}$
Sr-90	5.0×10^{-6}	$1.4\ \cancel{1.2} \times 10^{-9}$	$7.0\ \cancel{6.2} \times 10^{-12}$
Sr-91	5.6×10^{-4}	$6.6\ \cancel{6.2} \times 10^{-8}$	$3.3\ \cancel{3.1} \times 10^{-10}$
~~Y-90~~	$\cancel{6.3 \times 10^{-7}}$	$\cancel{1.4 \times 10^{-10}}$	$\cancel{7.2 \times 10^{-13}}$
Y-91m	$3.8\ \cancel{3.9} \times 10^{-4}$	6.3×10^{-9}	3.2×10^{-11}
Y-91	2.2×10^{-6}	$5.9\ \cancel{5.2} \times 10^{-10}$	$3.0\ \cancel{2.7} \times 10^{-12}$
Y-93	2.4×10^{-3}	$2.8\ \cancel{2.7} \times 10^{-7}$	1.4×10^{-9}
Zr-95	1.6×10^{-4}	$4.5\ \cancel{3.9} \times 10^{-8}$	$2.3\ \cancel{1.9} \times 10^{-10}$
Nb-95	1.2×10^{-4}	$3.1\ \cancel{2.7} \times 10^{-8}$	$1.6\ \cancel{1.4} \times 10^{-10}$
Mo-99	2.9×10^{-3}	$6.8\ \cancel{6.0} \times 10^{-7}$	$3.4\ \cancel{2.9} \times 10^{-9}$
Tc-99m	3.0×10^{-3}	$2.5\ \cancel{2.4} \times 10^{-7}$	1.2×10^{-9}
Ru-103	$3.2\ \cancel{3.1} \times 10^{-3}$	$8.8\ \cancel{7.6} \times 10^{-7}$	$4.4\ \cancel{3.9} \times 10^{-9}$
Ru-106	$3.8\ \cancel{3.7} \times 10^{-2}$	$1.0\ \cancel{9.1} \times 10^{-5\ \cancel{6}}$	$5.2\ \cancel{4.4} \times 10^{-8}$
Rh-103m	$3.2\ \cancel{6.3} \times 10^{-3}$	$8.8\ \cancel{6.2} \times 10^{-7}$	$4.4\ \cancel{3.2} \times 10^{-9}$
Rh-106	$3.8\ \cancel{8.3} \times 10^{-2}$	$1.0\ \cancel{7.1} \times 10^{-5\ \cancel{6}}$	$5.2\ \cancel{3.5} \times 10^{-8}$
Ag-110m	$5.5\ \cancel{5.4} \times 10^{-4}$	$1.5\ \cancel{1.3} \times 10^{-7}$	$7.5\ \cancel{6.7} \times 10^{-10}$
Te-129m	$8.0\ \cancel{7.9} \times 10^{-5}$	$2.2\ \cancel{1.9} \times 10^{-8}$	$1.1\ \cancel{9.6} \times 10^{-10\ \cancel{11}}$

Table 11.1-8 (Sheet 4 of 4)

Normal Plant Operation Source Terms

~~Group VI~~ Miscellaneous Nuclides

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Generator Liquid Activity (μCi/g)	Steam Generator Steam Activity (μCi/g)
Te-129	2.0 x 10^{-2}	4.4 x 10^{-7}	2.2 x 10^{-9}
Te-131m	7.3 ~~7.2~~ x 10^{-4}	1.4 ~~1.3~~ x 10^{-7}	7.0 ~~6.4~~ x 10^{-10}
Te-131	6.7 ~~6.8~~ x 10^{-3}	5.6 ~~5.7~~ x 10^{-8}	2.8 ~~2.9~~ x 10^{-10}
Te-132	7.6 ~~7.5~~ x 10^{-4}	1.8 ~~1.6~~ x 10^{-7}	9.0 ~~8.0~~ x 10^{-10}
Ba-137m	3.6 ~~3.7~~ x 10^{-3}	1.9 ~~1.6~~ x 10^{-6}	9.4 ~~8.2~~ x 10^{-9}
Ba-140	5.6 ~~5.5~~ x 10^{-3}	1.5 ~~1.3~~ x 10^{-6}	7.3 ~~6.4~~ x 10^{-9}
La-140	1.2 x 10^{-2}	2.5 ~~2.2~~ x 10^{-6}	1.2 ~~1.1~~ x 10^{-8}
Ce-141	6.3 ~~6.2~~ x 10^{-5}	1.7 ~~1.5~~ x 10^{-8}	8.6 ~~7.6~~ x 10^{-11}
Ce-143	1.4 ~~1.3~~ x 10^{-3}	2.6 ~~2.4~~ x 10^{-7}	1.3 ~~1.2~~ x 10^{-9}
Ce-144	1.6 ~~1.7~~ x 10^{-3}	4.5 ~~3.9~~ x 10^{-7}	2.3 ~~2.0~~ x 10^{-9}
Pr-143	1.5 x 10^{-3}	2.9 x 10^{-7}	1.4 ~~1.6~~ x 10^{-9}
Pr-144	1.6 ~~3.6~~ x 10^{-3}	4.5 ~~3.1~~ x 10^{-7}	2.3 ~~1.6~~ x 10^{-9}
W-187	1.2 x 10^{-3}	2.2 ~~2.0~~ x 10^{-7}	1.1 ~~1.0~~ x 10^{-9}
Np-239	1.0 x 10^{-3}	2.3 ~~2.0~~ x 10^{-7}	1.1 ~~1.0~~ x 10^{-9}

Table 11.2-6 (Sheet 1 of 4)

Input Parameters for the GALE Computer Code

Thermal power level (MWt)	1933
Mass of primary coolant (lb)	3.46×10^5
Primary system letdown rate (gpm)	100
Letdown cation demineralizer flow rate, annual average (gpm)	10
Number of steam generators	2
Total steam flow (lb/hr)	8.4×10^{6} ~~5~~
Mass of liquid in each steam generator (lb)	1.075×10^5
Total mass of secondary coolant (lb)	2.15×10^5
Total blowdown rate (lb/hr)	8.4×10^4
Blowdown treatment method	*
Condensate demineralizer regeneration time	N/A
Condensate demineralizer flow fraction	0.33
Primary coolant bleed for boron control	
Bleed flow rate (gpd)	658 ~~737~~
Decontamination factor for I	10^5
Decontamination factor for Cs and Rb	4×10^3
Decontamination factor for others	5×10^6
Collection time (day)	21 ~~27~~
Process and discharge time (day)	0 ~~1~~
Fraction discharged	1.0

* A "1" is input to indicate that the blowdown is recycled directly to the condensate system demineralizers without prior treatment in the blowdown system.

Table 11.2-7 (Sheet 1 of 2)

Releases to Discharge Canal (Ci/Yr) Calculated by GALE Code

Nuclide	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Adjusted Combined Releases[a]	Detergent Wastes	Total
			Corrosion and activation products				
Na-24	0.0[b]	0.0	0.00014	0.00014	0.00281 ~~323~~	0.0	0.00280 ~~320~~
P-32	0.0	0.0	0.0	0.0	0.0	0.00018	0.00018
Cr-51	0.0	0.0	0.00002	0.00002	0.00039 ~~45~~	0.00470	0.00510 ~~20~~
Mn-54	0.0	0.0	0.00001	0.00001	0.00020 ~~23~~	0.00380	0.00400
Fe-55	0.0	0.0	0.00001	0.00001	0.00015 ~~18~~	0.00720	0.00740
Fe-59	0.0	0.0	0.0	0.0	0.00004	0.00220	0.00220
Co-58	0.0	0.0	0.00003	0.00003	0.00059 ~~68~~	0.00790	0.00850 ~~60~~
Co-60	0.0	0.0	0.0	0.0	0.00007 ~~08~~	0.01400	0.01400
Ni-63	0.0	0.0	0.0	0.0	0.0	0.00170	0.00170
Zn-65	0.0	0.0	0.0	0.0	0.00007 ~~08~~	0.0	0.00007 ~~08~~
W-187	0.0	0.0	0.00001	0.00001	0.00019 ~~22~~	0.0	0.00019 ~~22~~
Np-239	0.0	0.0	0.00001	0.00001	0.00021 ~~25~~	0.0	0.00021 ~~25~~
			Fission products				
Br-84	0.0	0.0	0.0	0.0	0.00008	0.0	0.00008
Rb-88	0.00003	0.0	0.0	0.00003	0.00060	0.0	0.00060
Sr-89	0.0	0.0	0.0	0.0	0.00002	0.00009	0.00011
Sr-90	0.0	0.0	0.0	0.0	0.0	0.00001	0.00002
Sr-91	0.0	0.0	0.0	0.0	0.00004 ~~05~~	0.0	0.00004 ~~05~~
Y-91M	0.0	0.0	0.0	0.0	0.00003	0.0	0.00003
Y-91	0.0	0.0	0.0	0.0	0.0	0.00008	0.00009
Y-93	0.0	0.0	0.00001	0.00001	0.00019 ~~22~~	0.0	0.00019 ~~22~~
Zr-95	0.0	0.0	0.0	0.0	0.00005 ~~06~~	0.00110	0.00110 ~~20~~
Nb-95	0.0	0.0	0.0	0.0	0.00003 ~~04~~	0.00190	0.00190
Mo-99	0.0	0.0	0.00003	0.00003	0.00065 ~~75~~	0.00006	0.00071 ~~81~~
Tc-99M	0.0	0.0	0.00002	0.00002	0.00042 ~~48~~	0.0	0.00042 ~~48~~
Ru-103	0.0	0.0	0.00004	0.00005	0.00095 ~~109~~	0.00029	0.00120 ~~40~~
Rh-103M	0.0	0.0	0.00004	0.00005	0.00094 ~~108~~	0.0	0.00094 ~~110~~
Ru-106	0.00005	0.00001	0.00052	0.00058	0.01151 ~~332~~	0.00890	0.02000 ~~200~~
Rh-106	0.00005	0.00001	0.00052	0.00058	0.01151 ~~332~~	0.0	0.01200 ~~300~~
Ag-110M	0.0	0.0	0.00001	0.00001	0.00016 ~~19~~	0.00120	0.00140

