Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

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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 January 13, 1994 and January 26, 1994.

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 January 13, 1994 letter and the corresponding letters that provided our response.

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

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

Robert J. [illegible] for
Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

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

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NTD-NRC-94-4099
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED APRIL 14,1994

RAI No.	Issue
220.024	Wind-induced failure of nonsafety structures
220.028	Loading effects of air baffle on containment
220.029	Use of (1.,0.4,0.4) method vs SRSS method
220.041	Soil pressure effects on embedded wall section
220.045	Subcompartment global pressure/temperature effects
220.048	Capability of connection, reinforcement pattern
230.037	Cutoff frequencies of fixed base model
230.038	Seismic Cat I structures in stick model
230.039	Live loads in modeling shield & auxiliary building
230.041	Basemat in SSI analyses
230.046	Exclusion of additional accidental torsion
230.049	Modeling procedures
420.123	Unavailability values used for I&C unavailability
440.049	ADS Phase B test facility configuration
440.051	Limiting single failures in light of ADS change

ATTACHMENT B
CROSS REFERENCE OF WESTINGHOUSE RAI RESPONSE TRANSMITTALS TO NRC LETTER OF JANUARY 13, 1994

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
420.123	Unavailability values used for I&C unavailability	01/13/94	04/14/94
440.049	ADS Phase B test facility configuration	01/13/94	04/14/94
440.050	Impact of ADS design change on OSU & SPES	01/13/94	03/24/94
440.051	Limiting single failures in light of ADS change	01/13/94	04/14/94

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Question 220.24

Discuss the effects of wind-induced failure of non-safety-related structures on safety related structures, systems, and components (SSCs). If the collapse of non-safety-related structures due to wind loading does not adversely impact the function of the safety-related SSCs, the use of 1.0 as the importance factor is suitable. If not, an importance factor of 1.11 should be used in the design of such structures. Therefore, the SSAR should provide a commitment that all SSCs not designed for wind loads should be analyzed using the 1.11 importance factor or be checked that their mode of failure will not affect the ability of safety-related SSCs to perform their intended safety functions. Provide that commitment or justify deviation from such a commitment (Section 3.3.1 of the SSAR).

Response:

Wind induced failure of nonsafety-related structures due to the design wind (110 mph) does not adversely impact the function of the safety-related structures, systems and components. Wind induced failure of nonsafety-related structures is evaluated for the 300 mph tornado which is more severe than the design wind. SSAR Subsection 3.3.2.3 provides criteria for this evaluation. Nonsafety-related structures adjacent to the nuclear island are analyzed for the 300 mph tornado. Local failures, such as blow off of siding, are permitted to relieve loads on the structural frame. Such local failures are evaluated to confirm that they could not generate missiles more severe than those used in the design of the nuclear island. SSAR Subsection 3.5.1.4 describes the design of the safety related structures for tornado missiles. SSAR Subsection 3.5.2 describes those structures, systems and components to be protected against external missiles.

The tornado evaluation of nonsafety-related st uctures provides assurance that wind loading does not adversely impact the function of the safety-related structures, systems, and components. Therefore the use of 1.0 as the importance factor is suitable.

SSAR Revision: NONE

Question 220.28

Since the air baffle is supported at the top of the shield building and is attached to the steel containment through 3" diameter pipe supports, discuss what considerations were given to the design of the air baffle, and the effect of the baffle on the steel containment and shield building for all the loading conditions, specifically the seismic loads and the severe accident loads (thermal and pressure). Describe in more detail (possibly with figures) the flexible seal at the top of the air baffle and the connection to the shield building roof (Section 3.8.2 of the SSAR).

Response:

The containment air baffle is shown on the General Arrangement sections in SSAR Figures 1.2-12 and 1.2-13. The upper air baffle is attached to the shield building roof. The lower portion is attached to the steel containment. A flexible seal is provided at elevation 236'; this seal accommodates the differential deflections of the containment vessel and shield building under seismic, design basis and severe accident loads.

