Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NTD-NRC-94-4110
DCP/NRC0044
Docket No.: STN-52-003

April 28, 1994

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

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of January 26, 1994, January 27, 1994 and March 16, 1994. In addition, a revision of a previous response is included.

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 26, 1994 and January 27, 1994 letters 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.

Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

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

030067

E004

1678A

NTD-NRC-94-4110
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED APRIL 28,1994

RAI No.	Issue
220.027	Potential sources of missiles in containment
220.033	Electrical penetration assembly strength
220.037	Containment shell prebuckling stresses
220.047	Analysis methods for seismic Cat. II structures
220.049	Exclusion of Cat II structures for foundation anal
220.059	Radius and thickness of dome
220.068	Containment shell yield stress properties
230.035	Results of 2D SSI & 3D response spectrum analyses
230.047	Stability of containment vcessel during SSE
230.048	Description of "design by rule" analysis
230.051	COL commitment for reconciliation analysis
230.057	Integrity of cont. shell-shield bldg connection
440.034R01	CMT scaling tests
952.042	Pressurizer Balance Line Piping Diagrams
952.043	RELAP-5 Thermal Hydraulic Information

ATTACHMENT B
CROSS REFERENCE OF WESTINGHOUSE RAI RESPONSE TRANSMITTALS TO NRC LETTERS OF JANUARY 26, 1994 AND JANUARY 27, 1994

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
220.024	Wind-induced failure of nonsafety structures	01/26/94	04/14/94
220.025	Containment seals at transition region	01/26/94	03/24/94
220.026	Structural integrity testing of steel containment	01/26/94	03/04/94
220.027	Potential sources of missiles in containment	01/26/94	04/28/94
220.028	Loading effects of air baffle on containment	01/26/94	04/14/94
220.029	Use of (1.,0.4,0.4) method vs SRSS method	01/26/94	04/14/94
220.030	Justification for factor of safety of 1.67	01/26/94	03/24/94
220.031	Stress calculation by ASME criterion	01/26/94	03/04/94
220.032	Justification for factor of safety of 2.5	01/26/94	03/24/94
220.033	Electrical penetration assembly strength	01/26/94	04/28/94
220.034	Nonmetalic items under SA conditions	01/26/94	03/24/94
220.035	Corrosion allowance for steel containment plates	01/26/94	03/04/94
220.036	Containment shell stress analysis results	01/26/94	03/24/94
220.037	Containment shell prebuckling stresses	01/26/94	04/28/94
220.038	Axisymmetric model vs. Sandia criteria	01/26/94	03/24/94
220.039	Strains at discontinuities vs. Sandia criteria	01/26/94	03/24/94
220.040	Concrete cracking effects in seismic analysis	01/26/94	04/21/94
220.041	Soil pressure effects on embedded wall section	01/26/94	04/14/94
220.042	Design criteria for severe weather phenomena	01/26/94	03/24/94
220.043	Stability evaluations for safety-related structure	01/26/94	03/24/94
220.044	Methodology for seismic load calculations	01/26/94	03/24/94
220.045	Subcompartment global pressure/temperature effects	01/26/94	04/14/94
220.046	Use of epoxy-coated reinforcing steel	01/26/94	03/24/94
220.047	Analysis methods for seismic Cat. II structures	01/26/94	04/28/94
220.048	Capability of connection, reinforcement pattern	01/26/94	04/14/94
220.049	Exclusion of Cat II structures for foundation anal	01/26/94	04/28/94
220.050	Factor of safety for sliding & overturning	01/26/94	03/24/94
230.024	Difference between non-Cat I & non-seismic	01/26/94	03/24/94
230.025	Non-Cat I & seismic Cat II clarification	01/26/94	03/24/94
230.026	GDC for seismic Cat II	01/26/94	03/04/94
230.027	Frequency intervals in response spectra	01/26/94	03/24/94
230.028	Ground motion cross correlation coefficients	01/26/94	03/24/94
230.029	Basis for damping ratio	01/26/94	03/24/94
230.030	Basis for hard-rock, soft-rock damping values	01/26/94	03/24/94
230.031	Shear wave velocity profile for base rock	01/26/94	03/24/94
230.032	Location of input ground motion	01/26/94	03/24/94
230.033	Justification for envelope of potential sites	01/26/94	03/24/94
230.034	Use of "time history analysis"	01/26/94	03/24/94
230.035	Results of 2D SSI & 3D response spectrum analyses	01/26/94	04/28/94
230.036	SASSI code validation pachage	01/26/94	03/24/94
230.037	Cutoff frequencies of fixed base model	01/26/94	04/14/94
230.038	Seismic Cat I structures in stick model	01/26/94	04/14/94
230.039	Live loads in modeling shield & auxiliary building	01/26/94	04/14/94
230.040	Modeling of steel containment shell	01/26/94	03/24/94
230.041	Basemat in SSI analyses	01/26/94	04/14/94
230.042	Structural member forces used for design	01/26/94	03/24/94
230.043	Discrepancy between Sections 3.7.2.5 & 3.7.2.1.2	01/26/94	03/24/94
230.044	Application of 3 components of earthquake motion	01/26/94	03/24/94
230.045	Analyses for fixed base structural model	01/26/94	03/24/94
230.046	Exclusion of additional accidental torsion	01/26/94	04/14/94
230.047	Stability of containment vcessel during SSE	01/26/94	04/28/94
230.048	Description of "design by rule" analysis	01/26/94	04/28/94
230.049	Modeling procedures	01/26/94	04/14/94
952.042	Pressurizer Balance Line Piping Diagrams	01/27/94	04/28/94
952.043	RELAP-5 Thermal Hydraulic Information	01/27/94	04/28/94

Records printed: 55

Question 220.27

Provide the potential sources of a missile or sources of high pressure resulting from high-energy line break between the steel containment and the operating floor and refueling cavity walls, between the secondary shield walls and the steel containment, and between the steel containment and the shield building (Section 3.8.2 of the SSAR).