Table 11.2-7 (Sheet 2 of 2)

Releases to Discharge Canal (Ci/Yr) Calculated by GALE Code

Nuclide	Shim Bleed	Misc. Wastes	Turbine Building	Combined Releases	Adjusted Combined Releases[a]	Detergent Wastes	Total
Ag-110	0.0	0.0	0.0	0.0	0.00002	0.0	0.00002 ~~03~~
Sb-124	0.0	0.0	0.0	0.0	0.0	0.00043	0.00043
Te-129M	0.0	0.0	0.0	0.0	0.00002 ~~03~~	0.0	0.00002 ~~03~~
Te-129	0.0	0.0	0.0	0.0	0.00003	0.0	0.00003
Te-131M	0.0	0.0	0.00001	0.00001	0.00012 ~~14~~	0.0	0.00012 ~~14~~
Te-131	0.0	0.0	0.0	0.0	0.00002 ~~03~~	0.0	0.00002 ~~03~~
I-131	0.00121 ~~093~~	0.00033 ~~15~~	0.00025	0.00180 ~~34~~	0.03568 ~~090~~	0.00160	0.03700 ~~300~~
Te-132	0.0	0.0	0.00001	0.00001	0.00017 ~~20~~	0.0	0.00017 ~~20~~
I-132	0.00016	0.00007	0.00010	0.00033 ~~10~~	0.00658 ~~230~~	0.0	0.00660 ~~230~~
I-133	0.00057 ~~20~~	0.00026 ~~04~~	0.00062	0.00145 ~~086~~	0.02889 ~~1985~~	0.0	0.02900 ~~000~~
I-134	0.00009	0.00004	0.0	0.00014	0.00282 ~~009~~	0.0	0.00280 ~~009~~
Cs-134	0.00054 ~~57~~	0.00003 ~~02~~	0.00007	0.00064 ~~67~~	0.01281 ~~551~~	0.01100	0.02400 ~~700~~
I-135	0.00041 ~~03~~	0.00019 ~~01~~	0.00060	0.00121 ~~064~~	0.02402 ~~1472~~	0.0	0.02400 ~~1500~~
Cs-136	0.00004	0.0	0.00001	0.00005	0.00106 ~~10~~	0.00037	0.00140 ~~50~~
Cs-137	0.00072 ~~77~~	0.00005 ~~03~~	0.00010	0.00086 ~~90~~	0.01709 ~~2077~~	0.01600	0.03300 ~~700~~
Ba-137M	0.00001 ~~72~~	0.0~~0003~~	0.00009	0.00010 ~~84~~	0.00196 ~~1942~~	0.0	0.00200 ~~1900~~
Ba-140	0.0	0.0	0.00007	0.00008	0.00153 ~~75~~	0.00091	0.00240 ~~70~~
La-140	0.00001	0.0	0.00012	0.00013 ~~12~~	0.00249 ~~84~~	0.0	0.00250 ~~80~~
Ce-141	0.0	0.0	0.0	0.0	0.00002	0.00023	0.00025
Ce-143	0.0	0.0	0.00001	0.00001	0.00023 ~~27~~	0.0	0.00023 ~~27~~
Ce-144	0.0	0.0	0.00002	0.00003 ~~02~~	0.00050 ~~58~~	0.00390	0.00440 ~~50~~
Pr-144	0.0	0.0	0.00002	0.00003 ~~02~~	0.00050 ~~58~~	0.0	0.00050 ~~58~~
All others	0.0	0.0	0.0	0.0	0.00001	0.0	0.00001
Total (except tritium)	0.00391 ~~39~~	0.00101 ~~029~~	0.00355	0.00848 ~~723~~	0.16848 ~~723~~	0.08975	0.26000
Tritium release		690 curies per year					

(a) An adjustment of 0.16 Ci/yr is added by PWR-GALE code to account for anticipated operational occurrences such as operator errors that result in unplanned releases.
(b) An entry of 0.0 indicates that the value is less than 10^{-5} Ci/yr.

Table 11.2-8 (Sheet 1 of 2)

Comparison of Calculated Liquid Release Concentrations with 10 CFR 20 Maximum Permissible Concentrations

Nuclide	Discharge Conc. (μCi/ml)(a)	Maximum Permissible Conc. (μCi/ml)(b)	Fraction of MC
Na-24	1.2 ~~1.4~~E-09	5.0E-05	2.4 ~~2.7~~E-05
P-32	7.7E-11	9.0E-06	8.6E-06
Cr-51	2.2E-09	5.0E-04	4.4 ~~4.5~~E-06
Mn-54	1.7E-09	3.0E-05	5.7E-05
Fe-55	3.2E-09	1.0E-04	3.2E-05
Fe-59	9.4E-10	1.0E-05	9.4E-05
Co-58	3.6 ~~3.7~~E-09	2.0E-05	1.8E-04
Co-60	6.0E-09	3.0E-06	2.0E-03
Ni-63	7.3E-10	1.0E-04	7.3E-06
Zn-65	3.0 ~~3.4~~E-11	5.0E-06	6.0 ~~6.8~~E-06
W-187	8.1 ~~9.4~~E-11	3.0E-05	2.7 ~~3.1~~E-06
Np-239	9.0 ~~1.1~~E-11 ~~10~~	2.0E-05	4.5 ~~5.4~~E-06
Br-84	3.4E-11	4.0E-04	8.6E-08
Rb-88	2.6E-10	4.0E-04	6.4E-07
Sr-89	4.7E-11	8.0E-06	5.9E-06
Sr-90	8.6E-12	5.0E-07	1.7E-05
Sr-91	1.7 ~~2.1~~E-11	2.0E-05	8.6 ~~1.1~~E-07 ~~06~~
Y-91m	1.3E-11	2.0E-03	6.4E-09
Y-91	3.9E-11	8.0E-06	4.8E-06
Y-93	8.1 ~~9.4~~E-11	2.0E-05	4.1 ~~4.7~~E-06
Zr-95	4.7 ~~5.1~~E-10	2.0E-05	2.4 ~~2.6~~E-05
Nb-95	8.1E-10	3.0E-05	2.7E-05
Mo-99	3.0 ~~3.5~~E-10	2.0E-05	1.5 ~~1.7~~E-05
Tc-99m	1.8 ~~2.1~~E-10	1.0E-03	1.8 ~~2.1~~E-07
Ru-103	5.1 ~~6.0~~E-10	3.0E-05	1.7 ~~2.0~~E-05
Rh-103m	4.0 ~~4.7~~E-10	6.0E-03	6.7 ~~7.8~~E-08
Ru-106	8.6 ~~9.4~~E-09	3.0E-06	2.9 ~~3.1~~E-03
Rh-106	5.1 ~~5.6~~E-09	1.0E-04	5.1 ~~5.6~~E-05
Ag-100m	6.0E-10	6.0E-06	1.0E-04
Ag-110	8.6 ~~1.3~~E-12 ~~11~~	1.0E-08	8.6 ~~1.3~~E-04 ~~03~~
Sb-124	1.8E-10	7.0E-06	2.6E-05
Te-129m	8.6 ~~1.3~~E-12 ~~11~~	7.0E-06	1.2 ~~1.8~~E-06
Te-129	1.3E-11	4.0E-05	3.2E-07
Te-131m	5.6 ~~6.0~~E-11	8.0E-06	7.0 ~~7.5~~E-06
Te-131	8.6 ~~1.3~~E-12 ~~11~~	2.0E-04	4.3 ~~6.4~~E-08
I-131	1.6 ~~1.4~~E-08	1.0E-06	1.6 ~~1.4~~E-02

