Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NTD-NRC-94-4276
DCP/NRC0195
Docket No.: STN-52-003

August 26, 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. In addition, revisions of responses previously submitted are provided.

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

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

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

Brian A. McIntyre

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

/nja

Enclosure

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

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NTD-NRC-94-4276
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED AUGUST 26, 1994

220.33 R1
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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. Their performance under severe accident conditions is described in SSAR Subsection 3.8.2.4.2.5. 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:

Revise the last paragraph of Subsection 3.8.2.4.2.5 as follows:

Electrical penetrations have a pressure boundary consisting of the sleeve and an end plate containing a series of modules. The pressure capacity of these elements is large. Tests at pressures and temperatures representative of severe accident conditions are described in NUREG/CR-5334 (Reference 5), where the Westinghouse penetrations were irradiated, aged, then tested to 75 psia at 400°F. Other electrical penetration assemblies were tested to higher pressures and temperatures. The penetration assemblies for the AP600 are similar to one of those tested by Sandia as reported in NUREG/CR-5334. These tests showed that the electrical penetration assemblies would withstand severe accident conditions.

Question 231.23

In the November 30, 1992 response to Q231.6 regarding the lateral earth pressure loads, Westinghouse states that the seismic Category I retaining structures and below grade exterior walls are designed for the worst case enveloping the lateral earth pressure, and that the SSAR will be suitably revised. Westinghouse's response does not clearly address the fact that the lateral earth pressures along the walls of the NI are a function of the lateral extent and character of the backfill soils. Based on the above,

a. Specify, in the SSAR, acceptable ranges of backfill properties (such as compacted soil density, minimum acceptable degree of compaction, range of sizes, etc.) for backfill soils to ensure that the design is adequate, and

b. Justify the use of the Mononobe-Okabe (MO) method for calculating the lateral soil loads on walls of the NI where wall movements relative to the surrounding soil may not develop failure strains in the soil.

Response:

a. The design of the nuclear island is not influenced by backfill properties. Backfill material will not be used against the exterior walls of the nuclear island structures. The excavation will have a vertical face as described in the following revision to the SSAR.

b. Please see the response to RAI 220.41 for a discussion of the method for calculating the lateral soil loads.

SSAR Revision:

Add the following Subsection to the SSAR:

2.5.1 Excavation and Backfill

Excavation in soil for the nuclear island structures below grade will use a soil nailing method. Soil nailing is a method of retaining earth in-situ. As the nuclear island excavation progresses vertically downward, holes are drilled horizontally into the adjoining undisturbed soil, a metal rod is inserted into the hole, and grout is pumped into each hole to fill the hole and to anchor the "nail" rod.

As approximately each five feet depth of the nuclear island excavation is completed, nominal eight to ten inch diameter holes are drilled horizontally through the vertical face of the excavation into adjacent undisturbed soil. These "nail" holes, spaced horizontally and vertically on five to six feet centers, are drilled slightly downward at fifteen degrees to the horizontal. A "nail", normally a one inch diameter metal bar/rod, is center located for the full length of the hole. The nominal length of soil nails are 60% to 70% of the wall height, depending upon soil conditions. The hole is filled with grout to anchor the rod to the soil. A metal face plate is installed on the exposed end of the rod at the excavated wall vertical surface. Welded wire mesh is hung on the wall surface for wall

reinforcement and secured to the soil nail face plates for anchorage. A 4,000 psi to 5,000 psi non-expansive pea gravel shotcrete mix is blown onto the wire mesh to form a nominal four to six inch thick soil retaining wall. Installation of the soil retaining wall closely follows the progress of the excavation and is from the top down, with each wire mesh-reinforced, shotcreted wall section being supported by the soil "nails" and the preceding elevations of soil nailed wall placements.

Soil nailing as a method of soil retention has been successfully used on excavations up to 55' deep on projects in the US. Soils have been retained for up to 90' in Europe. The state of California CALTRANS uses soil nailing extensively for excavations and soil retention installations.

The specific soil nailing system is based on actual soil conditions, site conditions, and applied construction surcharge loads. The design of the external walls below grade does not take any credit for the soil nails. Since the exterior walls below grade are designed for a range of soil conditions, including hard rock and soft rock, soil nailing has no effect on the overall results of the SSI analysis.

The soil nailing method produces a vertical surface down to the bottom of the excavation and is used as the outside forms for the exterior walls below grade of the nuclear island. Concrete is placed directly against the vertical concrete surface of the excavation.

For excavation in rock, four to six inches of shotcrete are blown on to the rock surface. The concrete for the exterior walls is placed against the shotcrete. The shotcrete contains a crystalline waterproofing material as described in Subsection 3.4.1.1.1.

Revise Subsection 3.4.1.1.1 as shown below:

3.4.1.1.1 Protection from External Flooding

The probable maximum flood for the AP600 has been established at less than the finished grade as discussed previously in Section 2.4. The probable maximum flood results from site specific events, such as river flooding, upstream dam failure, or other natural causes.

Flooding does not occur from the probable maximum precipitation. Water from roof drains and/or scuppers, as well as runoff from the plant site and adjacent areas, is conveyed to catch basins, underground pipes, or directly to open ditches by sloping the tributary surface area. The site is graded to offer protection to the seismic Category I structures.

The high ground water table interface is at two feet below the grade elevation, as discussed previously in Section 2.4.

~~The seismic Category I structures which are located below grade elevation are protected against flooding by waterproofing membranes and waterstops. Waterproofing membranes are installed on horizontal and vertical exterior surfaces below grade. Waterstops are installed in exterior construction joints below grade.~~

~~Performance criteria for the waterproofing membranes and waterstops are based on the following considerations:~~

- ~~Interaction with the concrete throughout the lifetime of the plant~~
- ~~Ability to withstand the maximum hydrostatic pressure~~
- ~~Ease of installation, having minimum interference during the construction operations~~

- ~~Resistance to attack by soil bacteria~~

- ~~Weathering action~~

- ~~Low permeability~~

- ~~Capability of withstanding movements under seismic conditions~~

- ~~Integrity when subjected to plant radiation.~~

The seismic Category I structures below grade are protected against flooding by waterstops and a waterproofing system. The waterproofing system is provided by the introduction of a cementitious crystalline waterproofing additive to the nailed soil retention wall shotcrete or to the shotcrete applied to the rock surface as described in Subsection 2.5.1. For the horizontal surface under the basemat, the cementitious crystalline waterproofing additive is added to the mud mat. The waterproofing additive is a unique chemical treatment added to the concrete at the time of batching and consists of portland cement, very fine silica sand, and various active proprietary chemicals. The active chemicals react with the moisture in fresh concrete, and the byproducts of cement hydration cause a catalytic reaction generating a non-soluble crystalline formation of dendritic fibers throughout the pores and capillary tracts of the concrete. The concrete is thus sealed against penetration of water or liquid.

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: (Revision 1)

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~~ for the SSI analyses using the complex frequency response analysis method are 33 hertz in the horizontal and vertical directions for the soft rock site, and 15 and 21 hertz in the horizontal and vertical directions, respectively, for the soft-to-medium stiff soil site ~~in the horizontal and 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.~~

These cutoff frequencies for the SSI analyses are selected based on the following reasons:

(1) The 33 hertz cutoff frequency used in the soft rock case follows the requirement in NRC Regulatory Guide 1.60.