The containment air baffle and its effect on the containment vessel are described in SSAR Subsection 3.8.4.1.3. The portion of the air baffle attached to the containment vessel is shown in Figure 3.8.4-1. The design considers the pressure and thermal growth of the containment vessel under design basis and severe accident conditions. The seismic loads are transmitted through the pipe struts to the containment vessel.

SSAR Revision: NONE

Question 220.29

Overall seismic loads result in axial compression and tangential shear stresses which are greatest at the base of the cylindrical portion of the containment. Westinghouse evaluated the shell for dead load, live load, and seismic load at the critical section close to the bottom tangent line. Westinghouse reported that the calculated stress was 2721 psi and the corresponding allowable stress for the Level C Service Limit was 4438 psi based on NE-3133.6. Axial and tangential shear stresses were evaluated in accordance with the ASME Code Case N-284. The maximum value of the interaction ratio was reported as 0.5 and the allowable interaction ratio was 1.0. However, Westinghouse combined seismic loads by the (1.0, 0.4, 0.4) method and added to the dead load and live load. Provide the basis for the use of only this method in combining seismic loads and not the SRSS method as described in Section 3.7.2.6 (Section 3.8.2 of the SSAR).

Response:

SSAR Subsection 3.7.2.2 identifies the SRSS method or the (1.0, 0.4, 0.4) method as alternative methods of combination. Generally, the SRSS method is used to combine responses of a single variable such as the longitudinal stress in the vessel. The SRSS method is conservative for design evaluations in which two or more stresses must be considered, such as when calculating stress intensity or buckling interaction ratios. The (1.0, 0.4, 0.4) method is an acceptable alternative which considers the relative magnitudes of two or more stress components that must be further combined for evaluation against an allowable stress or interaction ratio.

The interaction ratio for buckling given in the response to RAI 220.4 was also calculated using both the SRSS method and the (1.0, 0.4, 0.4) method. The interaction ratio using the SRSS method was 0.504 compared with 0.502 using the (1.0, 0.4, 0.4) method. The two methods gave similar results for this case.

SSAR Revision: NONE

Question 220.41

Discuss the design of the embedded portion of the exterior walls of the nuclear island of seismic Category I structure and the methods for the consideration of static soil pressure and the soil pressure induced by the earthquake. Westinghouse should follow the guidelines documented in the staff position for the embedded wall and retaining wall design. Evaluate the potential local soil failure around the embedded walls during the design seismic event (Section 3.8.4 of the SSAR).

Response:

The embedded portions of the exterior walls of the nuclear island are designed for dead loads, live loads, SSE loads, hydrostatic loads due to groundwater and probable maximum flood, static soil pressure loads, surcharge loads, and soil pressure induced by the SSE.

The walls are designed according to ACI 349 with the load combinations given in Table 3.8.4-2.

The static soil pressure is based on at-rest soil pressure. The soil pressure induced by the SSE is based on the Mononobe-Okabe formula. Since the exterior walls are assumed to be non-yielding, the forces obtained by the Mononobe-Okabe formula are multiplied by two. Two-dimensional SSI analysis results are also used to establish the soil pressure induced by the SSE and to verify the structural integrity of the walls. The potential for local soil failure is also considered in the design.

SSAR Revision : None

Question 220.45

Provide a commitment to design all subcompartments for global pressure/temperature effects, and provide the actual pressure/temperature values to be used (Section 3.8.4 of the SSAR).

Response:

All subcompartments are designed for the pressure and temperature effects calculated for the postulated pipe breaks. The postulated pipe breaks are described in SSAR Section 3.6. In particular, SSAR Subsection 3.6.1.2.1 describes the pressurization response. SSAR Subsection 6.2.1.2 discusses the criteria and analysis methods for subcompartment pressurization. The pressures and temperatures resulting from these analyses are included in the loads and load combinations given in SSAR Subsection 3.8.4.

SSAR Revision: NONE

Question 220.48

Identify the particular areas that raise the concern about the capability of connection, reinforcement pattern or welded joint (Section 3.8.4 of the SSAR).