Table 11.2-8 (Sheet 2 of 2)

Comparison of Calculated Liquid Release Concentrations with 10 CFR 20 Maximum Permissible Concentrations

Nuclide	Discharge Conc. (μCi/ml)[a]	Maximum Permissible Conc. (μCi/ml)[b]	Fraction of MC
Te-132	7.3 ~~8.6~~E-11	9.0E-06	8.1 ~~9.5~~E-06
I-132	2.8 ~~9.8~~E-09 ~~10~~	1.0E-04	2.8 ~~9.8~~E-05 ~~06~~
I-133	1.2 ~~8.6~~E-08 ~~09~~	7.0E-06	1.8 ~~1.2~~E-03
I-134	1.2 ~~3.9~~E-09 ~~11~~	4.0E-04	3.0 ~~9.6~~E-06 ~~08~~
Cs-134	1.0 ~~1.2~~E-08	9.0E-07	1.1 ~~1.3~~E-02
I-135	1.0 ~~6.4~~E-08 ~~09~~	3.0E-05	3.4 ~~2.1~~E-04
Cs-136	6.0 ~~6.4~~E-10	6.0E-06	1.0 ~~1.1~~E-04
Cs-137	1.4 ~~1.6~~E-08	1.0E-06	1.4 ~~1.6~~E-02
Ba-137m	8.6 ~~8.1~~E-10 ~~09~~	1.0E-06	8.6 ~~8.1~~E-04 ~~03~~
Ba-140	1.0 ~~1.2~~E-09	8.0E-06	1.3 ~~1.4~~E-04
La-140	1.1 ~~1.2~~E-09	9.0E-06	1.2 ~~1.3~~E-04
Ce-141	1.1E-10	3.0E-05	3.6E-06
Ce-143	9.8 ~~1.2~~E-11 ~~10~~	2.0E-05	4.9 ~~5.8~~E-06
Ce-144	1.9E-09	3.0E-06	6.3 ~~6.4~~E-04
Pr-144	2.1 ~~2.5~~E-10	6.[illegible]-04	3.6 ~~4.1~~E-07
H-3	3.0 ~~2.7~~E-04	1.0E-03	3.0 ~~2.7~~E-01
		Total =	3.5 ~~3.3~~E-01

(a) Discharge concentration based on release of average daily discharge over an 8 hour period with 3500 gpm dilution flow.
(b) Maximum permissible concentrations are from Reference 1.

Question 460.9

Provide the following information regarding the liquid radwaste management system (Section 11.2):

a. Section 11.2.2.1.1 of the SSAR states that any of the four ion exchangers provided in series for processing all liquid radwaste streams can be manually bypassed. In addition, the SSAR does not provide sufficient test data that supports additional radioactivity removal from waste streams by a third and a fourth ion exchanger in series with the first two ion exchangers. Further, the staff notes that the additional credit due to the ion exchangers of the chemical volume control system for the shim bleed stream is already built into the code since such credit is used for calculating the primary coolant concentrations of radionuclides. For the above reasons, the staff estimates that the following DFs appear more appropriate than the ones used in Table 11.2-6 of the SSAR:

Halogens	10^3
Cs, Rb	20
Others	10^3

These values assume that there are at least two mixed beds in series.

In light of the above discussion, provide justification for the DFs in Table 11.2-6 of the SSAR or correct the values, as appropriate. In resolving the staff's concern regarding the DFs, clarify whether at least two of these four beds are mixed, because only a combination of two mixed beds in series can provide optimum removal capability for different categories of radionuclides.

b. Section 11.2.3 of the SSAR states that, except for the steam generator blowdown wastes, all other processed liquid radwastes will be discharged to the environment rather than recycled within the plant. However, Table 11.1-8 of the SSAR shows a tritium RCA of $1\mu Ci/gm$, which indicates that a moderate amount of tritium will be recycled (see Table 2-6 of Revision 1 to NUREG-0017). Further, the staff is concerned that, with minimum processing of Cs and Rb nuclides in the waste streams (DF of 20) and maximum discharge of the processed liquid radwastes, Cs and Rb radionuclide releases via liquid effluents can pose a problem in terms of compliance with the concentration limits of 10 CFR Part 20 and the offsite dose limits of Appendix I of 10 CFR Part 50, unless there is substantial dilution of the waste streams prior to their discharge. In light of the above discussion, describe why the AP600 is designed for minimum recycling, which is at variance with industry practice and standards (see Subsection 4.1.5 of ANSI/ANS 55.6)

c. Collection times for various wastes given in Table 11.2-6 of the SSAR appear to be inconsistent with the values that will result from using the methodology of Revision 1 to NUREG-0017. The NUREG recommends using only 80 percent of one collector tank's full capacity as the fill volume provided there are two tanks of equal capacity. It is not clear whether the volumes of all tanks given in Table 11.2-2 of the SSAR represent 80 percent of their full capacity. Justify or resolve the inconsistency.

d. The process and discharge times for the various liquid waste streams given in Table 11.2-6 of the SSAR (1 day for all of them) appear to be inconsistent with the methodology of Revision 1 to NUREG-0017. The

calculated process time cannot be further increased by half of the calculated discharge time when the monitor tanks have smaller volumes than their corresponding holdup or collection tanks. Further, the methodology recommends using only 80 percent of the full capacity of one monitor tank, provided that there are two monitor tanks of equal capacity. Justify or resolve the inconsistency.

e. The input from leakage of the spent fuel pit liner, the reactor containment cooling system, and the reactor coolant pump seal, and the sampling drains of the secondary coolant system are either not included in Table 11.2-1 of the SSAR or are much lower in the subject table than the inputs from these sources given in Section 3.2.1 of Revision 1 to NUREG-0017 and ANSI/ANS-55.6. The staff recognizes that the product of the expected activity level and the daily input given in the subject table for the applicable sources discussed above is greater than the corresponding NUREG products. However, the inputs have a bearing on the sizing of the liquid radwaste system equipment and collection and processing times. Additionally, the equipment drain tank may also be a source (NUREG-0017 gives a much higher combined total waste generation rate for equipment drains and clean wastes than the AP600 design). Provide the missing information. Provide the reasons for the inconsistencies or correct the values.

f. Appendix 1A of the SSAR demonstrates that the liquid radwaste management system for the AP600 design meets Position C.1.2 of RG 1.143 with respect to the design features for applicable tanks (i.e., tanks located outside the containment and carrying radioactive materials). However, Table 11.2-3 of the SSAR, which lists the applicable tanks with level and alarm features, does not include the condensate storage tank (CST). Clarify whether the CST has level indication and alarm features in accordance with Position C.1.2.1 of RG 1.143. Also, clarify whether the AP600 design includes the specific design features discussed in Positions C.1.2.2 through C.1.2.5 of RG 1.143 for all the applicable tanks.

g. Clarify whether the AP600 design has a single liquid waste discharge path as shown in Figure 11.2-1 of the SSAR. Also, clarify whether the AP600 design permits different categories of wastes to be discharge simultaneously at any time, provided such cumulative discharge is within the applicable regulatory limits. If not, identify the design features that preclude such a simultaneous discharge.

Response:

a. The system decontamination factors (DFs) specified in Table 11.2-6 are appropriate for use in determining anticipated annual releases of activity for the liquid pathway. The processing of liquid wastes is schematically shown in Figure 11.2-3 (provided as part of the response to Q460.3). This figure identifies the DF values assumed for each component in the process path. The system decontamination factors suggested in the question do not properly reflect the design of the liquid radwaste system which contains a series of two mixed resin ion exchangers and two cation bed ion exchangers (one specified as the deep bed filter). Nor do they reflect the fact that the shim bleed flow will be processed by the mixed bed ion exchanger in the chemical and volume control system. The specific issues raised in the question are addressed below:

<u>Bypassing of ion exchangers</u>: While it is true that any of the four ion exchangers in the liquid radwaste system can be bypassed, the bypassing of any ion exchanger(s) would be an operating plant decision based on the

activity levels in the water being processed for discharge. It is possible that if activity levels are low enough there would be no need for processing and the water could be discharged directly to environment. In modeling the anticipated annual releases to the environment it is assumed that all equipment is in operation. It is noted that the analysis to determine normal annual releases does not model what would happen in every expected circumstance. If the activity levels are low, the amount of processing can be reduced. If the activity levels are exceptionally high, it may be necessary to perform additional processing by recirculating the monitor tank contents through the series of ion exchangers. The calculation of normal releases to the environment is representative only.