(2) The 15 hertz and 21 hertz cutoff frequencies used in the soft-to-medium stiff soil case are selected based on the major composite natural frequencies of the soil-structures system. The calculated acceleration responses for the soft-to-medium stiff soil case are not intended to be "stand alone" in the design of AP600. Response accelerations from the hard rock, soft rock, and the soft-to-medium stiff soil cases are enveloped for design purposes. In the frequency range between the cutoff frequency and 33 hertz, acceleration responses from the soft-to-medium stiff soil case are not controlling.

(3) The maximum member forces and nodal displacements for the soft rock and the soft-to-medium stiff soil cases are calculated by converting responses in the frequency domain into responses in the time domain. Therefore, these responses are dominated by contributions from the low frequency ranges. Furthermore, the complex frequency response analysis method used has considered the entire model mass. Therefore, the "missing mass" adjustments necessitated in mode superposition analyses technique are not applicable. It is concluded that the SSI responses from the frequency range higher than the cutoff frequencies have negligible effects on the calculated maximum member forces and nodal displacements within each of the soft rock and the soft-to-medium stiff soil cases.

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.~~

For the hard rock case, member forces at the coupled auxiliary and shield buildings are calculated using the mode superposition time history technique including all modes up to 34 hertz. Because the mass participation of the coupled auxiliary and shield buildings is less than 90%, the member forces from the time history analysis are verified by comparing with equivalent member forces computed using the response spectrum analysis method with the AP600 design response spectra. The response spectrum analysis is performed using the double sum modal combination and the SRSS co-directional combination method, in conjunction with a 34 hertz frequency cutoff and consideration of "all missing mass". This comparison shows that the two analytical methods produce compatible force responses.

In addition, the combined fixed-base nuclear island model is reanalyzed using a 64 hertz cutoff frequency to include a minimum 90% mass participation of the coupled auxiliary and shield buildings. The procedures and models used in this mode superposition time history analysis are the same as those used to establish the design member forces, except the cutoff frequency is 64 hertz instead of 34 hertz, and the solution time step used is 0.0025 second instead of 0.005 second. For the coupled auxiliary and shield buildings model, the cumulative participating masses are 99%, 99%, and 94% for the NS, EW, and Vertical directions, respectively. The member forces from this analysis (Case 2 responses) are compared with design member forces (Case 1 responses) and are shown in Table 230.37-1.

From the comparison in Table 230.37-1, responses are shown to be closely matched except for vertical (axial) forces at the two bottom elevations where results including higher mass participation are larger. This mismatch is due to the high vertical natural frequencies at the lower portion of the structure near the fixed boundary condition; as a result, a portion of the nodal mass did not participate in the modal time history analysis with 34 hertz cutoff frequency. However, these vertical forces are conservative since they are smaller than those calculated using the response spectrum analysis method as described above.

In the fixed-base combined nuclear island lumped-mass stick model used for the hard rock site, the steel containment vessel stick and the containment internal structures stick are connected to the coupled auxiliary and shield buildings stick through "rigid" beam elements (representing concrete slabs) at elevations 100' and 82.5' respectively. Since these slabs are connected to the exterior walls and all exterior walls at and below grade (elevation 100') are supported by the hard rock site through fixities in boundary conditions, stick-to-stick interaction for the hard rock case is judged to be negligible. Furthermore, the coupled stick model was used in the fixed-base analyses where the various stick models are connected as described above. Any stick-to-stick interaction effect is captured automatically in the responses. Specifically, they are captured in the results from the mode superposition time history analysis with 64 hertz cutoff frequency and from the above response spectrum analysis which considered all "missing mass".

SSAR Revision: NONE

Table 230.37-1
Comparison of Seismic Forces and Moments
Coupled Auxiliary and Shield Buildings
Hard Rock Site Condition

Elev. (ft)	Axial Force ($x10^3$ kips)		N-S Shear ($x10^3$ kips)		E-W Shear ($x10^3$ kips)		Torque ($x10^3$ k-ft)		N-S Moment ($x10^3$ k-ft)		E-W Moment ($x10^3$ k-ft)	
	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2	Case 1	Case 2
307									25	25	32	31
	1.67	1.69	2.65	1.69	2.49	2.48	5	5				
297									73	73	88	85
	3.69	3.73	5.47	5.42	5.07	5.06	11	11				
284									190	190	223	220
	7.09	7.19	9.33	9.25	8.47	8.48	27	27				
272									391	391	443	443
	11.17	11.35	9.33	9.25	8.47	8.48	52	52				
246									631	631	659	656
	13.18	12.98	14.45	14.47	14.64	14.76	53	54				
241									697	697	722	719
	14.29	13.84	15.54	15.45	15.83	15.91	54	54				
230									843	843	892	889
	16.46	15.49	17.58	17.62	17.85	17.90	68	66				
210									1135	1134	1234	1233
	19.52	17.53	20.37	20.67	20.14	20.17	93	92				
180									1807	1800	1893	1884
	22.17	19.58	24.03	24.07	22.66	22.97	785	790				
161									2341	2313	2316	2305
	23.62	21.49	26.41	26.08	25.42	25.78	953	954				
153									2513	2484	2428	2466
	25.26	24.43	29.41	28.64	29.63	29.78	695	700				
135									2981	2942	2940	2949
	27.92	30.41	33.81	33.19	36.41	36.24	958	954				
118									3535	3497	3422	3429
	29.94	36.24	36.62	37.90	41.13	41.55	1173	1189				
100									4200	4359	3937	4154

Note: The seismic force and moment responses are determined using the mode superposition time history technique with 3 components of earthquake input simultaneously. Case 1 responses are computed using $\Delta t = 0.005$ second and cutoff frequency = 34 hz. Case 2 responses are computed using $\Delta t = 0.0025$ second and cutoff frequency = 64 hz.

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: (Revision 1)

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 solid elements and shell elements, respectively. 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. The location of the rigid beam elements at elevations 82.5' and 100' are shown in Figures 230.41-1 and 230.41-2, respectively.

   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.

   The location of the rigid beam elements at elevation 66.5' are shown in Figure 230.41-3.

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.

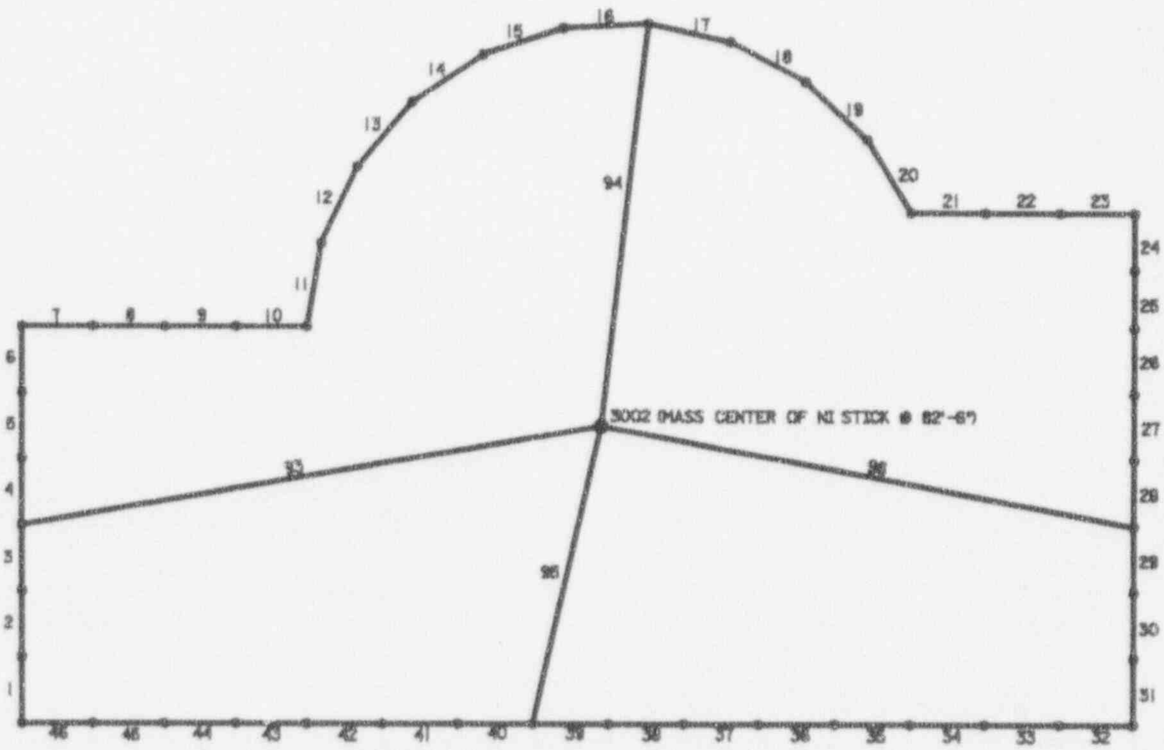
The containment air baffle, presented in Section 3.8.4.1.3, is supported from the surface of the steel containment vessel at regular intervals. It will displace together with the containment vessel during a seismic event. A flexible connection is provided between the air baffle and the shield building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