Response:

Potential sources of missiles inside the containment are discussed in SSAR Subsection 3.5.1.2. Criteria are defined to determine if certain rotating equipment or high energy systems could result in credible missiles. When the equipment is procured and detail design information is available, the equipment will be reviewed against the criteria defined in SSAR Subsection 3.5.1.2. If missiles are determined to be credible, an evaluation will be performed to confirm that such missiles do not jeopardize safe shutdown.

High energy piping is identified in SSAR Appendix 3E. These figures show the containment boundary. Subcompartments are designed for the pressure and temperature effects calculated for the postulated pipe breaks as described in SSAR Subsection 3.6.1.2.1. Table 220.27-1 lists the high energy piping (greater than 1 inch nominal diameter) inside each compartment within the containment, showing the nominal size of each line. The subcompartments are identified using the room numbers and room names given on SSAR non-proprietary Figures 1.2-4 thru 1.2-10. There is no high energy piping that can pressurize the annulus between the containment vessel and the shield building. Guard pipes are provided for the mainsteam, feedwater and steam generator blowdown containment penetrations passing through the annulus as shown in SSAR Figures 3.8.2-4. The CVS makeup piping is classified as high energy due to its design pressure but does not cause pressurization because it is at ambient temperature.

SSAR Revision: NONE

AP600

TABLE 220.27-1
AP600 SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

COMPARTMENT	LINES QUALIFIED TO LBB ASME Class 1 and 2	LINES NOT QUALIFIED TO LBB
ROOM NUMBERS 11201, 11301, 11401, 11501 STEAM GENERATOR COMPARTMENT 1	31" Hot Leg (RCS) 22" Cold Leg (RCS) 18" Surge Line (RCS) 12" Fourth stage ADS (RCS) 10" Passive RHR (RCS/PXS) 16" Feed Water (SGS) 4" Pressurizer spray (RCS) 4" SG Blowdown (SGS)	3" Purification (CVS) 2" SG Blowdown (SGS)
ROOM NUMBERS 11202, 11302, 11402, 11502 STEAM GENERATOR COMPARTMENT 2	31" Hot Leg (RCS) 22" Cold Leg (RCS) 12" Fourth stage ADS (RCS) 20", 12" Normal RHR (RCS/RNS) 16" Feed Water (SGS) 8" CL to CMT (PXS) 4" SG Blowdown (SGS)	2" SG Blowdown (SGS)
ROOM NUMBER 11205 REACTOR VESSEL NOZZLE AREA	31" Hot Leg (RCS) 22" Cold Leg (RCS) 8" Direct Vessel Injection (RCS)	None
ROOM NUMBER 11206 PXS VALVE AND ACCUMULATOR ROOM A	8" Direct Vessel Injection (RCS/PXS) 8" line from CMT (PXS) 6" line from IRWST (PXS)	2" CMT (PXS) 2" Accumulator (PXS)
ROOM NUMBER 11207 PXS VALVE AND ACCUMULATOR ROOM B	8" Direct Vessel Injection (RCS/PXS) 8" line from CMT (PXS) 6" line from IRWST (PXS)	2" CMT (PXS) 2" Accumulator (PXS)

ROOM NUMBER 11208 RNS VALVE ROOM	12", 10" Normal RHR (RNS)	None
ROOM NUMBER 11209 CVS ROOM	None	3" Purification (CVS) 2" (CVS)
ROOM NUMBER 11303 PIPE AND VALVE ROOM (BELOW PRESSURIZER)	18" RCS Surge Line (RCS) 10" Passive RHR (PXS) 4" Pressurizer spray (RCS) 4" SG Blowdown (SGS)	None
ROOM NUMBER 11403 LOWER PRESSURIZER COMPARTMENT	10" Passive RHR (PXS) 6" Pressurizer spray line (RCS)	None
ROOM NUMBER 11503 UPPER PRESSURIZER COMPARTMENT	14" ADS (RCS) 10" Passive RHR (PXS) 6" Pressurizer spray (RCS)	None
ROOM NUMBER 11300 MAINTENANCE FLOOR (LOWER COMPARTMENT)	32" Main Steam (SGS) 16" Feed Water (SGS) 10" Passive RHR (PXS) 4" SG Blowdown (SGS)	3" Purification (CVS) 2" (CVS)
ROOM NUMBER 11500 OPERATING DECK (UPPER COMPARTMENT)	32" ID Main Steam (SGS) 16" ID Feed Water (SGS) 10" PRHR (PXS) 14", 8", 4" ADS (RCS) 6" Pressurizer Safety (RCS)	None

Question 220.33

NUREG/CR-5334 reported that, during severe accident conditions, no leakage was detected from any of the three current electrical penetration assemblies (EPAs), under the following conditions (1) D. G. O'Brien EPA, 361°F, 155 psia for 10 days, (2) Westinghouse EPA, 400°F, 75 psia for 10 days, and (3) Conax EPA, 700°F, 135 psia for 10 days. However, the SSAR does not address what EPAs will be used for the AP600. Provide a commitment in the SSAR that EPS penetrating containment be at least as strong as the steel containment vessel (Section 3.8.2 of the SSAR).