Credit for mixed bed ion exchangers in the chemical and volume control system: As stated in the question, credit for the ion exchangers in the chemical and volume control system are built into the PWR-GALE code as part of the calculation of primary coolant activity levels. However, the mixed bed ion exchanger is also part of the shim bleed flow path and should be included as part of the processing train.

Credit for more than two ion exchangers in series: NUREG-0017 does not disallow credit for more than two ion exchangers in series but it also does not address DFs that would exist if there were more than two of one type in series; thus the use of two mixed bed ion exchangers plus two cation bed ion exchangers is consistent with the modeling described in NUREG-0017. Credit for a third ion exchanger of one type in series is not recognized by NUREG-0017; however, the situation that exists on the AP600 relative to the inclusion of the chemical and volume control system ion exchanger is not that of three ion exchangers operating in series. The shim bleed flow passes through the mixed bed ion exchanger in the chemical and volume control system and then is collected in the effluent drain tank. Also collected in this tank are the flows from equipment drains (90 gpd at 1.07 times primary coolant activity). After this combined solution accumulates to a volume justifying processing for discharge, it is assumed to be directed through the series of four ion exchangers; two of which are mixed bed and two of which are cation bed. The first of the mixed bed ion exchangers in the liquid waste processing train does not function as a "second ion exchanger in series." First of all, it is separated from the chemical and volume control system ion exchanger by both time and by the collection tank. Secondly, it is processing not only shim bleed but equipment drain water as well. The situation of a third ion exchanger of the same type operating in series does not exist.

b. No recycling of the liquid wastes is assumed. The value of 1.0 microcurie per gram for the tritium concentration in the primary coolant comes from the NUREG-0017 model which does not actually calculate a tritium concentration. The specified tritium concentration is used without modification by the GALE code. Even with no recycling of wastes there will be a buildup of tritium in the reactor coolant. The use of 1.0 microcurie per gram has been evaluated with respect to annual tritium production and average shim bleed flow and is found to be appropriate for the AP600.

Relative to the stated staff concerns regarding the low level of processing of Cs and Rb, see the above discussion in response to part a. As indicated in Figure 11.2-3, the DF for Cs and Rb is not the suggested value of 20 but is 4000 for the shim bleed and 2000 for the remaining processed wastes.

Current industry practice with regard to recycle of reactor coolant system effluents is to discharge a sufficient amount of coolant to prevent tritium levels in the coolant from building to a level which would cause

maintenance problems. The practice of many plant operators is to discharge between three and five system volumes of effluent per year for this purpose. The AP600 is designed to operate without recycle of effluents from the reactor coolant system, but also to minimize the amount of effluent generated. The 240,000 gallons per year of effluent generated and discharged corresponds to approximately four system volumes per year, which is comparable to current industry practice for tritium control.

c. The collection times have been recalculated to be consistent with the methodology of Revision 1 to NUREG-0017. Using the equations from Revision 1 to NUREG-0017, the collection times are:

   Combined shim bleed, equipment drains, and clean waste . . . . 21 days
   Dirty waste . . . . 8 days

   Table 11.2-6, Sheets 1 & 2, will be revised to reflect these collection times. The annual releases have been reanalyzed and Tables 11.2-7 & 11.2-8 will be revised per the new analysis. The reanalysis shows no significant impact on the total calculated releases.

d. Based on the equations listed in Revision 1 to NUREG-0017, the processing and discharge time would be 0.1 days. The annual releases have been reanalyzed assuming that there is no delay associated with processing and discharge.

   Table 11.2-6 has been revised to reflect this change in the processing and discharge times. The annual releases have been reanalyzed and Tables 11.2-7 & 11.2-8 will be revised per the new analysis. The reanalysis shows no significant impact on the total calculated releases.

e. The format of the information originally provided in Table 11.2-1 of the SSAR followed the November 1990 draft update of ANSI/ANS-55.6. The last approved version of this standard is ANSI/ANS-55.6-1979, and Table 11.2-1 has been revised to reflect this version; in particular, the value for spent fuel pit liner leakage has been changed from 25 gpd to 700 gpd, and the floor drains value of 675 gpd has been eliminated since the floor drains entry pertains to the November 1990 draft update of ANSI/ANS-55.6 and is not included in ANSI/ANS-55.6-1979. Note that there is no net change in waste volume.

   Also, some of the previous interpretations of ANSI/ANS-55.6 used for the AP600 have been revised as follows:

   - Reactor containment cooling, changed from 15 gpd to 500 gpd
   - Hot shower, changed from 800 gpy to 10 gpd.
   - Hand wash, changed from 30,000 gpy to 240 gpd.
   - Equipment and area decontamination, changed from 60,000 gpy to 130 gpd.
   - Laundry, changed from 300 gpd to 350 gpd.

   For each of these, the original value used reflects an off-normal event that was translated to annual average release rates. The replacement values reflect the expected annual waste generation.

A specific exception was taken to the values in ANSI/ANS-55.6 for reactor coolant pump seal leakage. Since the AP600 incorporates canned motor reactor coolant pumps, there is no seal leakage.

Secondary coolant system sampling drains are not routed to the radwaste system. These drains are routed to the plant's waste water system which is monitored for radioactivity before disposal.

f. The CST is level controlled and communicates with both the Demineralized Water Transfer and Storage System and the Condensate System. The storage tank level is maintained by a high and low level switch which opens and closes the demineralized water supply valve. On low level, the supply valve is opened and it is closed on high level. High and low level alarms are provided. The CST supplies water as makeup to the condenser. As level falls in the condenser, a system control valve modulates flow to the condenser hotwell. As level rises in the condenser hotwell, another system control valve modulates condensate pump discharge return to the CST. The CST is provided with high-high level alarm and overflow capability. Overflow from the CST and the CST drains are routed to the plant's waste water system where further monitoring, dilution, and routing to the liquid radwaste system can occur. Design features of the CST conform to Positions C.1.2.1 through C.1.2.5 of RG 1.143.

g. The AP600 design has a single liquid discharge path from the radwaste system as shown in Figure 11.2-1. The design does permit different categories of wastes (e.g. detergent waste and floor drain waste) to be discharged simultaneously provided the cumulative discharge is within applicable regulatory limits.

All liquid discharge from the AP600 radwaste system is from monitor tanks; these wastes are mixed and sampled such that total activity inventories may be accurately determined prior to discharge.

Sheets 1 & 2 of Table 11.2-6 will be revised to reflect changes in the PWR-GALE code input for the times for waste collection and for waste processing & discharge. Tables 11.2-7 and 11.2-8, which present the results of the PWR-GALE code analysis will be corrected to reflect the revised analysis of anticipated annual releases.
Table 11.2-1 will be revised to reflect the 1979 version of ANSI/ANS-55.6. The changes in the waste flow rates resulted in changes to the PWR-GALE code input listed in sheet 2 of Table 11.2-6. Tables 11.2-7 and 11.2-8, which present the results of the PWR-GALE code analysis, will also be corrected to reflect the revised analysis of anticipated annual releases. (See the response to Q460.8.)

SSAR Revision:

Table 11.2-1 (Sheet 1 of 2)

Liquid Inputs and Disposition

	Collection Tank and Sources	Expected Input Rate	Activity	Basis	Disposition
1.	Effluent holdup tanks				Filtered, demineralized, and discharged
	Chemical and volume control system letdown	240,000 gpy	100% of reactor coolant	AP600 specific calculations	
	Leakage inside containment (to reactor coolant drain tank)	10 gpd	167% of reactor coolant	ANSI/ANS-55.6	
	Leakage outside containment (valve and pump leakoffs piped to effluent holdup tanks)	80 gpd	100% of reactor coolant	ANSI/ANS-55.6	
2.	Waste holdup tank				Monitored, filtered, demineralized as required, and discharged
	Reactor containment cooling	500 ~~15~~ gpd	0.1% of reactor coolant	ANSI/ANS-55.6	
	Spent fuel pit liner leakage	700 ~~25~~ gpd	0.1% of reactor coolant	ANSI/ANS-55.6	
	~~Floor drains~~	~~675 gpd~~	~~0.1% of reactor coolant~~	~~ANSI/ANS-55.6~~	
	Sampling drains	200 gpd	5% of reactor coolant	ANSI/ANS-55.6	

Table 11.2-1 (Sheet 2 of 2)