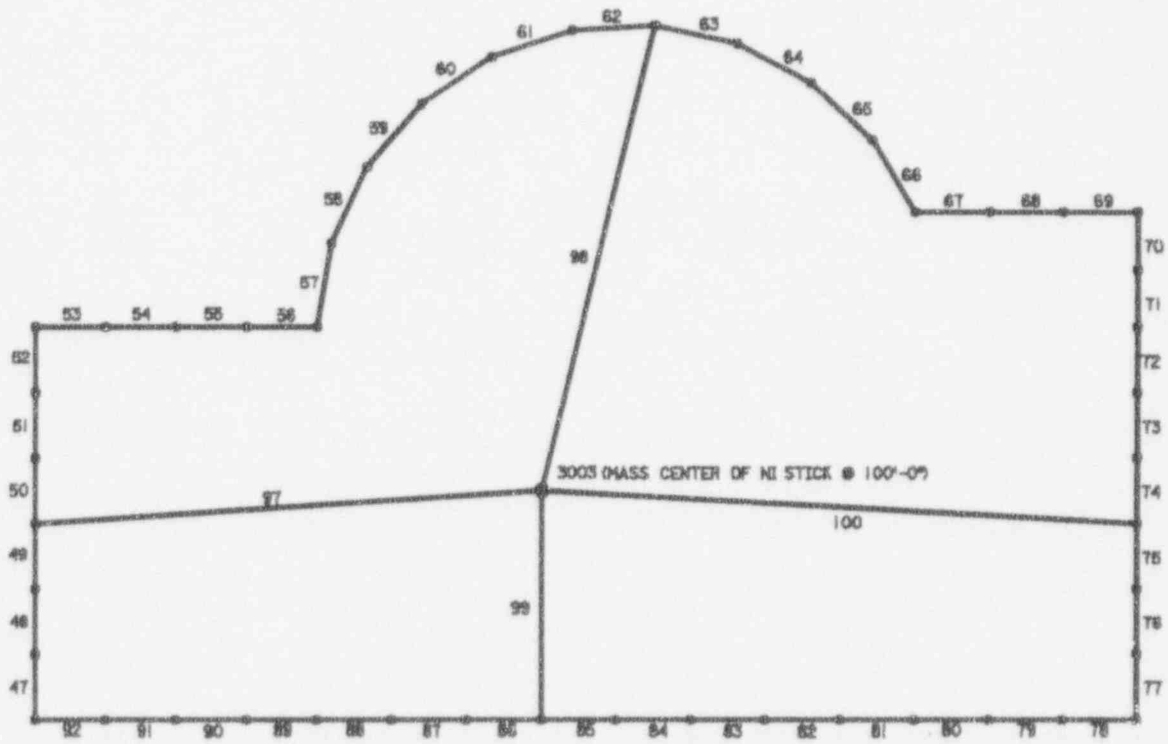
Figure 230.41-1
Foundation of the Seismic Analysis Model
for The Nuclear Island, Elevation 82'-6"



LOCATION OF RIGID BEAM ELEMENTS @ EL. 82'-6"

NOTE:
NUMBERED LINES INDICATE RIGID BEAM ELEMENTS

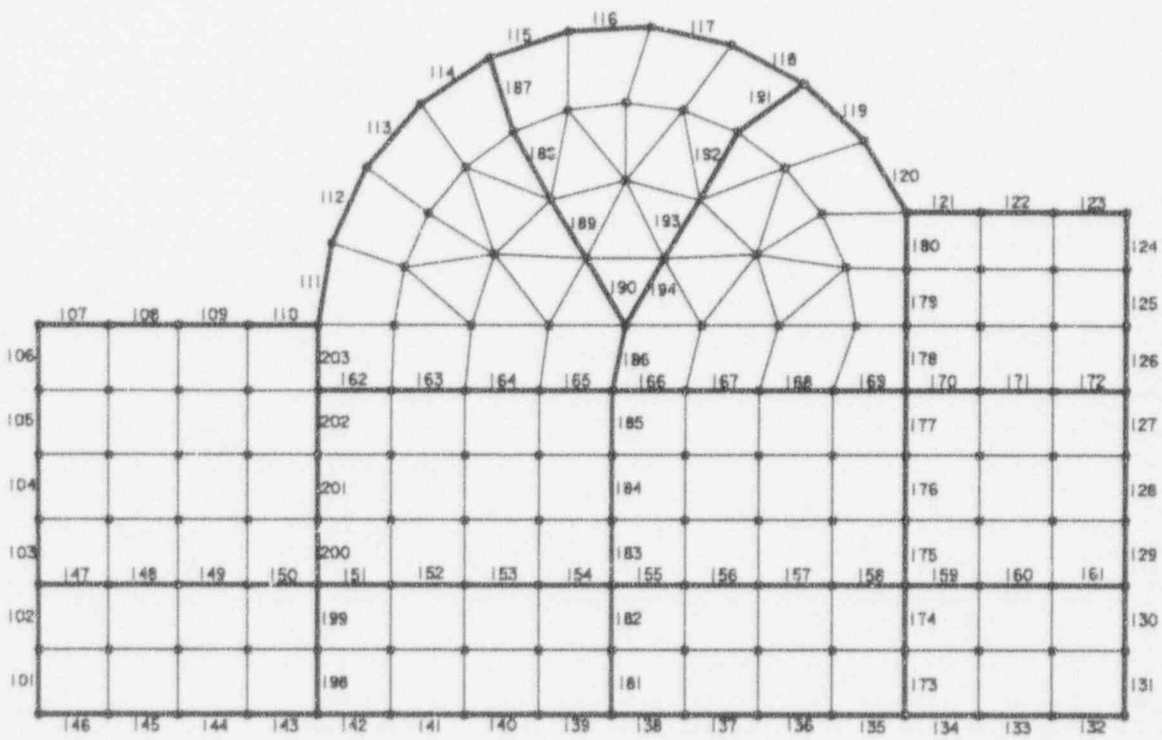
Figure 230.41-2
Foundation of the Seismic Analysis Model
for The Nuclear Island, Elevation 100'-0"