Response:

We are not aware of any areas that raise the concern about the capability of connection, reinforcement pattern or welded joint.

SSAR Revision: None

Question 230.37

Section 3.7.2.2 of the SSAR describes the importance of the mass participation from the high frequency structural modes of the stick model in the horizontal and, particularly, the vertical directions due to the rigidity of the containment internal structures. It is not clear how the contributions of the predominant high frequency modes to the structural responses were taken into account in the analyses. Particularly, provide the following information:

a. the cutoff frequencies used in the horizontal and vertical time-history analyses of the fixed base model (the case of structures founded on rock site).

b. the cutoff frequencies used in the horizontal and vertical SSI analyses using the complex frequency response analysis method, and provide the basis for the cutoff frequencies selected.

c. details of the separate seismic analysis using the coupled containment internal structures and reactor coolant loop (RCL) lumped-mass model (Page 3.7-5, first paragraph) and the difference in the response results between this separate seismic analysis and the original seismic analysis. Was this "separate analysis" done using the fixed base model for the rock site condition?

d. details for considering the high frequency effect to the vertical responses (forces and moments) of the containment internal structures (Page 3.7-5, first paragraph). Was this consideration applied only to the vertical seismic analysis of the fixed base structural model for the rock site condition?

Response:

a. The cutoff frequency used in the horizontal and vertical time-history analyses of the fixed-base model for the hard rock site is 34.0 hertz.

b. The cutoff frequencies used in the SSI analysis are 33 hertz for the soft rock site, and 15 hertz and 21 hertz for the soft-to-medium stiff soil site in the horizontal and the vertical directions, respectively.

These cutoff frequencies are selected based on the following:

- The 33 hertz cutoff frequency used in the SSI analyses for the soft rock site is in accordance with the requirement of Regulatory Guide 1.60. The 15 hertz and 21 hertz cutoff frequencies used in the SSI analyses of the soft-to-medium stiff soil site are selected based on the major composite natural frequencies of the coupled soil-structure system.

- The calculated acceleration time histories for the soft-to-medium stiff soil site are not intended to be "stand alone". Response accelerations from the hard rock, the soft rock, and the soft-to-medium stiff soil sites are enveloped for design purposes.

- Maximum member forces and nodal displacements are dominated by those modal frequencies lower than the cutoff frequencies. Therefore, responses from frequency range higher than the cutoff frequency will have very minor effects on the maximum member forces and nodal displacements.

c. The analysis referenced in the question is an earlier analysis presented in Revision 0 of the SSAR. The seismic analysis has been revised as presented in Revision 1 of the SSAR Section 3.7.2.2.

Refer to Revision 1 of SSAR Section 3.7.2.2. Member forces for the hard rock site are calculated using the fixed base combined nuclear island stick model. The mode superposition time history analysis method is used to calculate the seismic response member forces for the coupled shield and auxiliary buildings and for the steel containment vessel. The response spectrum analysis (RSA) technique is used to calculate the response member forces, of both horizontal and vertical excitations, for the containment internal structures.

For comparison purposes, seismic member forces for the containment internal structures are also calculated using the mode superposition time history analysis method and compared with the member forces from the RSA method. This comparison shows that:

- The vertical forces determined by the mode superposition time history analysis are approximately 10% to 30% of those calculated by the RSA, and
- The horizontal forces determined by the mode superposition time history analysis are approximately 10% to 30% less than those calculated by the RSA.

d. Details for considering the high frequency effect to the vertical responses (forces and moments) of the containment internal structures are described in (c) above. As described, this consideration was applied to both the horizontal and vertical seismic analysis of the fixed base structural model for the containment internal structures.

SSAR Revision: None

Question 230.38

Provide a description in the SSAR to show how the other seismic Category I structures such as containment air baffle (CAB), passive containment cooling system water storage tank (PCCSWST), and in- containment refueling water storage tank (IRWST) were included in the nuclear island seismic models (lumped-mass stick model and finite element model) (Section 3.7.2.3 of the SSAR).