Response:

The electrical penetration assemblies are described in SSAR Subsection 3.8.2.1.6 and are depicted in sheets 8 and 9 of Figure 3.8.2-4. The electrical penetration assemblies are procured as equipment and the details are dependent on the supplier. The assemblies will be qualified for the containment design basis event conditions as described in SSAR Appendix 3D. The assemblies will be procured to be similar to one of those tested by Sandia as reported in NUREG/CR-5334 and will have ultimate capacities consistent with those demonstrated in the Sandia tests. The ultimate capacity of the EPAs is primarily determined by the temperature. The maximum temperature of the containment vessel below the operating deck during a severe accident is reported in Appendix L of the PRA Report as 315°F. This is significantly below the capability of the assemblies tested from the three suppliers.

SSAR Revision: NONE

Question 220.37

Submit the pre-buckling stresses for the most highly stressed regions , and verify that stresses at buckling are within the elastic range (Section 3.8.2 of the SSAR).

Response:

The pre-buckling stresses under design basis conditions are discussed in the response to RAI 220.36 in both meridional and circumferential directions. The most highly stressed regions away from discontinuities for buckling are the knuckle area of the top head under internal pressure, the cylinder and top head under external pressure, and the base of the cylinder under safe shutdown earthquake. The pre-buckling stresses are within the elastic range for these locations.

The bottom head is embedded in the concrete base at elevation 100 feet. This leads to high circumferential stresses at the discontinuity under thermal loading associated with the design basis accident. Buckling close to the base is evaluated against the criteria of ASME Code Case N-284 using a BOSOR-5 model.

Detailed stress analysis results for the containment shell are available for staff review in the design calculations for the containment vessel.

SSAR Revision: NONE

Question 220.47

Specify the analysis methods and the design criteria for seismic Category II structures (Section 3.8.4 of the SSAR).

Response:

The analysis methods and design criteria for seismic Category II structures are described in SSAR Subsection 3.7.2.8. This subsection of the SSAR is being revised by the response to RAI 230.54.

SSAR Revisions: None

Question 220.49

The seismic Category II structures, such as the turbine building, the annex buildings I and II, and the solid radwaste building are sufficiently close to the nuclear island such that their collapse could affect the safety function of Category I structures. The structural integrity is the requirement for seismic Category II structures. Therefore, provide the reason why the seismic Category II structures are excluded for the foundation analyses (Section 3.8.5 of the SSAR).

Response:

The information requested will be provided in May concurrent with the responses to RAIs 230.54, 230.66, 230.68, and 230.73 which also relate to seismic Category II structures.

Question 220.59

Provide the radius and the thickness of the knuckle region and the dome in Section 3.8.2 of the SSAR.

Response:

Dimensions of the containment vessel are given on sheet 1 of SSAR Figure 3.8.2-1. The head is ellipsoidal with a major diameter of 130 feet and a height of 37 feet 7.5 inches. The thickness is 1.625 inches.

SSAR Revision: NONE

Question 220.68

With regard to the materials to be used for the containment shell, a stress-strain curve for SA-537 Class 2 material was presented that was reported to be obtained from Japanese data. The yield stress was shown to be 81.3 ksi. However, Table 2 of the ASME specification for SA-537 reports a minimum yield strength of 60 ksi for the same material. Clarify the correct properties to be used for design.

Response:

As described in SSAR Subsection 3.8.2.4.2.6, the containment vessel is designed using SA537, Class 2 material. The design is based on ASME specified properties, which at ambient temperature are 60 ksi yield stress and 80 ksi ultimate stress.

A typical stress strain curve is also described in SSAR Subsection 3.8.2.4.2.6. This had a yield stress of 81.3 ksi. This value was used in calculating the best estimate containment pressure for general membrane yield of the cylinder which is assumed to correspond to the loss of containment function.

SSAR Revision: NONE

Question 230.35

The following request for additional information pertains to Section 3.7.2.1.1 of the SSAR:

a. Provide the detailed comparison of the results obtained from the 2D SSI analyses and the 3D response spectrum analyses for the hard-rock site condition.

b. As described in Section 3.7.2.1.1, the structural member forces and moments are obtained from the response spectrum analysis of the finite element model for the hard-rock site, and from the SSI analysis of the stick model for the soil sites. Provide a comparison of responses from the response spectrum analyses of a stick model and a finite element model at rock site.

c. From the staff's review of Section 3.7.2.1.1 and Table 2A.17, the staff determined that the hard-rock site condition (R1) is not the governing case for the steel containment shell. Describe how the steel containment shell was analyzed for the rock site condition.

d. Provide the rationale for excluding the SB roof in the finite element model, as shown in Figure 3.7.2-1.

e. From the staff's review of Tables 3.7.2-1 through 3.7.2-4 of the SSAR, the staff determined that the AP600 nuclear island structures (except the steel containment shell) are very rigid. Some predominant frequencies are much higher than 33 Hz. Provide justification for the statement "since the shear wave velocity for the hard rock site is in excess of 8000 feet per second, the soil- structure interaction effect is negligible." This statement has also been made in Sections 3.7.2.1.2 and 3.7.2.4.