LOCATION OF RIGID BEAM ELEMENTS @ EL. 100'-0"

NOTE:
NUMBERED LINES INDICATE RIGID BEAM ELEMENTS

Figure 230.41-3
Foundation of the Seismic Analysis Model
for The Nuclear Island, Top of Basemat



LOCATION OF RIGID BEAM ELEMENTS @ EL. 66'-6"
(TOP OF BASEMAT)

NOTE:
NUMBERED LINES INDICATE RIGID BEAM ELEMENTS

Question 230.42

The following request for additional information pertains to Section 3.7.2.4 of the SSAR:

a. The first paragraph of Section 3.7.2.4 states that the nuclear island SSI responses generated for the analysis and design of seismic subsystems include nodal displacements, nodal accelerations and floor response spectra (FRS). Explain how the structural member forces (forces and moments) used for the structural design were generated for a soil site condition.

b. The last paragraph of Section 3.7.2.4 (Page 3.7-7) states that the selected soil conditions envelop the potential variation of soil properties and, therefore, the guidelines of SRP Section 3.7.2 for the variation of soil properties were not considered. Justify this statement, especially, when structures are founded on soft soil site for which the variation (uncertainty) in soil properties should be carefully considered.

c. Explain the differences between the two phrases "the time-history SSI analysis using the program SASSI" and "the complex frequency response analysis using the program SASSI."

d. When the computer code SHAKE was applied, which soil degradation curve was used?

Response: (Revision 1)

a. The structural member forces used for the structural design for the soil site condition are generated as described in the last paragraph of subsection 3.7.2.1.1.

b. The soft soil profiles considered in the SSI analysis (see Figure 2A-7 of the SSAR, Appendix 2A) has a linear shear wave velocity profile varying from 1000 ft/sec at the ground surface to 1200 ft/sec at 240 ft depth. This profile is considered the minimum "best estimate" velocity profile among the soil cases. In order to study the effect of lower bound shear wave velocities associated with the variation in the "best estimate" profile, a reduction of 50 percent in the low-strain shear modulus was analyzed. The resultant shear wave velocity for the lower bound case has a low-strain velocity of 707 ft/sec at the ground surface increasing to 850 ft/sec at 240 ft depth. This profile was analyzed using SHAKE and the Idriss 1990 soil curves (see response to RAI 230.79). The resultant strain-compatible soil properties were used in the 2D SSI analysis in the NS direction considering the depth to base rock of 120 ft and the water table at grade level. The SSI results of this case (marked as the lower bound soft soil case) are compared with the SSI results of the minimum "best estimate" profile (marked as the soft soil case) and with those of other design profiles (hard rock, soft rock, soft-to-medium soil) in Figures 230.42-1 through 230.42-11. As shown in these figures, the results of the lower bound soft soil case are enveloped by the results of other SSI cases except for small exceedences at very low frequencies (less than 1 Hertz) with no significant effect on the design responses. Based on these results, the soft soil profile (1000 ft/sec at the surface to 1200 ft/sec at 240 depth) is considered to be the minimum "best estimate" profile in the site interface conditions.

~~The SRP requirement for variation of soil properties is not considered because of the following:~~

- ~~Sensitivity studies have been performed for a broad range of soil and rock site conditions (see Appendix 2A of the SSAR) to assess the impact of site parametric uncertainties because of the seismic design basis is to provide design coverage for as many plant sites as practical.~~

- ~~A suitable set of design soil profiles covering sites with shear velocities from 1000 fps to 8000 fps, have been established for the plant seismic design based on the above evaluation of the generic sites.~~

- ~~The site interface requirements established in Section 2.5 require the proposed sites to be within the generic site sensitivity analyses, such as :~~

  - ~~The shear wave velocity (based on low strain best estimate soil properties) is greater than or equal to 1000 fps.~~
  - ~~There is no potential for fault displacement at plant site.~~

c. As discussed in responses to RAI 230.34, Section 3.7.2.4 will be revised and the statements are clarified to read as follows:

- The soil-structure interaction (SSI) analyses of the nuclear island are performed using the program SASSI, and

- SSI analyses are performed using the complex frequency response method with computer program SASSI.

d. As discussed in Section 2A.4, the strain-dependent shear modulus and damping curves used in the free-field SHAKE analysis are presented in Figures 2A-8 for soil materials and in Figure 2A-9 for rock materials. These curves are obtained from references 6, 7 and 8 shown in Subsection 2A.7.

SSAR Revision:

See SSAR revision identified in response to RAI 230.34.