Response:

Sections 3.7.2.3.1 and 3.7.2.3.2 will be revised as shown below to provide the requested information.

SSAR Revision:

- Add the following paragraph to the end of Section 3.7.2.3.2:

  The containment air baffle, presented in Section 3.8.4.1.3, is supported from the steel containment vessel at regular intervals so that a gap is maintained for airflow. It is constructed with individual panels which do not contribute to the stiffness of the containment vessel. The fundamental frequency of the baffle panels is greater than twice the fundamental frequency of the containment vessel. The mass of the air baffle is small, equal to approximately 10% of the vessel plates to which it is attached. The air baffle, therefore, is assumed to have negligible interaction with the steel containment vessel. Only the mass of the air baffle is considered and added at the appropriate elevations of the steel containment vessel stick model.

- Add the following after the third paragraph of Section 3.7.2.3.1:

  The passive containment cooling system water storage tank is represented by a lumped-mass stick model simulating the dynamic behavior of this portion of the roof structure. This lumped-mass stick model is combined with the lumped-mass stick model representing the lower portion of the shield building. In the 3-D finite element model, the lumped-mass stick model of the passive containment cooling system water storage tank is located at the center of the shield building represented using cylindrical shell elements. The lumped-mass stick model of the passive containment cooling system water storage tank is connected to the 3-D shell elements using 18 horizontal rigid beams.

  The in-containment refueling water storage tank (IRWST) is included in the 3-D FEM used in the development of the lumped-mass stick model representing the containment internal structures (CIS). Therefore, the lumped-mass stick model of the CIS includes the stiffness and mass effect of the IRWST.

Question 230.39

The following request for additional information pertains to Section 3.7.2.3.1 of the SSAR:

a. Explain how the live loads were considered in the modeling of the coupled shield and auxiliary buildings and the containment internal structures.

b. In the second paragraph of Page 3.7-6, the SSAR states that two sticks were used to represent each structure (shield building, auxiliary building or containment internal structures). The first stick represents the axial areas and the second stick represents the beam element properties other than the axial areas. It seems that this modeling technique is trying to decouple (a) the axial and bending responses, and (b) the horizontal and vertical responses. Explain and justify this modelling technique.

c. If the containment internal structures are represented by two separate sticks, explain how to couple the RCL model with the internal structural model.

Response:

a. In the modelling of the coupled shield/auxiliary buildings and the containment internal structures, expected live loads during plant operation were considered by applying a uniform load of 50 psf on all slabs and floor areas.

b. The 2-sticks modelling technique is used to properly locate the translational and rotational stiffnesses of the structures by placing the stick with axial property at the centroid and the stick with the beam properties at the shear center. By using this modelling technique, the vertical stiffness is located at the center of vertical stiffness and the shear and torsional stiffnesses are properly located at the shear center.

With the 2-sticks modelling technique, the axial and bending responses and the horizontal and vertical responses are not decoupled because of the rigid connections provided at each floor elevation. As shown in Figure 3.7.2-4 for the coupled shield and auxiliary buildings lumped mass stick model and in Figure 3.7.2-6 for the internal structures, two vertical sticks are used between floor elevations. The total mass at each floor elevation is lumped into a single mass point at the center of mass. The mass point is rigidly connected to the 2 sticks connecting to the upper floor and the 2 sticks connecting to the lower floor.

c. As stated in (b), for the containment internal structures (CIS) lumped mass stick model, all nodes within each floor elevation are rigidly connected such that "plane section remains plane" and there is zero relative displacement between the node points at the same elevation. The RCL model is coupled to the CIS stick model as follows:

1. Local stiffnesses of the RCL supports are calculated.

2. Support points of the RCL are identified and additional nodes are provided at the RCL support locations. Rigid links are added between the RCL support nodal points and the nodal points at the proper elevation in the CIS stick model.

3. The "end nodes" in the RCL model are connected to the RCL support nodal points using the equivalent support springs determined from item 1 above.