Liquid Inputs and Disposition

	Collection Tank and Sources	Expected Input Rate	Activity	Basis	Disposition
3.	Detergent waste				Filtered, monitored, and discharged. If necessary, processed with mobile equipment.
	Hot shower	10 gpd ~~800 gpy~~	10^{-7} μCi/g	ANSI/ANS-55.6-1979, applied to AP600 specific design (minimum 18 month fuel cycle, 17 day refueling outage). ~~adjusted for 18 month refueling cycle (ANS value multiplied by 12/18). This envelopes the 24 month cycle.~~	
	Hand wash	240 gpd ~~30,000 gpy~~	10^{-7} μCi/g	ANSI/ANS-55.6-1979, applied to AP600 specific design (minimum 18 month fuel cycle, 17 day refueling outage). ~~adjusted for 18 month refueling cycle (ANS value multiplied by 12/18). This envelopes the 24 month cycle.~~	
	Equipment and area decontamination	130 gpd ~~60,000 gpy~~	0.1% of reactor coolant	ANSI/ANS-55.6-1979, applied to AP600 specific design (minimum 18 month fuel cycle, 17 day refueling outage).	
	Laundry	350 gpd ~~300~~	10^{-4} μCi/g	ANSI/ANS-55.6-1979, applied to AP600 specific design (minimum 18 month fuel cycle, 17 day refueling outage).	
4.	Chemical Wastes	Varies			Processed with mobile equipment

Table 11.2-6 (Sheet 2 of 4)

Input Parameters for the GALE Computer Code

Equipment Drains and Clean Waste	
Equipment drains flow rate (gpd)	90
Fraction of reactor coolant activity	1.07
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	2×10^3
Decontamination factor for others	10^5
Collection time (day)	21 ~~27~~
Process and discharge time (day)	0 ~~1~~
Fraction discharged	1.0
Dirty Waste	
Dirty waste input flow rate (gpd)	1400 ~~915~~
Fraction of reactor coolant activity	0.012
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	2×10^3
Decontamination factor for others	10^5
Collection time (day)	8 ~~16~~
Process and discharge time (day)	0 ~~1~~
Fraction discharged	1.0
Blowdown Waste	
Blowdown fraction processed	0
Decontamination factor for I	N/A
Decontamination factor for Cs and Rb	N/A

Question 460.10

Provide the following information regarding the gaseous radwaste management system (Section 11.3):

a. Tables 11.2-6 and 11.3-1 of the SSAR give different holdup times for Xenon and Krypton in the charcoal delay beds. Clarify the discrepancy between the tables.

b. Describe the basis for the RCS degassing days (17.4 and 1.0 for Xenon and Krypton, respectively) given in Table 11.2-6 of the SSAR. The waste gas system releases given in Table 11.3-3 of the SSAR do not appear to be correct. Confirm the acceptability of this information or correct it, as appropriate.

c. Discuss the provisions for monitoring the individual performance of the equipment within the charcoal delay bed system. Include a list of alarmed process parameters for the delay bed system.

Response:

a. The gaseous radwaste management system includes two 100 percent capacity charcoal delay beds, both of which would normally be in use. The holdup times presented in Table 11.3-1 reflect credit for only one of the two delay beds in operation. This is reflected in the footnote at the bottom of Table 11.3-1. The holdup times currently presented in Table 11.2-6 reflect credit for both of the delay beds in operation.

The anticipated annual releases have been reanalyzed using the assumption of only one of the two delay beds in operation and Table 11.2-6, Sheet 3, has been revised to reflect this change in assumptions.

b. The holdup times specified for RCS degassing (17.4 days for xenons and 1.0 days for kryptons) do not represent the time associated with the RCS degassing process. These holdup times are the delay times that would exist if the continuous gas flow rate through the charcoal delay beds were at the maximum gas flow rate associated with the RCS degassing process (0.5 cfm instead of the average flow of 0.01 cfm for normal reactor operation). This is a conservative modelling of RCS degassing.

As discussed above in the response to part a, the anticipated annual releases have been reanalyzed assuming that only one of the two delay beds is in service. With this assumed mode of operation, the holdup times for the RCS degassing operation are reduced to 8.7 days for xenons and 0.5 days for kryptons.

c. As shown on Figure 11.3-2, the AP600 gaseous radwaste system contains provisions for continuously monitoring the moisture level at the inlet of the guard bed in order to provide

confidence that moisture will not intrude beyond the moisture separator, which may adversely affect system performance.

Monitoring performance of individual components in the gaseous radwaste system is done by collecting and analyzing grab samples. As shown on Figure 11.3-2 sample pumps and connections are provided which allow the collection of grab samples at the inlet and outlet of the guard bed, between the two delay beds, and at the outlet of the second delay bed.

SSAR Table 11.2-6, Sheet 3 will be revised to reflect the changes in analysis assumptions discussed in the responses to parts a and b. The reanalysis of anticipated annual releases is reflected in revisions made to Table 11.3-3, Sheet 1 and to Table 11.3-4.

The first paragraph of SSAR Subsection 11.3.3.4 will be revised to reflect the changes in the calculated site boundary whole body and skin doses associated with the annual gaseous releases

SSAR Revision:

(Subsection 11.3.3.4)

With the annual releases of noble gases listed in Table 11.3-3, the immersion doses to an individual located at the site boundary are ~~1.1~~ 1.2 mrem to the whole body and ~~3.2~~ 3.5 mrem to the skin. These doses are based on the annual average atmospheric dispersion factor from Section 2.3 (2.0×10^{-5} seconds per cubic meter). These doses are below the 10 CFR 50, Appendix I, limits of five mrem per year to the whole body and 15 mrem per year skin dose.

Table 11.2-6 (Sheet 3 of 4)

Input Parameters for the GALE Computer Code

Decontamination factor for others	N/A
Collection time	N/A
Process and discharge time	N/A
Fraction discharged	0
Regenerant waste	N/A
Gaseous Waste System	
Continuous gas stripping of full letdown purification flow	None
Holdup time for xenon, normal operation (days)	435 ~~870~~
RCS degassing (days)	8.7 ~~17.4~~
Holdup time for krypton, normal operation (days)	24 ~~49~~
RCS degassing (days)	0.5 ~~1.0~~
Fill time of decay tanks for gas stripper	N/A
Gas waste system: HEPA filter	None
Auxiliary building: Charcoal filter	None
Auxiliary building: HEPA filter	None
Containment volume (ft^3)	1.73×10^6
Containment atmosphere internal cleanup rate (ft^3/min)	N/A
Containment high volume purge:	
Number of purges per year (in addition to two shutdown purges)	0
Charcoal filter efficiency (%)	90
HEPA filter efficiency (%)	99

Table 11.3-3 (Sheet 1 of 3)

Expected Annual Average Release of Airborne Radionuclides As Determined by the PWR-GALE Code, Revision 1

(Release Rates in Ci/yr)

Noble Gases[(a)(b)]

	Waste Gas System	Building Ventilation			Condenser Air Removal System	Total
		Cont.	Aux. Bldg.	Turbine		
Kr-85m	7.0 ~~1.0~~E+00	2.0 ~~1.9~~E+01	3.0E+00	0.	1.0E+00	3.1 ~~2.4~~E+01
Kr-85	8.1 ~~8.6~~E+01	9.8 ~~9.7~~E+01	1.5E+00	0.	0.	1.84E+02
Kr-87	0.	6.0E+00	3.0E+00	0.	1.0E+00	1.0E+01
Kr-88	4.0E+00 ~~0.~~	2.3 ~~2.2~~E+01	5.0E+00	0.	3.0E+00	3.5 ~~3.0~~E+01
Xe-131m	7.8E+01	8.6 ~~8.3~~E+02	1.5E+01	0.	7.0E+00	1.0E+03 ~~9.3E+02~~
Xe-133m	1.0E+00 ~~0.~~	5.1 ~~5.0~~E+01	1.0E+00	0.	0.	5.3 ~~5.1~~E+01
Xe-133	2.4E+02 ~~7.7E+01~~	2.6 ~~2.5~~E+03	5.2 ~~5.1~~E+01	0.	2.4E+01	2.9 ~~2.65~~E+03
Xe-135m	0.	1.0E+00	2.0E+00	0.	1.0E+00	4.0E+00
Xe-135	0.	2.0 ~~1.9~~E+02	1.6E+01	0.	8.0E+00	2.2 ~~2.14~~E+02
Xe-138	0.	0.	2.0E+00	0.	1.0E+00	3.0E+00
~~Total~~	~~2.42E+02~~	~~3.72E+03~~	~~9.95E+01~~	~~0.~~	~~4.6E+01~~	~~4.1E+03~~
					Total	4.4E+03

Additionally

H-3 released via gaseous pathway	77
C-14 released via gaseous pathway	7.3
Ar-41 released via containment vent	34