Figure 230.42-1
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

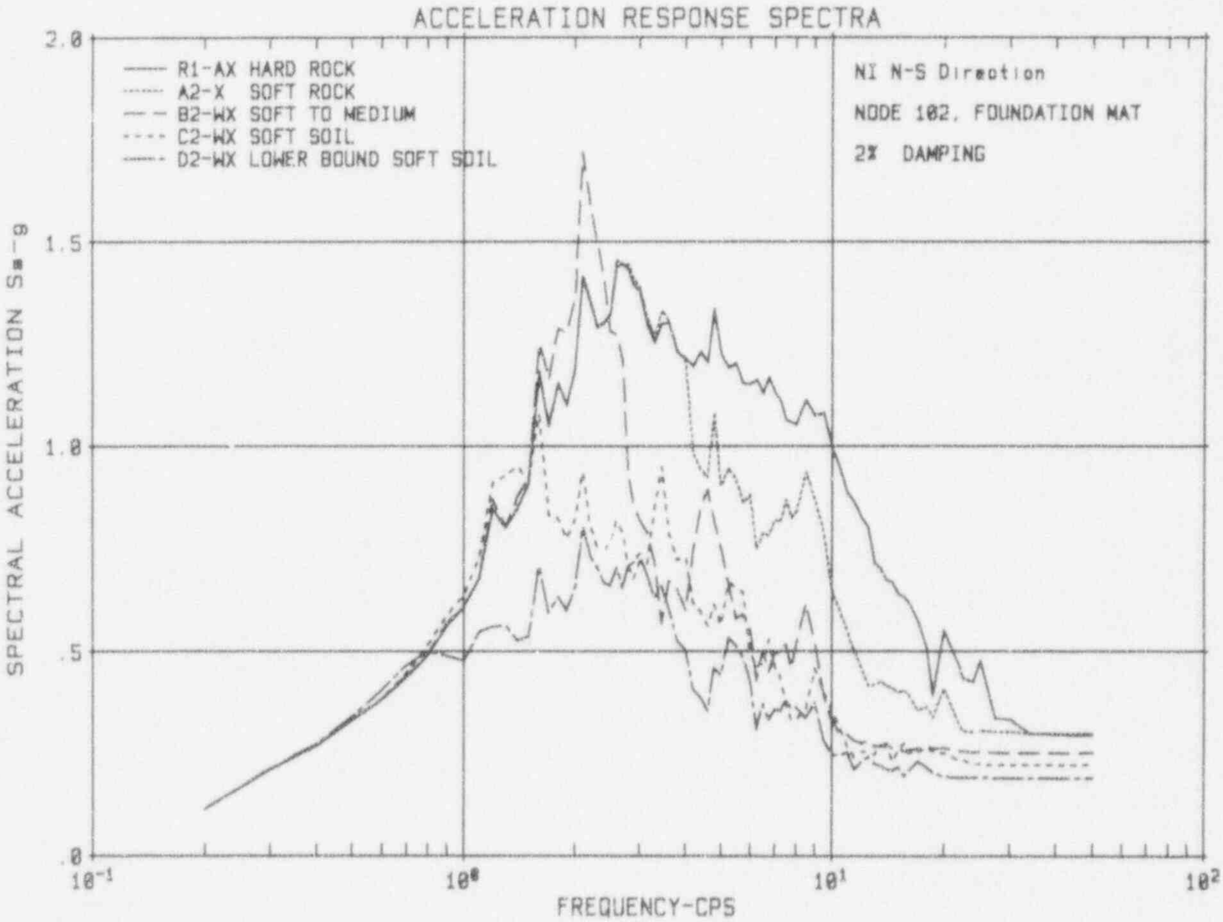


Figure 230.42-2
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

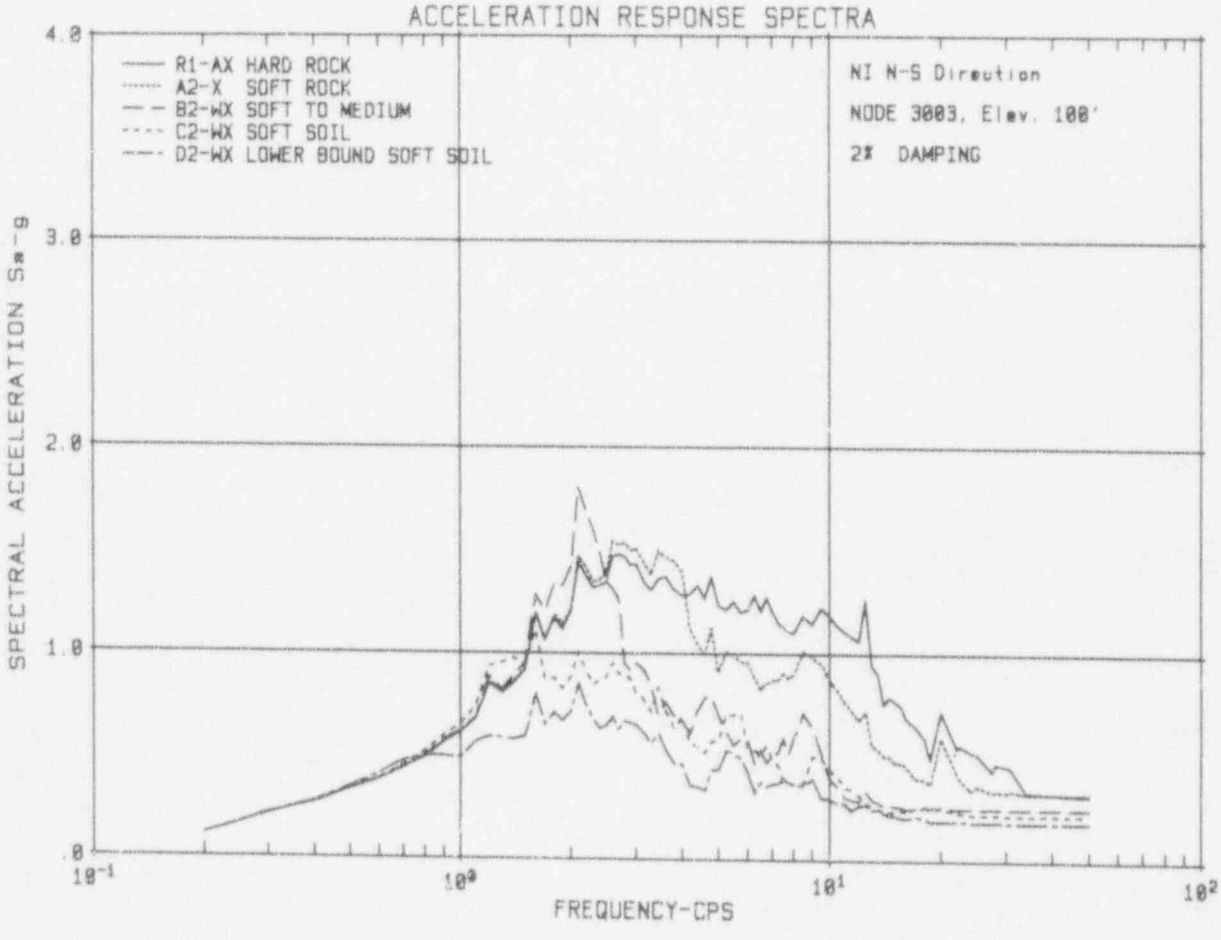


Figure 230.42-3
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