SSAR Revision: NONE

Question 230.41

a. Section 3.7.2.3.3 of the SSAR states that for soil-structure interaction (SSI) analyses, the nuclear island basemat and the periphery walls of the embedded portion of the nuclear island are represented by a three-dimensional finite element model. When the basemat was modeled, has the flexibility of the basemat been considered in the SSI analyses?

b. Evaluate the possibility of the out-of-phase interaction between the shield building, steel containment vessel and containment air baffle (Section 3.7.2.3.3 of the SSAR).

Response:

a. In the soil-structure interaction analysis, the entire nuclear island is represented by a stick model except for the basemat and the embedded portions of the exterior walls which are modelled with 3D shell elements. This model of the embedded portion models the boundary of the nuclear island, but not the flexibility of the basemat. However, considering that the basemat thickness is 6 feet and the interior walls are closely spaced, any local flexibility of the basemat in the vertical direction is negligible.

   At the 3 slab elevations (grade at 100', 82.5', and basemat at 66.5'), horizontal rigid beam elements are modelled along the exterior wall (shell elements) to simulate the stiffening effect provided by the slab to the wall. At these same elevations, horizontal rigid beams are also used to connect the shell elements with the stick model.

   At the basemat (elevation 66.5'), horizontal rigid beams are used:

   (1) at the exterior wall to simulate slab rigidity and to connect the stick model with the exterior wall (as stated above), and

   (2) to simulate the stiffening effects provided by the internal walls to the basemat.

b. The design configuration of the steel containment vessel, the containment air baffle and the shield building is shown in Figures 1.2-12 and 1.2-13.

   The steel containment vessel, presented in Section 3.8.2.1.2, is designed as an independent, free-standing structure. The bottom head is embedded in concrete, with concrete up to elevation 100 feet on the outside and elevation 108 feet on the inside. Above elevation 100 feet, seismic gaps are provided between the steel containment vessel and the shield building.

   The containment air baffle, presented in Section 3.8.4.1.3, is supported from the surface of the steel containment vessel at regular intervals. It will displace together with the containment vessel during a seismic event. A flexible connection is provided between the air baffle and the shield building roof structure. This flexible connection is designed to accommodate the differential displacement between the containment vessel and the shield building. Therefore, seismic interaction between the shield building and the containment vessel through the air baffle is negligible.

The maximum seismic displacements relative to top of basemat for the shield building and the steel containment vessel are given in tables 3.7.2-8 and 3.7.2-9 respectively. The maximum horizontal seismic displacements relative to the top of basemat, at the top of the containment vessel and the top of the shield building are 0.95 inches and 0.42 inches, respectively. The maximum relative displacements between these structures are negligible in comparison with the design gap provided, see Figures 1.2-12 and 1.2-13. There is no out-of-phase interaction between the shield building and the containment vessel/air baffle.

Structure to structure interaction between the steel containment vessel and the shield building through the common foundation during a seismic event is considered, because a coupled model connecting the nuclear island structures to the same foundation is used in the seismic analyses.

SSAR Revision: NONE

Question 230.46

Section 3.7.2.11 of the SSAR states that the seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom and, hence, additional accidental torsion is not added to the actual calculated torsional responses. According to SRP Section 3.7.2, to exclude the accidental torsion to the overall seismic responses is not acceptable to the staff. Provide justification for this deviation to the SRP.

Response:

Accidental torsion will be included in the design as described in the SSAR revision shown below.

SSAR Revision:

Revise Section 3.7.2.11 as shown below:

The seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom. ~~Hence, additional accidental torsion is not added to the actual calculated torsional responses.~~ An accidental torsional moment is included in the design of the nuclear island structures. The accidental torsional moment due to the eccentricity of each mass is determined using the following:

- Horizontal mass properties of the building stick models shown in Figures 3.7.2-4, 3.7.2-5 and 3.7.2-6,
- The enveloping value of the horizontal nodal accelerations shown in Table 3.7.2-5, 3.7.2-6 and 3.7.2-7,
- An assumed accidental eccentricity equal to ±5% of the maximum building dimensions at the elevation of the mass.