Table 11.3-4

Comparison of Calculated Offsite Airborne Concentrations With 10 CFR 20 Limits

	Maximum Permissible Conc. µCi/ml	Expected Site Boundary Conc. µCi/ml	Fraction of MPC (expected)	Maximum Site Boundary Conc. µCi/ml	Fraction of MPC (maximum)
Kr-85m	1.0E-7	2.5 ~~1.9~~E-11	2.5 ~~1.9~~E-4	9.8 ~~7.6~~E-11	9.8 ~~7.6~~E-4
Kr-85	7.0E-7	1.4 ~~1.5~~E-10	2.0 ~~2.1~~E-4	5.4 ~~6.2~~E-10	7.7 ~~8.9~~E-4
Kr-87	2.0E-8	7.9E-12	4.0E-4	1.9E-11	9.5E-4
Kr-88	9.0E-9	2.8 ~~2.4~~E-11	3.1 ~~2.7~~E-3	1.1 ~~9.4~~E-10 ~~11~~	1.2 ~~1.0~~E-2
Xe-131m	2.0E-6	8.0 ~~7.4~~E-10	4.0 ~~3.7~~E-4	6.2 ~~5.8~~E-10	3.1 ~~2.9~~E-4
Xe-133m	6.0E-7	4.2 ~~4.0~~E-11	7.0 ~~6.7~~E-5	3.3 ~~3.2~~E-10	5.5 ~~5.3~~E-4
Xe-133	5.0E-7	2.3 ~~2.1~~E-9	4.6 ~~4.2~~E-3	7.6 ~~7.0~~E-8	1.5 ~~1.4~~E-1
Xe-135m	4.0E-8	3.2E-12	8.0E-5	2.4E-12	6.0E-5
Xe-135	7.0E-8	1.8 ~~1.7~~E-10	2.6 ~~2.4~~E-3	5.3 ~~5.1~~E-10	7.6 ~~7.3~~E-3
Xe-138	2.0E-8	2.4E-12	1.2E-4	3.2E-12	1.6E-4
I-131	2.0E-10	5.2E-14	2.6E-4	7.9 ~~8.0~~E-13	4.0E-3
I-133	1.0E-9	1.9E-13	1.9E-4	1.3 ~~1.4~~E-12	1.3 ~~1.4~~E-3

a. Maximum permissible concentration is from Reference 1.

b. Expected site boundary concentration based on annual releases predicted by the PWR-GALE code (Table 11.3-3) and an annual average X/Q of 2.0 x 10^{-5} seconds per cubic meter.

c. Maximum site boundary concentration based on adjusting the releases predicted by the PWR-GALE code (Table 11.3-3) to reflect operation with design basis fuel defect level of 0.25% and an annual average X/Q of 2.0 x 10^{-5} seconds per cubic meter.

Question 460.11

Provide the following information regarding the solid radwaste management system (Section 11.4):

a. Since solidification and encapsulation are not the same, clarify whether either of the above two options may be used for processing spent resins in addition to a third option, namely dewatering the resins (note that encapsulation is not generally used for processing spent resins). Also, clarify whether the AP600 solid radwaste management system design deviates from the EPRI Requirements Document for passive reactor designs. The Requirements Document recommends only dewatering for processing the spent resins.

b. Identify the specific design features provided in the system design to comply with GDCs 60, 63 and 64 as they relate to (1) control of release of radioactive materials to the environment from the plant areas where the solid radwastes are processed, and (2) monitoring radiation levels and leakage.

c. Clarify whether the description and discussion of acceptability of the portable grouting unit that may be used for processing spent filters is within the COL applicant's scope. If it is within the AP600 design scope, provide specific details of the unit.

d. The staff is concerned that the projected (Tables 11.4-4 and 11.4-5 of the SSAR) annual solid radwaste volumes to be disposed (1729 CF for the expected case and 3843 CF for the maximum case) are significantly lower than that actually shipped volume for operating PWRs (EPRI NP-5528, February 1988, Volume 2 - Plants Without Evaporators for the Years 1985 and 1986: 9550 CF). The staff recognizes that the projected volume agrees with the value proposed in the EPRI Requirements Document (1750 CF per year). The EPRI-proposed value depends on following what EPRI regards as sound design and operating techniques outlined in the document (Paragraph B.1.2.2 of Appendix B of Chapter 12) for reducing the shipment of processed solid waste volume. One of the operating techniques is to avoid solidification and instead use only dewatering for solidifying the wet solid wastes. As stated above, the AP600 design includes solidification as one of the options. The staff is concerned that the storage volume allotted for processed solid wastes may be inadequate if it is to be based on the projected shipment volumes given in the SSAR tables. Therefore, provide justification for the projected volumes given in the subject SSAR tables or revise the values as appropriate.

e. Clarify why the AP600 design does not include phase separator tanks, as recommended in the EPRI Requirements Document for passive reactor designs.

f. Section 11.4.1.3 of the SSAR identifies the capability to store processed and packaged solid wastes at the site for at least six months to account for possible delay or disruption of offsite shipping of the wastes as one of the design objectives of the solid waste management system. However, there is no description of the on-site storage facility in the SSAR. Provide a description of the facility, and clarify whether it conforms with the recommendations identified for such a facility in Section 5.4 of Chapter 12 of the EPRI Requirements Document for passive reactor designs.

Response:

a. As indicated in SSAR Subsection 11.4.2.4.1, an alternative to the base design of spent resin dewatering is to solidify (not encapsulate) the spent resins using a qualified binding agent. The portable components needed for solidification, however, are not provided as part of the AP600 solid waste management system. Dewatering is the design basis as specified in the Utility Requirements Document, and the AP600 design is in full conformance.

b. Relative to solid wastes, General Design Criterion 60, Control of Releases of Radioactive Materials to the Environment, requires that means be provided to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Means are provided in the solid waste management system to handle the applicable categories of solid radwaste as indicated in SSAR Section 11.4. To control the release of radioactive materials to the environment, the areas and components in the radwaste building that process radioactive solid wastes are located in rooms with exhaust ventilation that discharges through HEPA filters, as indicated in SSAR Subsection 9.4.8.2. In addition, local ventilation exhausts with local HEPA, filters as described in SSAR Subsection 9.4.8.2, are provided as follows:

   - Drum Compactor
   - Low Activity Waste Monitoring Unit
   - Low Activity Waste Dryer
   - Sorting Glove Box and Box Compactor
   - Laundry Sorting Table Hood
   - Respirator Cleaning Disassembly Hood
   - Respirator Cleaning Decon Glove Box
   - Respirator Cleaning Filter Decon Hood
   - High Pressure Hot Water Decon Booth
   - Abrasive Decon Booth

   To control the release of radioactive material that could be removed from stored solid waste by water contact, the external access ways to the radwaste building have raised thresholds. The holdup volume is at least 30,000 gallons, which is equivalent to the operation of the fire water system at 1000 gpm for 30 minutes. In addition, the tanks within the radwaste building that contain a significant liquid volume are within a seismically designed area that retains the maximum liquid volume as described in the response to Q460.6.

   General Design Criterion 63, Monitoring Fuel and Waste Storage, requires that means be provided to detect conditions that may result in excessive radiation levels and to initiate appropriate actions. For the solid waste management system, the wastes with the most potential for high radiation levels are the spent ion exchange resins and filter cartridges, especially those from the chemical and volume control system ion exchangers and filters. The radiation levels of the spent resin tanks are monitored without entering the rooms. Floor penetrations with shield plugs above the spent resin tanks are provided to allow the radiation levels in the tank rooms to be monitored by lowering detectors down the outside of the tanks. The shield doors to the spent resin tank rooms are normally locked to prevent inadvertent entry.

As described in SSAR Subsection 11.4.2.3.2, the dose rates of high-activity filter cartridges are measured during the changeout process when the filter is raised into the high-activity filter transfer cask (but before the bottom cover of the shield cask is secured) using a long-handled radiation probe. The measured dose rate determines the precautions taken during subsequent handling operations. The high-activity filter cartridges can be transferred into and out of the high-activity filter storage tubes using the high-activity filter transfer cask without direct exposure to personnel. The filters in storage can be monitored with minimal exposure at any time through sampling ports (normally closed by shielded plugs) of each storage tube. Ports with shield plugs that may be used for monitoring stored waste containers are also provided on the high-activity filter processing cask and the onsite storage casks.

Dry, solid wastes are normally monitored when received at the radwaste building and are then transferred to the appropriate temporary storage area (low, moderate, or high) depending on the measured dose rate as described in SSAR Subsection 11.4.2.3.3. Local shielding can be used within the temporary and packaged waste storage areas to segregate the higher dose rate items and thereby minimize the dose rate in the rest of the storage areas. The radiation levels in the temporary and packaged waste storage areas should be relatively low. The areas can be entered for monitoring at any time. These storage areas have doors that can be locked to control access.

Sample hoods are provided for monitoring and analyzing potentially hazardous and/or mixed wastes. These sample hoods have reinforced counter tops to support temporary shielding. The waste storage drums can be monitored at any time, and local shielding can be used to control the dose rate in the room.