Response:

a. Maximum member forces for the hard rock (R1) case of the 2D SSI analysis are given in Table 2A-17. Maximum member forces for the hard rock analyses of the 3D stick model using the computer program BSAP are given in Table 3.7.2-11 (sheet 1). Floor response spectra for the R1 case of the 2D analyses are given in Figures 2A-29, 2A--30 and 2A-31. Floor response spectra for the BSAP hard rock analyses of the 3D stick model are given in Figures 3.7.2-29, 3.7.2-30 and 3.7.2-31. The 3D stick model was developed from the finite element model and the frequencies and modal participation of the 3D stick model and finite element model are consistent. The 2D SSI analysis was performed to establish the design soil profiles for the AP600 plant. The 3D response spectrum analysis reported in Revision 1 of SSAR Subsection 3.7.2 for the hard rock site condition was performed to obtain in-plane member forces in the individual elements of the finite element model. There were slight differences in the plant configuration considered in the 2D SSI model and the 3D models. More detailed comparison of results from the two analyses is not meaningful.

b. The response spectrum analysis described in Subsection 3.7.2.1.1 was performed only for the hard rock site and used the three-dimensional finite element models. It generated forces and moments in the various elements such as individual walls and slabs. The member forces and moments obtained in the time history three-dimensional analyses of the lumped-mass stick models (described in the last paragraph of Subsection 3.7.2.1.1) are typically the total shear force, axial force, and moment at a given elevation in the structure. A direct comparison is not available.

c. Table 3.7.2-12 shows the maximum member forces in the containment vessel stick model for the three design soil conditions (hard rock, soft rock and soft-to-medium stiff soil). These results show that the hard rock case gives the maximum forces. Table 3.7.2-6 shows the maximum absolute accelerations for the same soil conditions. The hard rock case results in the highest accelerations of the vessel, except in the node representing the polar crane where the acceleration in the east-west direction is 6% higher for the soft rock case than for the hard rock case. This is considered in design of the crane girder which uses the crane wheel loads from the polar crane design analyses. These design analyses will be reconciled by the Combined License applicant once the final design of the crane is established.

The steel containment vessel is analyzed using the shell of revolution model for the equivalent static accelerations from the SSI seismic analyses reported in SSAR Table 3.7.2-6.

The analyses of Appendix 2A are intended to select the appropriate soils cases for the 3D analyses reported in SSAR Section 3.7.2. They are not used to define the governing case for the containment vessel design. Table 2A.17 shows the seismic member forces for the containment vessel for these parametric soils analyses. This data is for a configuration in which the containment vessel was supported up to elevation 82'-6". As reported in Table 2A.15 this model had a fundamental frequency of 2.14 Hz in the east-west direction. Based on review of these results a design change was incorporated to raise concrete around the vessel to elevation 100'. This increased the fundamental frequency of the containment vessel to 7.61 Hz (see SSAR Figure 3.7.2-10). This model is included in the analyses of SSAR Section 3.7.2. The analyses of Appendix 2A are appropriate for the selection of soil conditions because the mass of the containment vessel is small compared with that of the rest of the nuclear island.

d. A lumped-mass stick model of the shield building roof structure was constructed and coupled with the finite element model and the stick model of the coupled auxiliary and shield buildings. The stick model of the shield building roof structure was included in all seismic analyses performed. The lumped-mass stick model of the shield building roof was not shown in Figure 3.7.2-1 to maintain visual clarity of the finite element model.

e. For the hard rock site, a fixed-base analysis was performed based on the acceptance criteria specified in Revision 2 of SRP 3.7.2, "For structures supported on rock or rock-like material, a fixed base assumption is acceptable. Such materials are defined by a shear wave velocity of 3500 feet per second or greater at a shear strain of 10^{-3} percent or smaller ...etc." Furthermore, as noted in Section 3.7.2.2, the total cumulative mass of the nuclear island participating in the seismic response, up to the frequency limit of 34 Hz, constitute 90, 90 and 83 percent of the total mass, excluding the building mass within the embedded portion. The predominant frequencies of the coupled auxiliary/shield buildings and the steel containment vessel are below 34 Hz. The relatively rigid containment internal structures, coupled to the other flexible structures on a common basemat, are expected to have negligible effect in the over all soil-structure interaction responses of the nuclear island. Therefore, for the hard rock site, only a fixed-base analysis is required.

SSAR Revision: NONE

Question 230.47

Section 3.8.2.1.2 of the SSAR states that the vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by friction and bearing. This statement implies that there are no shear studs or anchors between the internal structures, steel containment vessel and reinforced concrete basemat. Provide an analysis to demonstrate the dynamic stability of the containment vessel during an SSE event or a seismic margins earthquake.

Response:

There are no shear studs or anchors between the internal structures, steel containment vessel and reinforced concrete basemat. The dynamic stability of the containment vessel was evaluated for a SSE event using a conservative friction coefficient of 0.4 at the concrete/steel interface. The factors of safety computed are equal to 2.5 and 3.0 against overturning and sliding, respectively.