The torsional moments due to eccentricities of the masses at each elevation are combined absolutely.

Question 230.49

The following request for additional information pertains to Section 3.7.3.3 of the SSAR:

a. As described in Section 3.7.3, the structural frames and miscellaneous steel platforms are also considered subsystems. Provide detailed modeling procedures and the analysis methods used for these substructures.

b. Were the modelling procedures described in Section 3.7.3.3 also used for modelling the cable tray, HVAC and conduit systems? Clarify this section.

c. If some safety related piping systems and/or components are supported by those structural frames and miscellaneous platforms described in (a) above, discuss in detail: (1) how the structural frames and platforms were modeled together with the piping systems and components, and (2) how the potential amplification of motion through these frames and platforms were considered or are to be considered.

d. Discuss how the polar crane system was modeled, analyzed and designed.

Response:

a. The finite element models of the nuclear island structures, presented in Section 3.7.2.3, included the structural elements which may influence the global seismic response of the nuclear island. Minor structural framing and miscellaneous steel platforms, judged to have negligible effect on the global seismic response of the nuclear island, are considered to be seismic subsystems and are subjected to the modelling procedures presented in Section 3.7.3.3 and to the analysis method shown in Section 3.7.3.1.

b. Cable tray, HVAC and conduit systems, which have negligible influence in the global seismic response of the nuclear island, are considered as a seismic subsystem. These seismic subsystems, therefore, are subjected to the modelling procedure presented in Section 3.7.3.3 and to the analysis methods presented in Section 3.7.3.1.

c. The information was provided in Revision 1 of SSAR Subsection 3.7.3.8.3.

d. The polar crane is evaluated as a seismic subsystem using seismic responses at the polar crane support ring girder obtained from the seismic analysis of the nuclear island. The procedures presented in Section 3.7.3 are applicable to the modelling, analysis and design of the polar crane system. The modelling of the polar crane in the seismic analysis of the nuclear island is provided in the response to RAI 230.40.

SSAR Revision: NONE

Question 420.123

The probabilistic risk assessment (PRA) for the AP600 design assumes that software common-mode failure among instrumentation and control (I&C) cards result in an unavailability for I&C cards within subsystems of 1.2x10-6 failures per demand (f/d). Software common-mode failures within safety and control subsystems result in an unavailability of 1.1x10-5 f/d (Table E-5.3 of the PRA).

The probability of failures/demand for I&C cards is two decades lower than the value many experts agree can be demonstrated by a practical test program (10-4 f/d). The probability of failures/demand for subsystems (made up from a collection of cards and connections between cards) is at least one order of magnitude better than the practical test value, and two orders of magnitude better than the value being considered by the Nuclear Installation Inspectorate (NII) of the United Kingdom for the Sizewell B PPS (10-3 f/d).

The summary of results for the internal events Level 1 analysis at power (Section 8.2 of the PRA) show that the common cause hardware failure of I&C cards is a significant contributor to the core damage frequency for most of the at-power initiating events. The data in Table E-5.3 gives values for hardware board CMF unavailability in the range of 1.2x10-6 f/d to 4.4x10-5 f/d. A software common-mode board failure in the range of 10-4 f/d to 10-3 f/d (instead of the value of 10-6 f/d used in the PRA) could have a significant impact on the core damage frequency results for several of the initiating events given in Table 8-1.

Perform a sensitivity assessment of the effect on the core damage frequency for events studied in the PRA resulting from software common-mode failures on I&C card unavailability in the range stated above. Provide a description of the test program that Westinghouse is proposing to demonstrate a common-mode card unavailability of 10-6.

Response:

The expert opinion guideline value of 1E-04 failures per demand (f/d) rate is applied at the overall system level and not at the individual card level. The AP600 PRA uses input rates for software and hardware common-mode failure at the component level in the general range of 1E-06 f/d to 4E-05 f/d. The summated common-mode failure contribution at the system level equates to approximately 1E-04 f/d. Therefore, the AP600 PRA common-mode failure contribution is consistent with the guideline value of 1E-04 f/d.