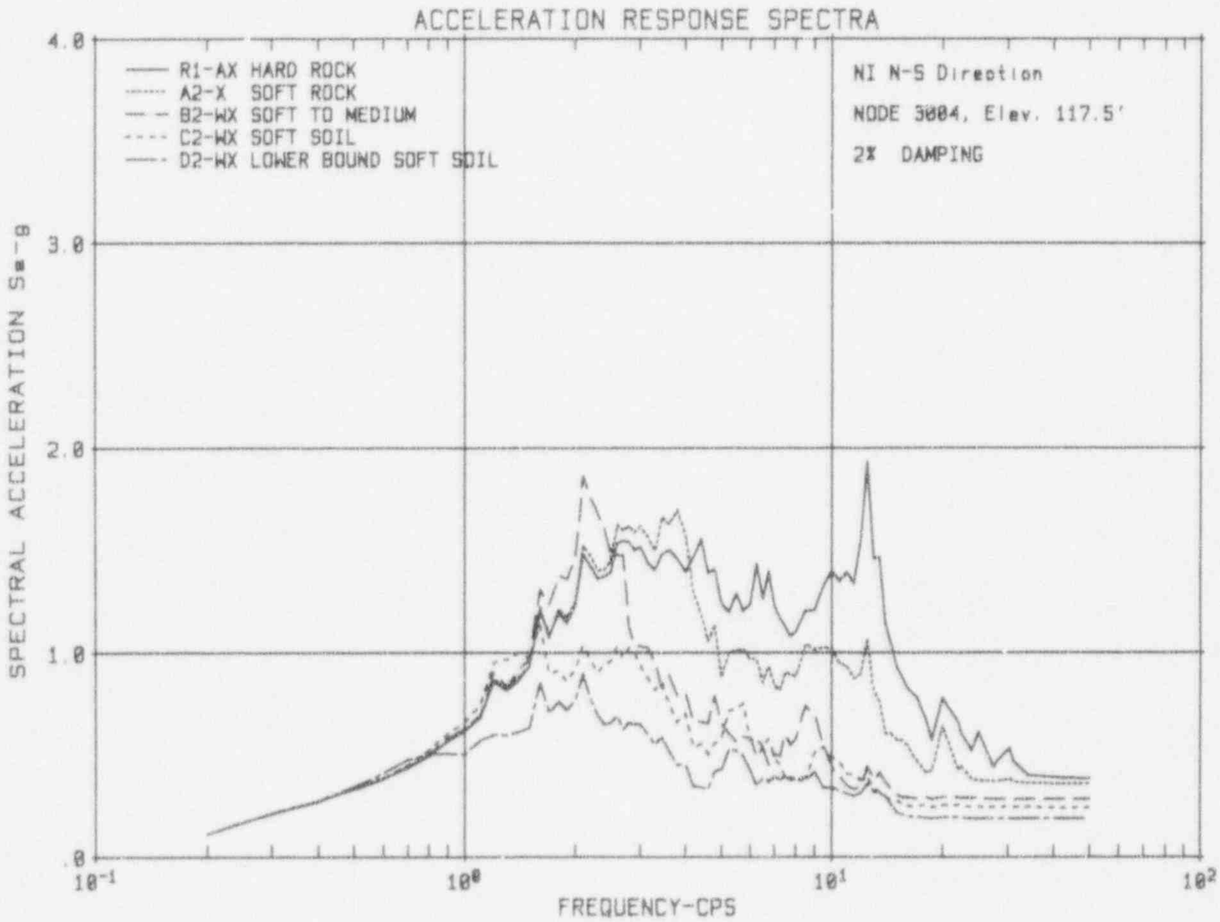


Figure 230.42-4
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

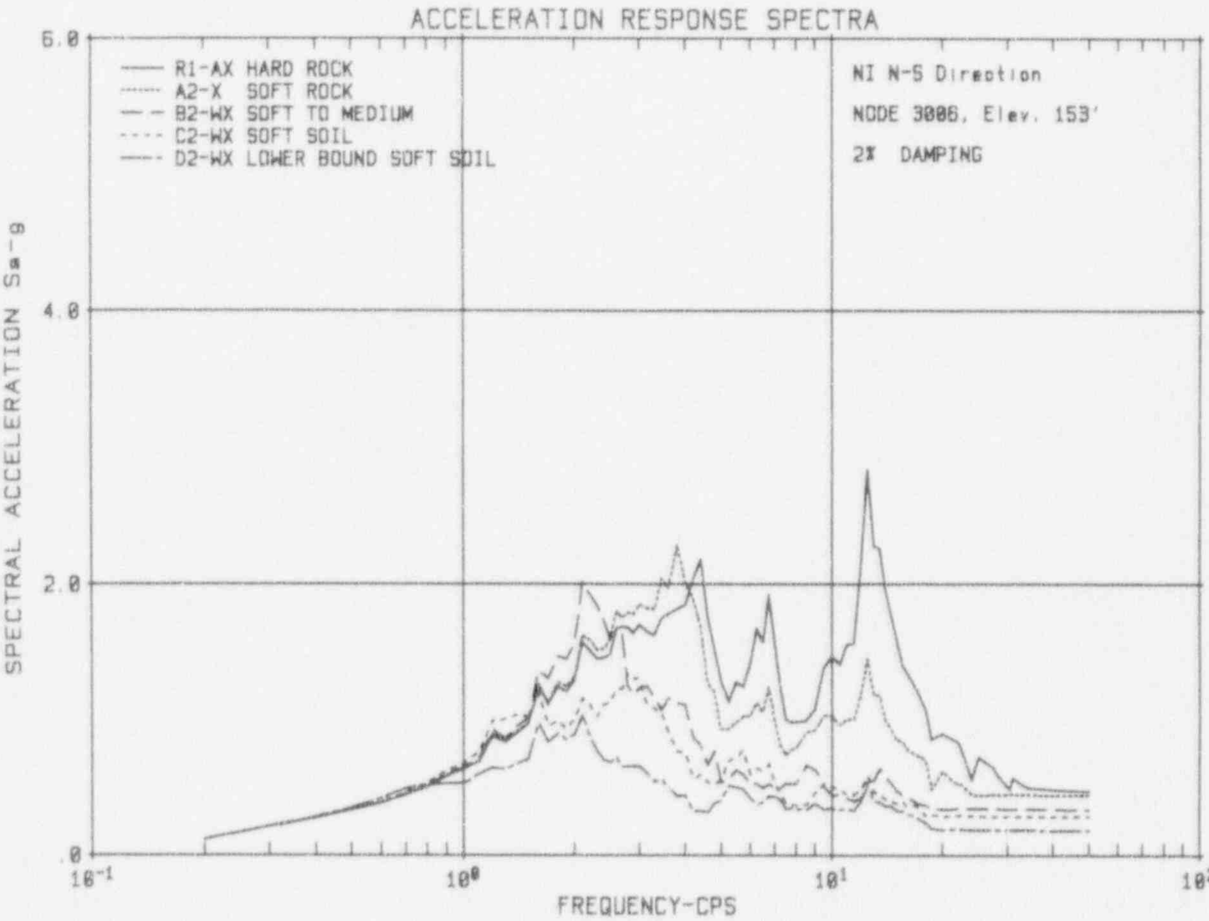


Figure 230.42-5
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

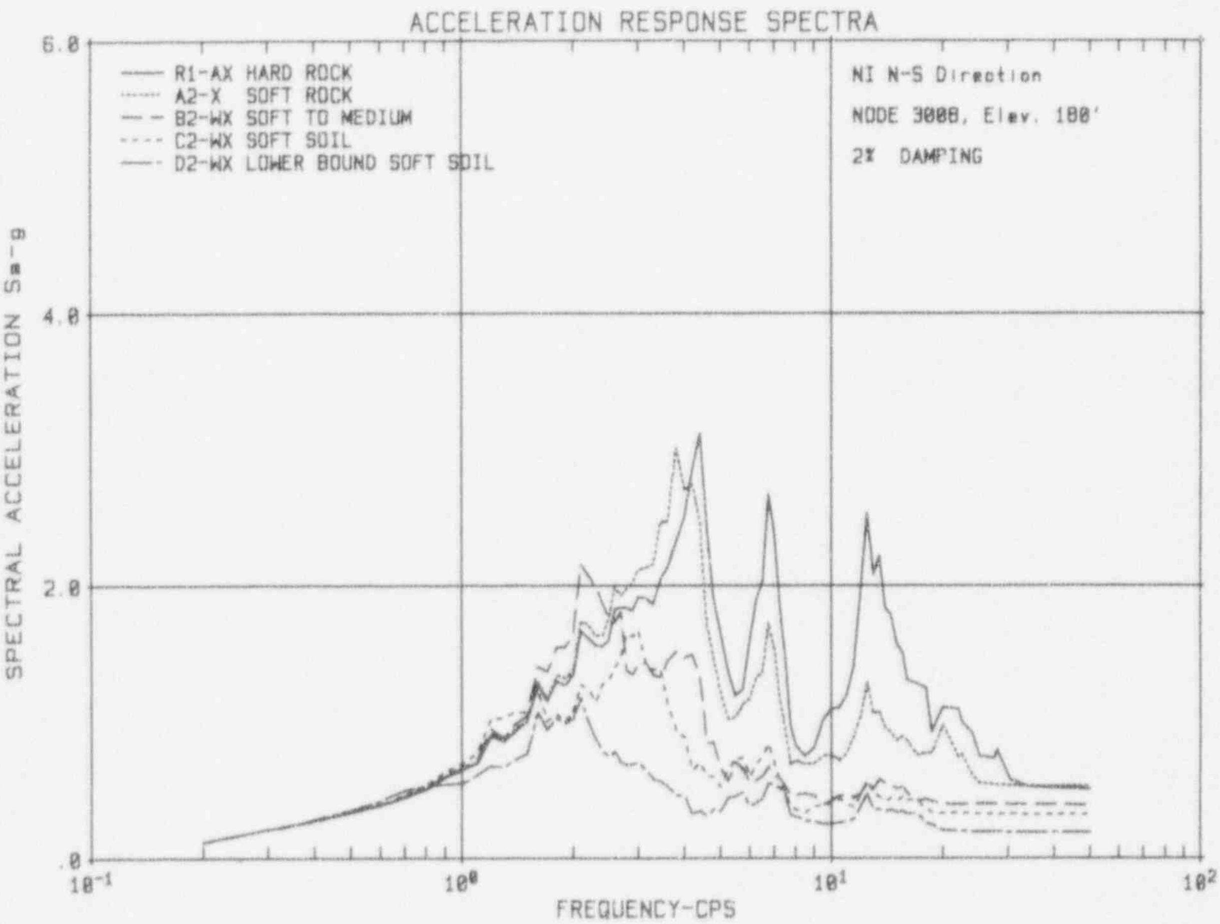