The evaluation is graphically presented in Figures 230.47-1 and 230.47-2 using the following input data:

- The total dead weight of the steel containment vessel (SCV), the containment internal structures (CIS), and major equipment,

$$W = 61{,}266 \ Kips$$

- The peak SSE response forces and moments of the SCV and the CIS for the three design soil profiles are enveloped. The enveloped SSE response forces and moments of the SCV and the CIS are assumed to occur simultaneously and combined at Elevation 66'-6". Using the (1.0, 0.4, 0.4) method, the combined SSE response forces and moments used in the evaluation are:

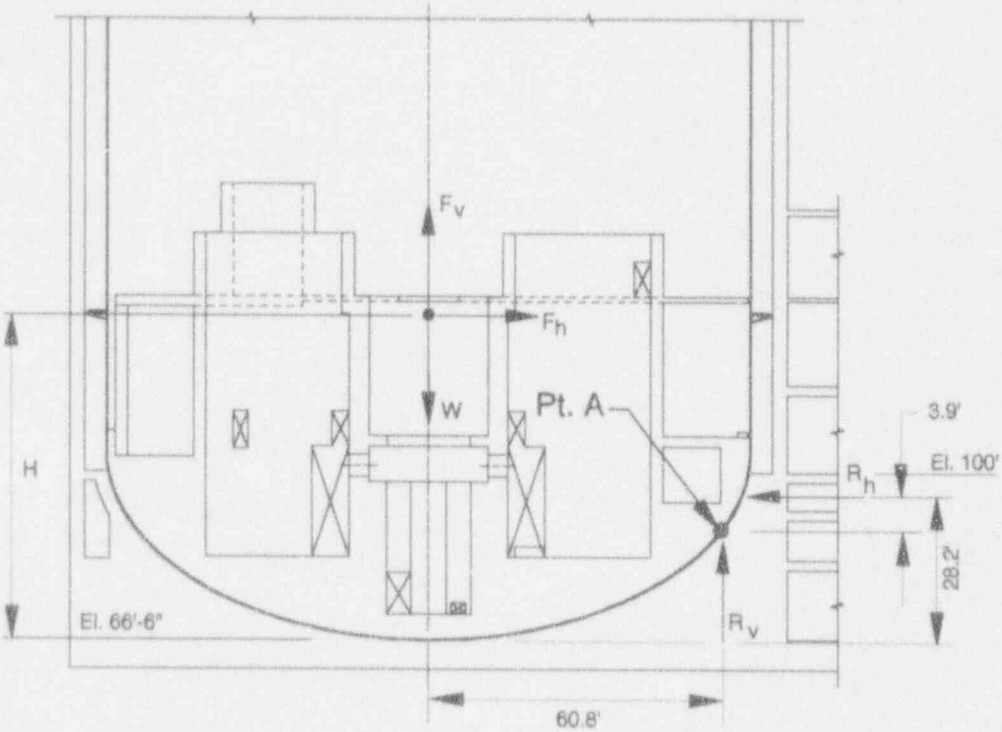
$$F_v = 9{,}070 \ Kips$$

$$F_h = 23{,}045 \ Kips$$

$$M = 1{,}506{,}770 \ K\text{-}ft$$

The dynamic stability analysis for the safe shutdown earthquake shows a large safety factor of 2.5 against overturning. Based on an analysis similar to that shown in Figure 230.47-1, incipient overturning would be predicted at a ground input level of 0.79g for an earthquake having a similar response spectrum shape as the safe shutdown earthquake. This would not represent failure since small lift-off would not prevent safe shutdown. Dynamic stability is therefore assured for the seismic margins review level earthquake.

SSAR Revision: NONE



$$H = \frac{M}{F_h} = \frac{1,506,770}{23,045} = 65.4\ ft$$

Assumed Overturning about Point "A", the low point of the minimum bearing area required to support the dead weight + SSE loads at the concrete cradle.

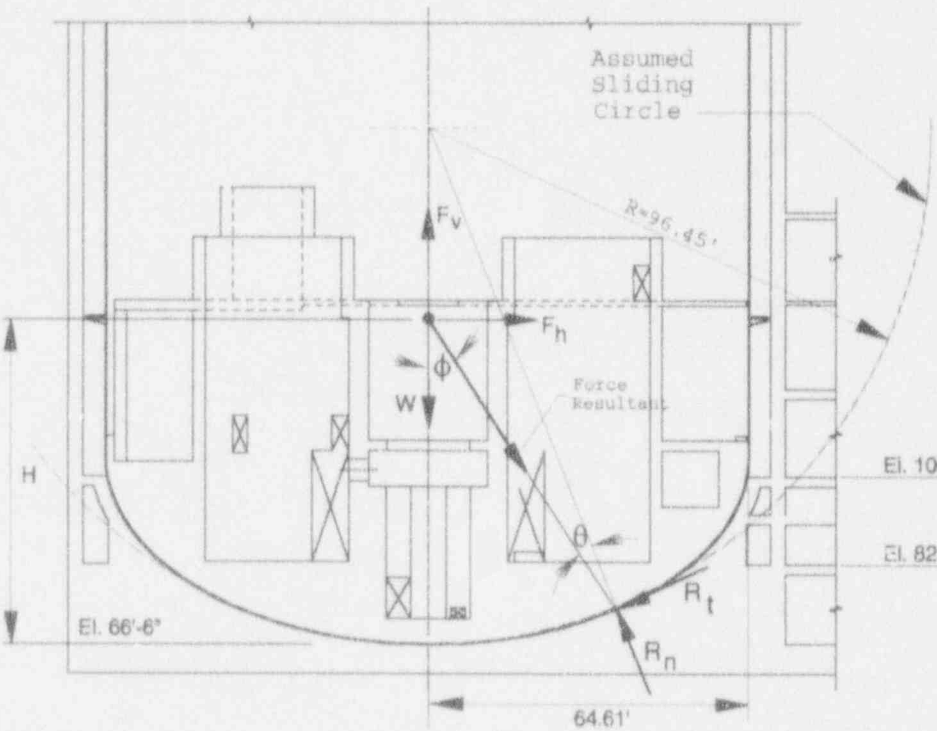
$$\text{Overturning Moment} = F_h * 41.1 + F_v * 60.8 = 1,498,606\ K\text{-}ft$$