The rate of 1.2E-06 f/d represents the software common-mode failure contribution of cards across different subsystems of the I&C system such as the protection and safety monitoring system and the plant control system. The rate of 1.1E-05 f/d represents the software common-mode failure contribution of cards across subsystems of the same type (such as engineered safety features, protection logic cabinets, and control logic cabinets). This rate represents the software common-mode failure rate that is applied across the redundant channels in the AP600 PRA. Further discussion regarding the development and support of these rates has been provided under the response to RAI 720.91. Common cause failure between the diverse actuation and diverse indication systems is considered separately, as discussed in Appendix E.3.4.6.2 of the AP600 PRA.

No tests or sensitivity study are required as the methods and rates that are applied in the AP600 PRA regarding the application of common-mode failure are consistent with industry guidelines.

PRA Revision: NONE

SSAR Revision: NONE

Question 440.49

In late 1993, Westinghouse proposed changes to the design of the AP600 automatic depressurization system (ADS) and the manner in which it will be operated. As a result of these changes, the staff concludes that Westinghouse should reevaluate the design of the ADS test facility, both in terms of hardware and configuration.

The staff understands that Westinghouse's current plans are to test one valve in each of trains 1-3, with the second valve represented by an orifice. For the first stage, the orifice is upstream of the valve, while for stage 2 and 3, the orifice is downstream of the valve. The staff is concerned with this approach because:

a. The "critical flow" behavior of an orifice is substantially different from that of a nozzle. Orifices do not "choke," although the flow rate does exhibit a limiting behavior as upstream pressure is increased. On the other hand, short nozzles do "choke." To a first approximation, a valve appears to resemble a converging-diverging nozzle more than it does an orifice. Accordingly, in the absence of a second valve in the test train, the staff recommends that Westinghouse replace the orifice with a short nozzle that has approximately the same length and minimum flow area as the valve body.

b. Previously, both valves in an ADS stage were to be opened simultaneously. Having a valve in the upstream position would maximize the upstream pressure seen by an ADS valve, and allow testing over a greater range of pressures. The new operating procedure, however, calls for the upstream "isolation" valve in an ADS stage to be opened first, and the downstream "control" valve to be opened thereafter. Thus, the configuration suggests a fixed nozzle upstream of the valve to be opened. The staff recommends that, in concert with item (a) above, the valve to be tested be positioned downstream of the nozzle simulating the open isolation valve in each stage.

Westinghouse shoud also review the test matrix for the ADS valves in light of the proposed design change to ensure that the range of test parameters still adequately covers the range of thermal-hydraulic conditions that the valves are expected to experience in the AP600 plant.

Response:

The objective of the Automatic Depressurization System (ADS) tests for design certification is to verify the overall system performance of the ADS. The tests will be conducted with the ADS valves in a fully open position. For this reason, the ADS design certification tests will be performed both with an orifice or flow nozzle that represents a valve with minimum flow area/maximum flow resistance and with no orifice or flow nozzl. to provide the maximum flow area/minimum resistance. The following responses reflect the fact that the AP600 ADS arrangement will include an isolation valve followed by a control valve in each of the ADS stage 1,2 and 3 flowpaths.

a) The ADS valve piping package has been modified to include a valve and a spool piece in each stage of the system. Stage 1 contains a spool piece upstream of a 4 inch globe valve. Stages 2 and 3 have an 8 inch gate valve positioned upstream of a spool piece. Tests will be performed for both maximum flow conditions and minimum venting conditions for each stage of the ADS. For the minimum venting tests, the stage 2 and 3 spool pieces will contain an orifice, which will simulate the flow area and resistance of a globe valve body. The ADS stage 1 spool piece will contain a flow nozzle, to simulate the flow area and resistance of a gate valve body. The orifices and flow nozzles are designed to provide the minimum flow area and pressure drop corresponding to the most restrictive ADS valve. For the maximum flow tests, the orifices and flow nozzle will be removed from the spool pieces in order to achieve the maximum flow possible with valves that have a very large flow area.

b) As described in a) above, the spool pieces both with and without orifices and a flow nozzle will be used to simulate the range of potential valve areas and resistances. These spool pieces are located at the proper positions to correspond with the gate valve followed by globe valve ADS flowpath arrangement.