Figure 230.42-6
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

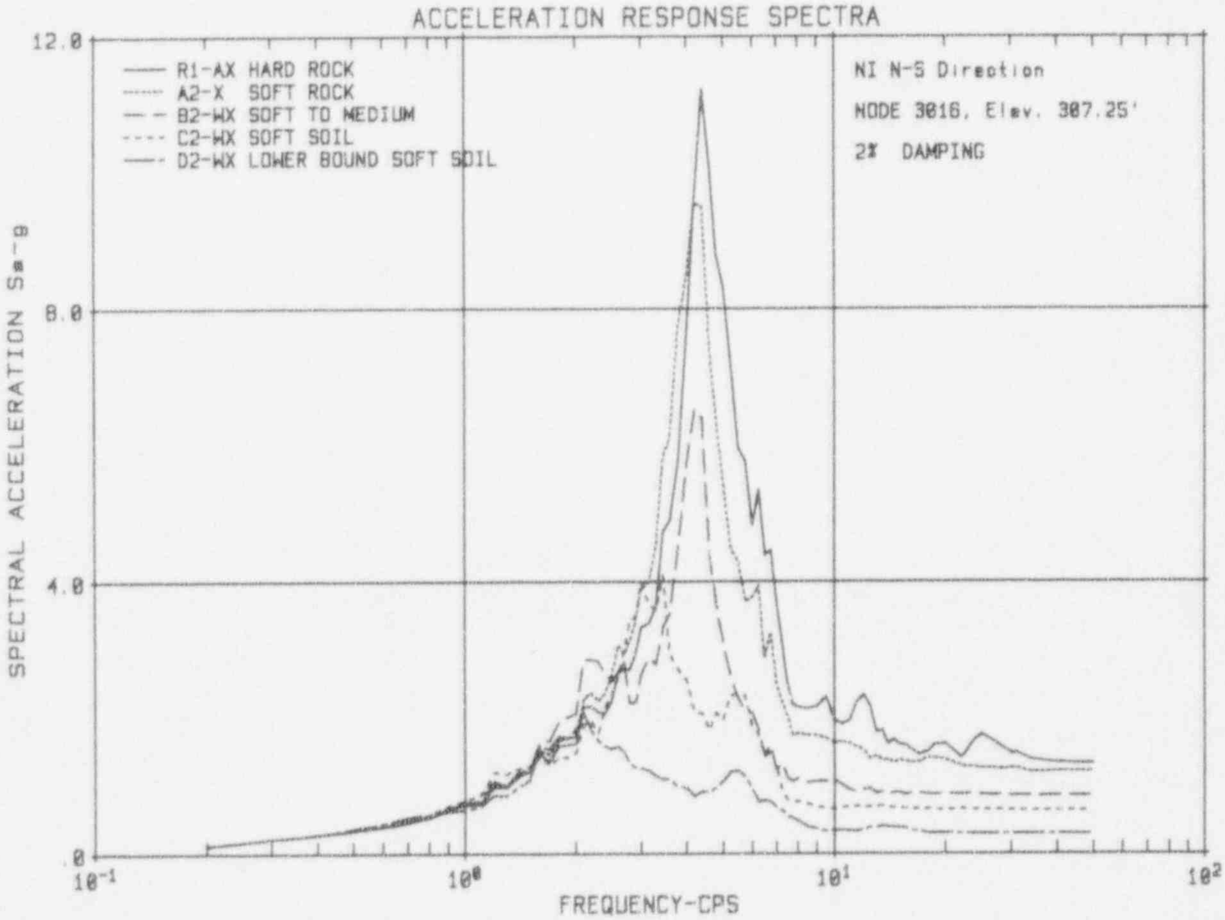


Figure 230.42-7
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

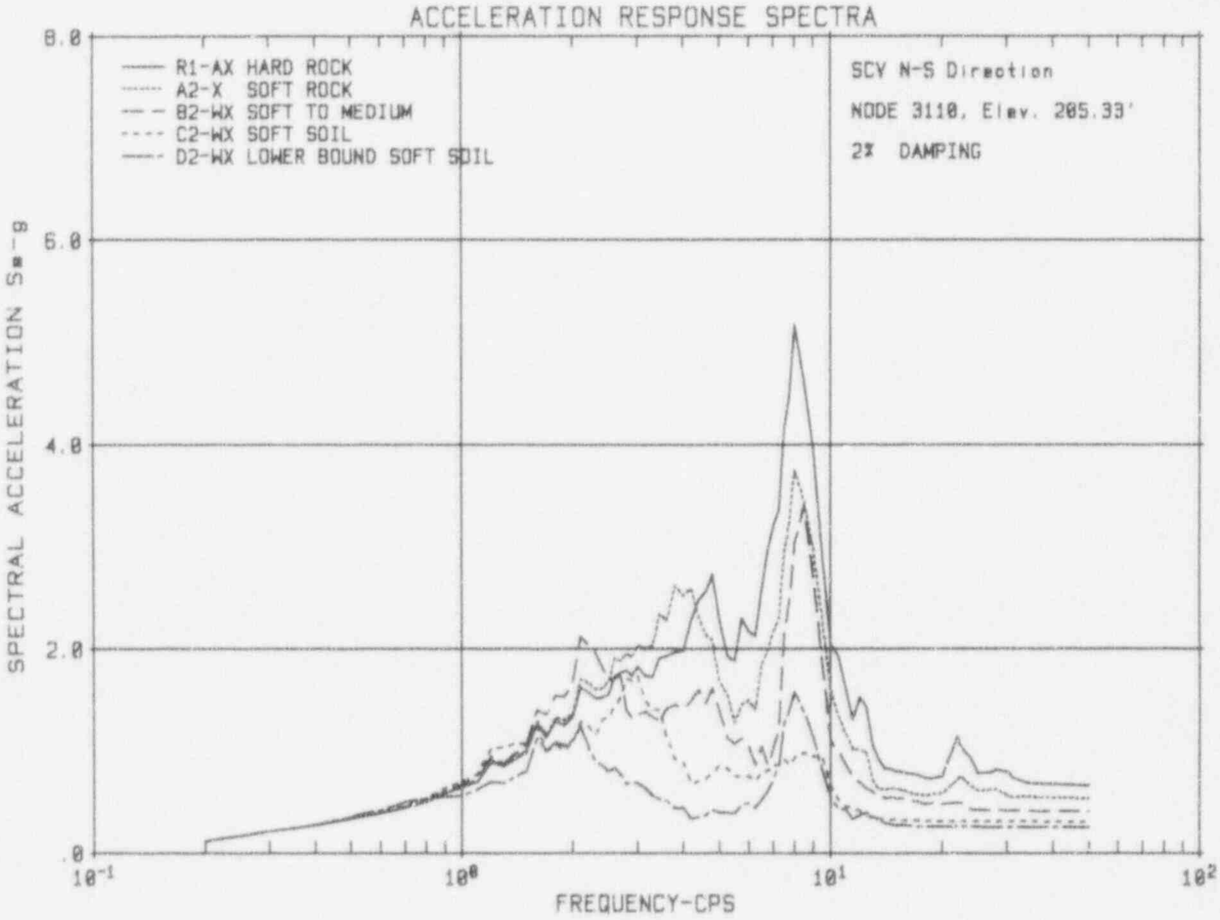


Figure 230.42-8