$$\text{Resisting Moment} = W * 60.8 + R_h * 3.9 = 3,812,417\ K\text{-}ft$$

$$\text{Factor of Safety} = \frac{\text{Resisting Moment}}{\text{Overturning Moment}} = 2.5 \geq 1.1$$

Figure 230.47-1 Factor of Safety To Resist Overturning

Westinghouse



$$H = \frac{M}{F_h} = \frac{1{,}506{,}770}{23{,}045} = 65.4\ ft$$

$$\phi = Tan^{-1}\frac{F_h}{W - F_v} = \frac{23{,}045}{52{,}196} = 23.82\ ^\circ$$

$$\theta = Sin^{-1}\frac{(96.45 - H) \times Sin(180 - \phi)}{96.45} = 7.47^\circ$$

Reactions, normal and tangential, to assumed sliding circle:

$$R_t = F_r\ Sin\ \theta = 7{,}422\ Kips\ , \qquad R_n = F_r\ Cos\ \theta = 56{,}572\ Kips$$

$$Factor\ of\ Safety = \frac{(R_n \times \mu)}{R_t} = \frac{56{,}572 \times 0.4}{7{,}422} = 3.0 \geq 1.1$$

Figure 230.47-2 Factor of Safety To Resist Sliding

Question 230.48

Provide a detailed description regarding the "design by rule" analysis method in the SSAR and discuss what activities are underway for adoption of this method by a consensus code or standard (Section 3.7.3.1 of the SSAR).

Response:

The "design by rule" method for small bore piping is based on EPRI Report NP6628 as described in SSAR, Revision 1, subsection 3.7.3.8.2.2.

SSAR Revision: NONE

Question 230.51

The seismic design bases for the AP600 standard design are essentially defined by a safe-shutdown earthquake (SSE) with peak ground acceleration of 0.3g and the soil profiles characterized in Section 2 of the SSAR (assuming no liquefaction and fault displacement at the plant site). If these generic design bases are not satisfied, design certification will no longer hold, and site-specific analyses and evaluations must be performed in accordance with the SRP. Provide a COL commitment, in the SSAR, to perform this reconciliation analysis.

Response:

Chapter 2 of the SSAR defines the site-related parameters for which the AP600 plant is designed. These parameters envelope most potential sites in the United States. This chapter discusses how the specific interfaces are to be used in the AP600 design. The Combined License applicant is responsible to demonstrate that the selected site meets the interface. Section 2.5 provides seismology criteria by which acceptability may be demonstrated.

For cases where a site characteristic exceeds the envelope parameter, it is the responsibility of the Combined License applicant referencing the AP600 to demonstrate that the site characteristic does not exceed the capability of the design. Thus, it is not necessary or appropriate to include in the design certification of the AP600, requirements and commitments for applicants with sites that do not meet the site characteristics for the standard design.

SSAR Revision: NONE

Question 230.57

One of the drawings displayed shows a physical connection between the containment shell and the shield building near the upper spring line. If the function of the connection is important, its integrity should be evaluated when the connection is subject to relative displacement (between the containment shell and the shield building) during a seismic event.

Response:

The shield building, containment vessel and air baffle are shown on the General Arrangement sections in SSAR Figures 1.2-12 and 1.2-13. These figures show the containment air baffle and the pipe strut attaching the air baffle to the containment vessel. They also show the flexible seal between the air baffle attached to the containment vessel and the portion of the air baffle attached to the shield building. The only physical connection between the shield building roof and its attached structures, and the containment vessel and its attached structures, is the flexible seal. The upper air baffle is attached to the shield building roof. The lower portion is attached to the steel containment. The flexible seal, at elevation 236', 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.

SSAR Revision: NONE

Question 440.34

In its January 19, 1993 response to a question on the core makeup tank (CMT) tests dated July 21, 1992, Westinghouse states that "[t]here will be no formal scaling report for the CMT tests. Since the CMT test is a separate effects test, ...the boundary conditions for the test can be separately controlled....[thus] there is no need for a detailed scaling report." While it is true that some conditions can be more closely controlled in a separate effects test environment than in a systems test environment, once the test starts, the conditions that evolve, such as natural convective flows and temperature distributions, are governed by the physical processes occurring during the test itself, including heat transfer to and from the CMT and depressurization of the test loop (simulating ADS actuation). If the geometry of the test article is substantially different from the prototypic component, the thermal-hydraulic behavior of the two could be different. This is the case with the CMT test. The component in the plant has an aspect ratio (height to diameter) of about 1.7, whereas the test article has an aspect ratio of about 5. Multi-dimensional behavior in the actual CMT, including stratification, internal recirculation, and energy transport, may not be adequately represented in the test article, which looks much more one-dimensional. This behavior may have a substantial impact on the response of the CMT during an accident. Therefore, provide a detailed scaling analysis showing that the thermal-hydraulic phenomenology observed in the CMT test can be directly related to that expected in the plant component during the range of events where the CMT is expected to be in operation.