Relative to solid radwastes, Criterion 64, Monitoring Radioactivity Releases, requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. Airborne effluents from the radwaste building are monitored by an exhaust radiation monitor, as described in SSAR Subsection 11.5.2.3.2. The spent resin waste container fill stations and the high-activity filter processing cask include provisions for smear survey and decontamination of the external surfaces of the waste containers after filling, as described in SSAR Subsections 11.4.2.3.1 and 11.4.2.3.2, respectively. Boxes and drums containing lower-activity dry wastes are also surveyed and decontaminated, as described in SSAR Subsection 11.4.2.3.3. A clean waste monitoring unit and bag monitor are used to verify that wastes segregated and sorted for nonradioactive disposal are nonradioactive. Hand-held survey meters are used to prevent removal of radioactivity from the radwaste building by personnel. Portal monitors are provided where personnel exit from the radiologically controlled area in the radwaste building. The arrangement of the radwaste building allows the corridors and vehicle access areas to be very low radioactivity areas, thereby minimizing the need for any decontamination operations.

c. The portable grouting unit is an AP600 component as indicated in SSAR Table 11.4-12 (Sheet 4). The unit will be procured to a performance-type specification, and the detailed design will be by the selected vendor. The unit will be designed to prepare at least 60 gallons of grout. Water will be supplied by local hose stations, and the mixing tank will have a hinged cover for adding water and cement. The mixer and pumps types and power ratings will be selected for efficient mixing and pumping of a wide range of cement grouts. The grout will be delivered to the waste container through a hose. After grouting operations, the tank and mixer would

be hosed down and the flush water would be discharged through the pump and hose to a 55-gallon drum for settling and decanting. The settled solids would be accumulated and disposed as nonradioactive waste.

d. As indicated in Chapter 12, Section 1.5.1 of Volume III of the Utility Requirements Document, the goal established for low-level dry and wet waste volumes of 1750 ft^3/yr for PWRs is based on the performance of the best 10 percent of currently operating PWR plants. Since the AP600 is generally smaller and greatly simplified relative to current plants, including significant reductions in the quantity of valves, pumps, and other components requiring maintenance, the generation of solid wastes for the AP600 should be within that produced by the best 10 percent of current plants. The AP600 also has integrated solid radwaste and support systems and facilities that are based on the best of current practices. These systems and facilities are not all available in currently operating plants. Also, techniques described in Appendix B to Chapter 12 of the Utility Requirements Document are to be used for radwaste minimization.

The AP600 waste quantities are based on dewatering wet solid wastes and does not include solidification. However, the alternative described for plants that may be required to solidify spent resins would result in no change in the waste volume shipped from the plant. The vinyl ester styrene binder process does not increase the waste volume but fills the space existing between resin beads with binder.

The overall performance of the nuclear power industry in reducing the volume of solid radioactive waste shipments may be observed by evaluating the historical data provided in NUREG/CR-2907[1]. The average solid radioactive waste shipments for all operating PWR power plants from 1978 to 1988 is about 1.06 x 10^{-4} m^3/yr per MWe-Hr. The Utility Requirements Document (URD) solid radioactive waste goal of 1750 ft^3/yr is equivalent to 1.18 x 10^{-5} m^3/yr per MWe-Hr based on a 600-MWe plant operating with an annual average capacity factor of 80 percent. Thus, the 1978 to 1988 average waste shipments for all PWRs is about nine times greater than the URD goal. The average solid radwaste shipments for all PWRs reduces to about 4.4 x 10^{-5} m^3/yr per MWe-Hr from 1985 to 1988, which is about four times the URD goal. This reduction by over a factor of two shows the results of early efforts by utilities to reduce the volume of solid radioactive waste shipments due to rapidly rising disposal costs and decreasing disposal site volume allotments.

Some plants with aggressive radioactive waste volume reduction programs did much better than the average. For example, between 1985 and 1988, Diablo Canyon 1 and 2 shipped an average of 1.03 x 10^{-5} m^3/year per MWe-Hr. This is about 10 percent less than the URD goal. Also Calvert Cliffs, which has had a program for solid radioactive waste reduction in place since 1985[2], shipped about 2 x 10^{-5} m^3/yr per MWe-Hr, which is about 1.7 times greater than the URD goal. For 1987 and 1988 when the program was in full operation, the shipped solid radioactive waste was only 15 percent greater than the URD goal.

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1 NUREG/CR-2907 (BNL-NUREG-51581, Vol. 9) Radioactive Materials Released from Nuclear Power Plants, Annual Report 1988, published July 1991.

2 Reducing LLW Generation at Calvert Cliffs, Nuclear News, March 1988.

These few examples demonstrate that with aggressive solid radioactive waste programs, current nuclear power plants are being operated within the URD goal for solid radioactive waste volume. It is expected that evaluation of shipped radwaste volumes since 1988 would show continued reductions as more plants increase their waste minimization efforts.

For the AP600 there is a large reduction in the number of components (pumps, valves, etc) that can become radioactively contaminated. This will result in a large reduction in the generation of solid radioactive wastes due to maintenance operations. The waste generated during maintenance operations is a large fraction of the volume of dry, compressible waste and contaminated equipment. For Diablo Canyon in 1988, this waste accounted for about 89 percent of the total radioactive waste shipments. For an AP600 radwaste estimate, this type of waste accounts for about 79 percent of the total expected annual disposal volume (SSAR Table 11.4-4).

In addition to inherent simplifications, the AP600 has features that permit convenient recycle of materials rather than disposal. The laundry includes two 135-pound-capacity pass-through commercial washer/extractors and four 110-pound-capacity dryers that together provide a high throughput capability. This allows some plastic items, which can result in large solid radwaste generation, to be replaced with reusable items. This includes items such as drum liner bags used to collect solid wastes and personnel protection garments and accessories. The respirator cleaning and decontamination facilities provided in the radwaste building will increase the reuse of tools and components and will allow decontamination of items for disposal as nonradioactive. The clean waste verification facility permits the maximum segregation of nonradioactive wastes from radioactive wastes. Thus, the AP600 has integrated capabilities (that are generally not all available at current nuclear power plants) to minimize the solid radioactive waste disposal volume.

In the response to Q460.5c., the solid radioactive waste storage facilities provided in the radwaste building were evaluated relative to the 30-day storage duration specified in Branch Technical Position ETSB 11-3. The storage durations for four general types of solid radioactive waste based on the expected annual radwaste volumes of SSAR Table 11.4-4 are as follows:

Radwaste Category	Estimated Packaged Radwaste Storage Duration
Spent Ion Exchange Resins Filter Charcoal	12 months [1]
High-Activity Filter Cartridges	17 months [1]
Other Wastes in Drums	28 months [2]
Wastes in Boxes	36 months [2]

1 These storage durations conservatively use only two of three onsite storage casks.

2 These drums and boxes use the same storage area. The space allocated to one (for example, drums) could be increased to increase its storage duration, while the storage duration for the other (for example, boxes) would decrease, assuming generation rates remain the same.

e. Chapter 12 of the Utility Requirements Document specifies phase separators only for plants that use backwash or cross-flow filters that reject solids in a sludge or slurry form as indicated in URD Paragraph 5.2.2.2.2 of Chapter 12. As noted in the Rationale for that paragraph, PWR plants that use only cartridge filters do not require this tank. The AP600 employs only cartridge filters and does not generate sludges that require settling and decanting.

f. There are four locations where packaged wastes may be stored until shipped to a disposal facility: (1) the two spent resin container fill stations, (2) the onsite storage casks, (3) the high-activity filter processing cask and (4) the packaged waste storage room. The performance of these storage areas has been described in the responses to Q460.5c. and Q460.11d. The following is a physical description of each storage area.

The spent resin container fill stations (SSAR Subsection 11.4.2.5.2) are two cells with thick concrete walls. Each cell is about 10 x 10 x 16 feet high. A thick shield cover, with lifting provisions and shield plugged ports for fill head access and smear and dose rate survey, forms the top of each cell. The platform at elevation 119'-0" is designed for shield cover laydown and provides work space around the top of the cells.

The high-activity filter processing cask is described in SSAR Subsection 11.4.2.2.14 and Table 11.4-12 (Sheet 3).

The packaged waste storage room (SSAR Subsection 11.4.2.5.2) is a shielded, unobstructed area 24 feet wide by 36 feet long and has a clear height of at least 18 feet. The shielded fork lift (SSAR Subsection 11.4.2.2.3 and Table 11.4-12 (Sheet 3), is used to handle waste boxes and palletized waste drums into and out of storage (SSAR Subsection 11.4.2.3.3). Planned positioning of waste containers and portable shielding may be used to minimize the dose rate in the portions of the room periodically accessed by personnel. Mobile racks for hanging lead blankets and shield panels on casters are available for flexible response to changing conditions in the storage room.