Because the tests will be conducted with the ADS flowpaths fully open, the test conditions will be determined by the facility supply tank initial conditions; i.e., pressure, temperature, level, and by appropriate positioning and operation of the VAPORE facility control valve. Pre-test analyses are used to determine these conditions such that the range of test parameters adequately covers the range of thermal hydraulic conditions for the AP600.

SSAR Revision: NONE

Question 440.51

The new configuration of the ADS announced in late 1993 appears to have implications regarding the single failure assumptions for AP600 accident analyses. For most accidents, the most limiting single failure has been assumed to be one valve of the 4th stage of the ADS, resulting in the loss of venting capability in the affected train. The new configuration of stage 4 of the ADS, however, results in a single failure of an ADS valve that does not eliminate the venting capability of that entire train, which may mean that the failure of a stage 4 ADS valve is no longer the most limiting single failure for many, if not all, design basis accidents involving depressurization.

Reanalyze the Chapter 15 events involving actuation of the ADS to determine the most limiting single failure with the new ADS configuration.

Response:

Only the loss of coolant accident (LOCA) analyses among the Chapter 15 events actuate the ADS. The potential ADS single failure assumptions were reviewed in performing the LOCA analyses presented in the February 15, 1994 submittal "AP600 Design Change Description Report." The limiting small break LOCA cases from the SSAR (double-ended direct vessel injection line break and inadvertent ADS actuation case) are presented in that report, and the appropriate ADS single failure was modeled in each. No credible single failure in the passive safety systems other than those postulated within the ADS have significant impact on the Chapter 15 LOCA analyses.

The double-ended direct vessel injection (DEDVI) line break in the February 15, 1994 report represents the limiting case for safety injection. Because depressurization to the accumulator setpoint adds significant water injection capability, a single electrical failure which results in failure of a first and a third stage ADS path to open is assumed in the DEDVI transient; with the redesign of the ADS and the availabilty of the postulated break to aid in venting, the prompt achievement of IRWST injection is not a concern, as demonstrated by the NOTRUMP result and discussed in the report. The single failure resulting in the loss of a first and third stage ADS path is the limiting possible single failure for potential core uncovery for the postulated DEDVI break, and no uncovery is predicted for this transient when this single failure is modeled.

The inadvertent ADS actuation case in the February 15, 1994 report is the limiting case in terms of depressurization capability. For this case the failure of a fourth stage ADS valve remains appropriate because it still deprives the plant of more venting area than does any other assumed single active failure. The AP600 ADS design capability is demonstrated in the February 15, 1994 report result of this postulated event. As discussed in the report, IRWST injection is readily achieved with three of the four fourth stage ADS paths operational.

The SSAR small break LOCA analyses each assume the failure of a fourth stage ADS valve. This continues to be the limiting failure for the remaining SSAR small break LOCA cases. In each of these SSAR cases the ADS is actuated upon a low CMT level signal. Even though the first through third stage valve areas are smaller in the February 15, 1994 report design than in the SSAR analysis, these valves depressurize the plant more quickly because they are actuated via timers rather than CMT level signals. Just as the DEDVI break and the inadvertent ADS actuation case in the February 15, 1994 report, each of these SSAR cases will exhibit approximately the same reactor coolant system inventory at the time of fourth stage actuation. While the system pressure at fourth stage actuation time might be higher if the failure of a valve(s) in the first three stages of ADS is assumed, any adverse impact of this will be compensated for by the additional fourth stage path being available.

The SSAR results bound the small break LOCA performance of the AP600 in terms of system inventory when the February 15, 1994 report design changes are considered, whatever single active failure is postulated.

PRA Impact: None
SSAR Impact: None