Response (Revision 1):

Westinghouse Topical Report, WCAP-13963, "Scaling Logic for the Core Makeup Tank Test," Revision 0, provides the requested scaling analysis for the Core Makeup Tank test facility. The report was provided to the NRC via Westinghouse letter NTD-NRC-94-4068, dated February 22, 1994.

SSAR Revision: None

Question 952.42

The information provided for the pressure balance line (PBL) from the pressurizer to CMT is incomplete for the short angled run upstream of the pressurizer. Provide piping diagrams in a plan view from the reactor vessel center line to the west, and show the PBL piping from the reactor vessel center line to the pressurizer.

Response:

Subsequent to the receipt of this RAI, an AP600 design change was implemented which removed the pressure balance line from the pressurizer to the CMT. Therefore, the requested piping diagrams are no longer relevant to the AP600 design. This change has been discussed with NRC staff and will be documented in the next SSAR revision. A design change report documenting the change and impacts on plant safety analyses will be submitted by June 30, 1994.

SSAR Revision: NONE
PRA Revision: NONE

Question 952.43

Provide the following information:

Steam Generator

a. Hydraulic diameter at the tube support plates.

b. Hydraulic diameter at the downcomer annulus obstructions.

c. Steam drier metal volume.

d. Inner diameter of the feedwater J-nozzles.

e. Feedwater line piping diagrams.

f. Steam line piping diagrams.

g. Feedwater level control system information.

h. Turbine stop valve area.

i. Post-trip T-avg trending information (turbine bypass valve control).

In the interim plant deck developed by the staff's contractor, parameters that were developed from knowledge of the steam generators typical of existing PWRs, and from knowledge of the changes that will be incorporated in the AP600 design were used. Although the NRR interim deck should provide a close approximation of the AP600 design response, no quality assurance for the deck that is traceable to a primary source (i.e., WEC engineering drawings or other documentation of the AP600 specific information) can be established without the requested information.

Other

j. Normal RHR piping diagrams and pump curves.

k. CVCS makeup/letdown system information.

l. Pumps curves, if different from those in the Westinghouse COBRA-TRAC workbook previously transmitted to INEL.

m. Pressurizer inlet nozzle retaining basket-number of holes (known hole diameter).

n. PRHR drawing illustrating the configuration and dimensions of the baffle surrounding each tube bundle.

o. Plot of void versus reactivity for the core.

Response:

Steam Generator

a. Hydraulic diameter at the tube support plates

The hydraulic diameter at the tube support plates is 0.189 inches.

b. Hydraulic diameter at the downcomer annulus obstructions

The hydraulic diameter of each downcomer annulus obstruction is identified below:

Height above tube sheet	Hydraulic Diameter
353.82 in	3.36 in
312.56 in	3.97 in.
292.50 in	5.44 in.
273.56 in	4.39 in.
234.56 in	4.39 in.
195.56 in	4.39 in.
156.56 in	4.39 in.
117.56 in	4.39 in.
78.56 in	4.39 in.
39.56 in	4.71 in.
20.00 in	5.19 in.

c. Steam drier metal volume

The steam drier metal volume is approximately 90,000 cubic inches with a metal mass of approximately 25500 lbs.

d. Inner diameter of the feedwater J-nozzles

The AP600 steam generator uses spray nozzles not J-nozzles. Each of the 34 spray nozzles, located on top of the feedwater ring, has 130 holes of 0.25 inch diameter. A sketch of the spray nozzles was provided via Westinghouse letter NTD-NRC-94-4108.

e. **Feedwater line piping diagrams**

Feedwater line piping sketches were provided as an enclosure to Westinghouse letter NTD-NRC-94-4108, dated 4/29/94. The piping sketches represent preliminary routings. The details are subject to change as a result of ongoing design activities.

f. **Steam line piping diagrams**

Main steam line piping sketches were provided as an enclosure to Westinghouse letter NTD-NRC-94-4108, dated 4/29/94. The piping sketches represent preliminary routings. The details are subject to change as a result of ongoing design activities.

g. **Feedwater level control system information**

The requested information can be found in SSAR Section 7.7.1.8. The logic is shown on Figure 7.2-1, Sheets 25 and 26.

The steam generator level control logic is similar to the Advanced Digital Feedwater Control System (ADFCS) which has been implemented at several Westinghouse nuclear plants, including Catawba, Diablo Canyon, and Prairie Island.

h. **Turbine stop valve area**

The AP600 has four turbine throttle (stop) valves. Each throttle valve has a nominal throat area of 380 square inches.

i. **Post-trip T-avg trending information (turbine bypass valve control)**

The requested information can be found in SSAR Section 7.7.1.9. The logic is shown on Figure 7.2-1, Sheet 22.

Initially following the reactor trip, the Plant Trip mode of steam dump control will be used. This control mode is described in detail in SSAR Section 7.7.1.9.2. This mode of control is no different than that at operating Westinghouse nuclear plants.