The storage facilities are in conformance with URD Chapter 12, Section 5.4, except that remote viewing (URD Paragraph 5.4.2.11) is not provided inside the high-activity filter processing cask or inside the onsite storage casks. Records track the contents of these casks, and ports provided for processing and surveying can be used to remotely observe the contents, using portable TV cameras if necessary. Also, the standard transportation equipment (URD Paragraph 5.3.3.5) is considered the responsibility of the

SSAR Revision: NONE

Question 460.13

Clarify whether the monitors provided in the exhaust ducts of the Annex II building, the fuel handling area of the auxiliary building, and the radiologically-controlled portion of the auxiliary building automatically facilitate connection of the applicable exhaust (i.e., monitor detects high radiation in the associated exhaust duct) to the containment air filtration system (Sections 9.4.3.1.2 and 11.5).

Response:

The radiation monitors in the exhaust ducts of the annex II building, the fuel handling area of the auxiliary building, and the radiologically controlled portion of the auxiliary building are shown in SSAR Figure 9.4.3-1, Sheets 5, 6, and 8. A high radiation signal from these monitors isolates the normal supply and (unfiltered) exhaust ducts to the affected area and automatically connects the containment air filtration system (VFS) exhaust filters and fans (SSAR Figure 9.4.7-1, Sheets 1 and 2) to the isolated area. As discussed in SSAR Subsections 9.4.3.2.4 and 9.4.7.1.2, the VFS exhaust fans are used to maintain a slightly negative air pressure in the isolated zones to prevent unfiltered releases to the environment during conditions of high airborne radioactivity in these areas.

SSAR Revision: NONE

Question 460.14

Clarify whether the steam generator blowdown system and component cooling water (CCW) radiation monitors provide any automatic control features. If not, indicate for what essential purpose these monitors are provided (e.g., manual actions to isolate the affected CCW loop or terminate SG blowdown on detection of high radiation by the subject monitor) (Section 11.5).

Response:

Radiation monitors are provided on the component cooling system (CCS) for detection of radiological leakages into the CCS. This feature is described in SSAR Subsection 9.2.2.7. With this detection, the CCS may be manually isolated and leaks repaired.

The steam generator blowdown system (BDS) contains a radiation monitor for detection of radioactivity. When pre-set levels of the radioactivity are detected in BDS, a remote manual diversion valve directs BDS fluid to the liquid radwaste system for processing. Automatic isolation, if high levels of radiation are detected, is provided through the system's blowdown control valve and an automatic isolation valve located upstream of the system's heat exchangers. Refer to SSAR Subsection 10.4.8 on BDS.

SSAR Revision: NONE

Question 460.15

Section 9.3.3, 11.5.3, and 11.5.4 of the SSAR provide incomplete information on radiological sampling provisions for process and effluent streams. For example, the sampling provisions for the waste monitor tank contents, the detergent waste monitor tank contents, the steam generator blowdown, and the condenser air removal system have not been identified. Further, there is no reference to tritium measurements. Identify how the sample provisions for the liquid and gaseous process and effluent streams for the AP600 design meet the sampling provisions for such streams identified in Tables 1 and 2 of Section 11.5 of the SRP.

Response:

Table 9.3.3-2 of SSAR Subsection 9.3.3 will be updated to include all local sample points in the primary and secondary systems. The revised table will be available in July 1993.

SSAR Revision: NONE

Question 460.16

Provide the following information regarding accident monitoring instrumentation (Section 11.5). State if any of these items are outside the design scope of the AP600, but are within the design scope of the COL applicant.

a. The recommended range for the noble gas effluent monitor for the condenser air removal system (Revision 3 to RG 1.97) is 10^{-6} μCi/cc to 10^{5} μCi/cc. The monitor is not needed if the effluent discharges through a common plant vent (however, this is not the case for the AP600). Table 11.5-1 of the SSAR provides a much narrower range, i.e., 10^{-6} μCi/cc to 10^{-1} μCi/cc. Why is the range so limited?

b. Describe the calibration frequency and technique for calibrating the monitors.

c. Describe the methods used to ensure representative measurements are taken with appropriate background correction.

d. Describe the location of instrument readout(s) and the methods of recording this information, including the method or procedure for transmitting or disseminating the information or data.

e. Provide assurance of the capability to obtain readings at least every 15 minutes during and following an accident.

f. Describe the procedures or calculation methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and consideration of radionuclide spectrum distribution as a function of time after shutdown.

g. Describe the sampling system design, including the sampling media, to demonstrate how the design meets the requirements identified in Clarification No. 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (page II.F.1-7).

h. Describe the sampling technique to be used under accident conditions to demonstrate how the technique meets the requirements identified in Clarification No. 3 of NUREG-0737 (pages II.F.1-7 and II.F.1-8).

i. Describe the sampling technique to ensure the system capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident as identified in Table II.F.1-2 of NUREG-0737 (page II.F.1-9).

Response:

a. The condenser air removal effluent discharges through a vent that is independent of the plant vent. The condenser air removal effluent noble gas range will be modified to detect from 10^{-6} to 10^{5} μCi/cc.

b. The radiation monitors are calibrated every refueling outage or following an unsatisfactory functional test or replacement of a major monitor component. Each detector is given an isotopic calibration using decay-corrected sources. The monitor electronics are independently calibrated using simulated (electronic) input signals.

c. Grab samples of the process and effluent streams are taken and analyzed in the plant chemistry laboratory. The results of the laboratory analysis are used to correlate the readings obtained from the in-situ radiation measurements. The installed radiation monitors are then electronically adjusted.

d. The radiation monitoring system displays and records monitored data at the central radiation processor and displays the data on the plant instrumentation system displays in the main control room. Long-term historical radiation data is stored in the central radiation processor. The data is transmitted from the local radiation processors to the central radiation processor by serial data link. The central radiation processor transmits radiation monitoring data and displays to the plant instrumentation system.

e. Radiation readings for those monitors that are required post accident are continuous. The monitors are provided with reliable, uninterruptible power, which ensures their availability for postaccident monitoring.

f. The local radiation processor associated with the plant vent monitor accepts analog signal inputs from plant vent effluent flow and temperature. These analog signals are used to calculate concentrations and flow rates at standard conditions. These signals are also used by the radiation processor to calculate total process and sample flow for an operator-selected period and total discharge for an operator-selected period.

   Software is provided to determine the radioactive release rates as a function of time by combining the gross concentrations and flow rate provided by the plant vent monitor with internally stored radionuclide spectrum distribution as a function of time for different types of accidents. Decay of the spectra from the times that the accidents begin to the time that the release rates are calculated is also included. Capabilities to manually or automatically enter actual spectra from laboratory analysis of grab samples are also included in the software.

g. The particulate and iodine filters of the extended range plant vent monitor are used as grab sample modules to provide the capability to collect representative samples of iodines and particulates for onsite analysis during and following an accident. Filters on the high-activity sample flow path are housed in shielded enclosures designed for quick removal and replacement of filter media. Filter removal is provided by quick-disconnect couplings. After removal, the filter is placed in a shielded cask for transport to the onsite laboratory. The filter enclosure and transport cask are designed and the access routes are selected in accordance with the requirements of NUREG-0737 to keep personnel exposure in sample handling and transport below the GDC 19 limits of 5 rem whole-body exposure and 75 rem to the extremities during the accident. The iodine filter is a fixed silver zeolite filter, hydrated in accordance with the recommendations of Information Notice No. 86-43.

h. SSAR Subsection 11.5.2.3.3 describes the technique used to keep the sample flow isokinetic with the exhaust flow. The radiation processor associated with the plant vent monitor accepts analog signals from plant vent effluent flow and temperature. These signals are used by the radiation processor to control the sample flow to maintain isokinetic extraction at the sample nozzles for vent exhaust flow rate variation of ± 20 percent. The normal-range detectors are deactivated automatically when the concentrations exceed their normal ranges. Also, the sample flow bypasses the normal range skid, and only a small portion is extracted for the extended range skid.

i. A representative sample of the plant vent effluent is collected isokinetically from an array of nozzles in the plant vent. The sample tubing run from the nozzles to the radiation monitor skids and the number of bends and elbows are minimized to reduce the losses of iodides and particulates. Heat-tracing is provided to eliminate entrained moisture that could degrade the sample filter media. As described in item g, provisions are made to limit the occupational dose to personnel below the GDC 19 limits during sample handling and transport.

   The onsite laboratory is fully equipped with the instrumentation and equipment required to perform isotopic analyses from grab samples.

SSAR Table 11.5-1 will be revised as follows:

SSAR Revision:

Table 11.5-1

Radiation Monitor Detector Parameters

Detector	Type	Service	Isotopes	Nominal Range
TDS-JE-RE001	B	Condenser Air Removal Discharge	Kr-85 Xe-133	1.0E-6 to 1.0E~~-4~~+5 µCi/cc