After the plant conditions stabilize at no-load following a reactor trip, the operator is instructed to switch control modes to Pressure control. This mode is described in SSAR Section 7.7.1.9.3. For holding conditions at no-load, this mode is no different than that used on present Westinghouse nuclear plants. For plant cooldown, the operator will input a desired plant cooldown rate; this will be converted into a pressure setpoint for use by the control system (see SSAR section 7.7.1.9.3).

Other

j. Normal RHR piping diagrams and pump curves

The AP600 Normal Residual Heat Removal System (RNS) system distribution and pump performance curve were provided in Westinghouse letter NTD-NRC-94-4108, dated 4/29/94. This letter provides calculated flow path resistances throughout the RNS rather than piping diagrams. The plant orifices will be sized to obtain the desired overall system resistance, and thus flow performance, once the piping layouts are finalized.

k. CVCS makeup/letdown system information

The chemical and volume control system operates to provide reactor coolant system purification, makeup and letdown. At a constant power level, the CVS purification loop operates as a closed loop around the reactor coolant pumps. The CVS makeup pumps and the letdown lines to the liquid radwaste system are not normally operating. The following paragraphs outline the makeup and letdown functions of the CVS. Refer to SSAR subsection 9.3.6 for additional information on the CVS.

On a low-pressurizer level signal (relative to the programmed level) one of the chemical and volume control system makeup pumps starts automatically to provide makeup at a controlled rate of 100 gpm. The chemical and volume control system makeup is also controlled to within a pressurizer level band following receipt of a core makeup tank actuation signal. One makeup pump is started when level reaches the low end of the band and is stopped when level reaches the high end of the pressurizer level band. This prevents pressurizer overfill or pressurizer safety valve lift on a best estimate basis. It also provides for reactor coolant system makeup which reduces the chance that the core makeup tanks will drain to the automatic depressurization system setpoint.

The chemical and volume control system letdown, when required, is taken out of the purification loop at a point downstream of the reactor coolant filters. On a high pressurizer level the letdown orifice isolation valves automatically open to divert flow to the liquid radwaste system. These valves automatically close on a containment isolation signal, high liquid radwaste system degasifier level, or a low pressurizer level signal.

l. Pump curves, if different from those in the Westinghouse COBRA-TRAC workbook previously transmitted to INEL

The AP600 RCP pump curves were transmitted to the NRC via Westinghouse letter ET-NRC-93-4013, dated 11/12/93.

m. Pressurizer inlet nozzle retaining basket-number of holes (known hole diameter).

The pressurizer inlet nozzle retaining basket has 2520 holes of 0.375 inch diameter.

n. **PRHR drawing illustrating the configuration and dimensions of the baffle surrounding each tube bundle**

The drawing was provided as an attachment with letter NTD-NRC-94-4108 dated 4/29/94.

o. **Plot of void versus reactivity for the core**

The following specifies the reactivity versus density curve which is applicable to Westinghouse cores, including AP600:

$$\frac{d\delta k}{d\rho} = A + B\rho + C\rho^2$$

where δk is the reactivity insertion and ρ is the spatial moderator density, gm/cc

The shape of this curve (defining constants B and C) is determined by specifying the changes in $d\delta k/d\rho$ from 0.8 to 0.4 gm/cc and from 0.8 to 0.6 gm/cc as 0.37 and 0.09 respectively. The vertical position of the curve (constant A) is zero.

SSAR Revision:

k. The fourth paragraph of section 9.3.6.3.1 (Chemical and Volume Control System Makeup Pumps) will be revised as follows.

One makeup pump is started on a core makeup tank actuation signal. ~~The makeup pumps are started on a safety injection signal.~~ The makeup ~~These~~ pumps provide an additional injection source to contribute to the overall reliability of the makeup function during accident conditions.

The second paragraph of section 9.3.6.4.5 (Accident Operation) will be revised as follows.

One ~~The~~ chemical and volume control system makeup pump~~s are~~ is initiated upon receipt of a ~~safety injection signal~~ core makeup tank actuation signal. Although these pumps do not provide a safety function, they are available to provide reactor coolant system makeup and pressurizer auxiliary spray as an additional means to improve reliability of the makeup function during accident conditions.

The eighth bulleted paragraph of section 9.3.6.7 (Instrumentation Requirements) will be revised as follows.

- ***Makeup pump control*** - The makeup pumps controls are located in the main control room. On a low-pressurizer level signal (relative to the programmed level) one of the chemical and volume control system makeup pumps start automatically to provide makeup. The operating pump automatically stops when the pressurizer level increases to the correct value. During reactor coolant system boron changes (fuel depletion, startups, shutdowns, and refueling), the operator will start one of the makeup pumps after selecting the desired amount of boric acid.

One chemical and volume control system makeup pump will start upon receipt of a core makeup tank actuation signal. The chemical and volume control system makeup is also controlled to within a pressurizer level band following receipt of a core makeup tank actuation signal. One makeup pump is started when level reaches the low end of the band and is stopped when level reaches the high end of the pressurizer level band. The stop signal prevents pressurizer overfill or pressurizer safety valve lift on a best estimate basis. The start of the makeup pump also provides for reactor coolant system makeup which reduces the chance that the core makeup tanks drain to the automatic depressurization system setpoint.

The operators can start the second makeup pump in case the core makeup tank drain down is approaching the automatic depressurization system setpoint. The pressurizer level control setpoints at zero power (without a core makeup tank actuation) are greater than the post core makeup tank level stop (approximately 23 percent of the span).