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

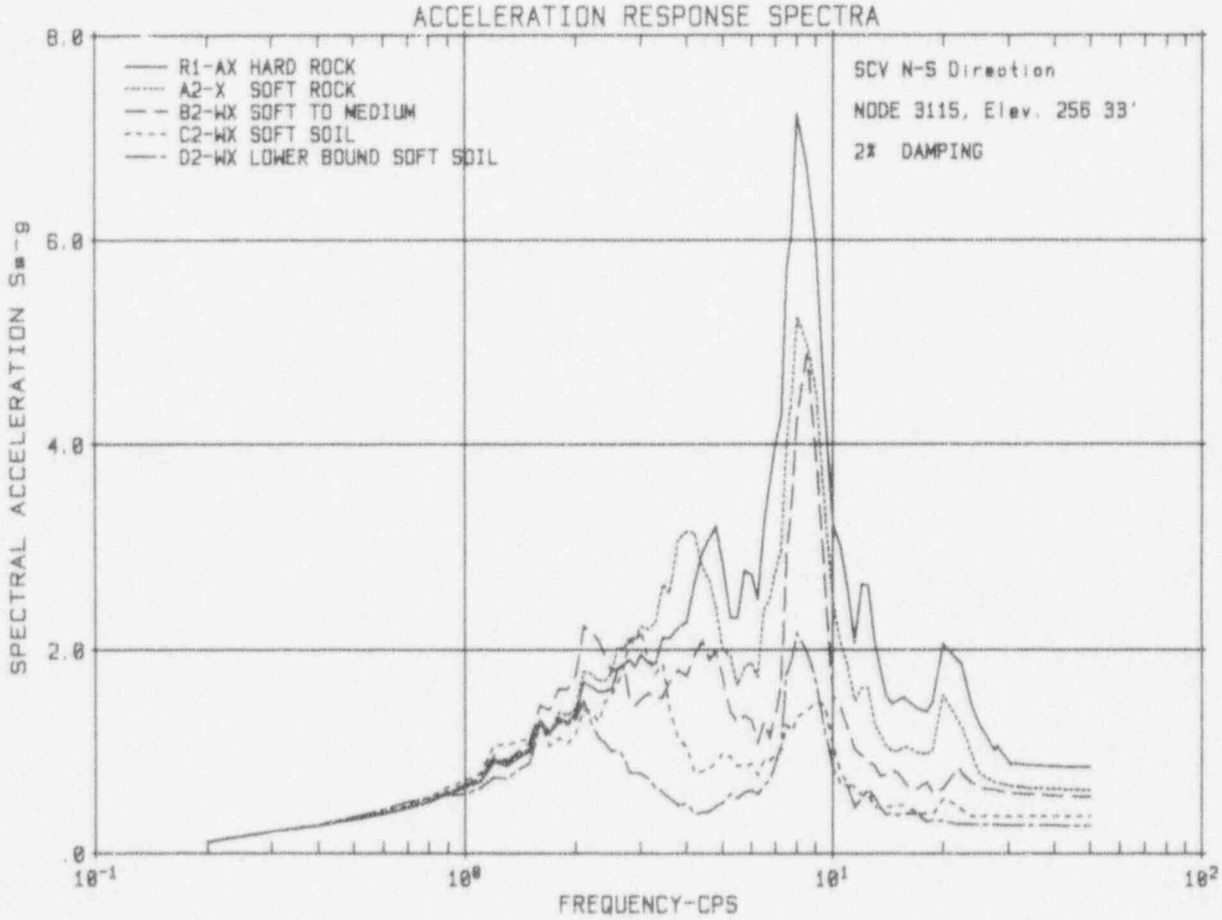


Figure 230.42-9
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

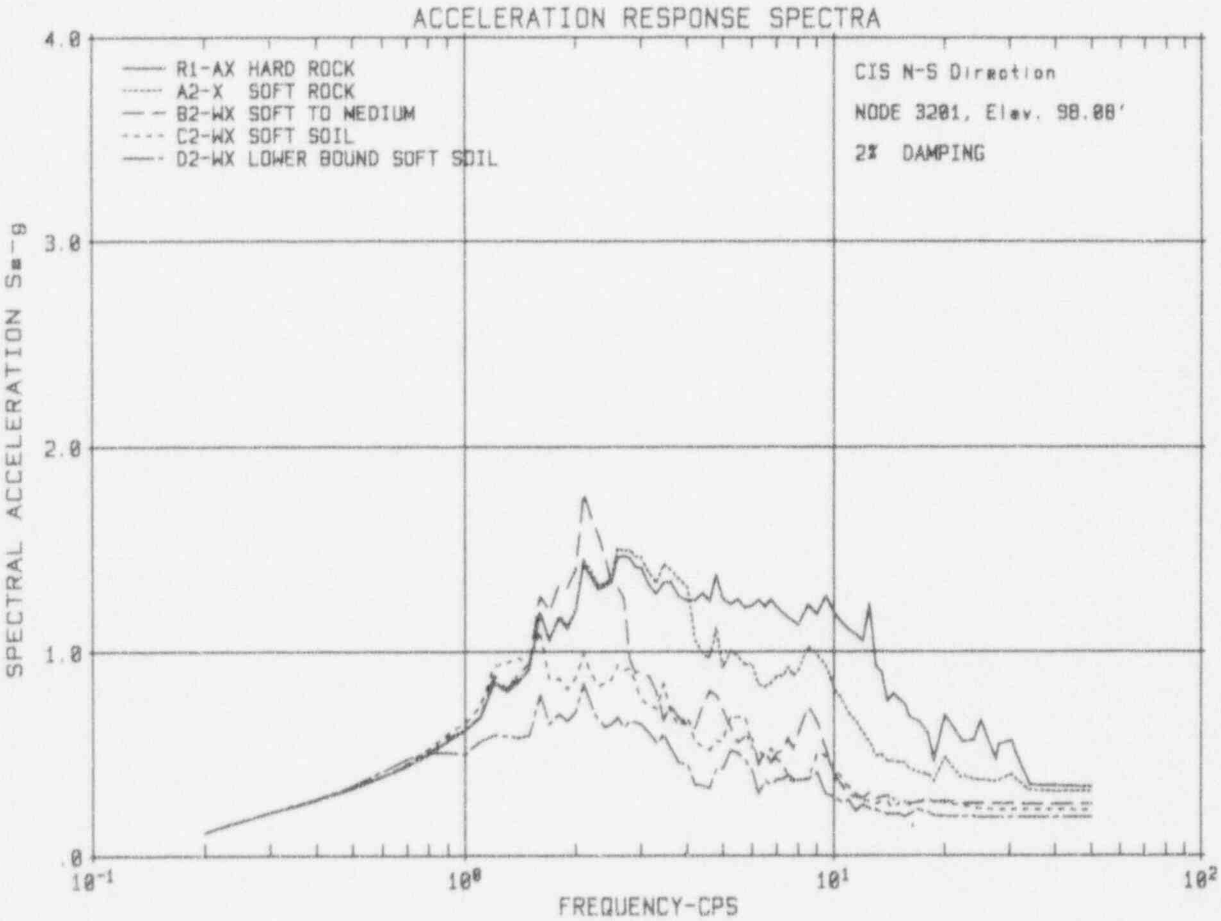


Figure 230.42-10
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile

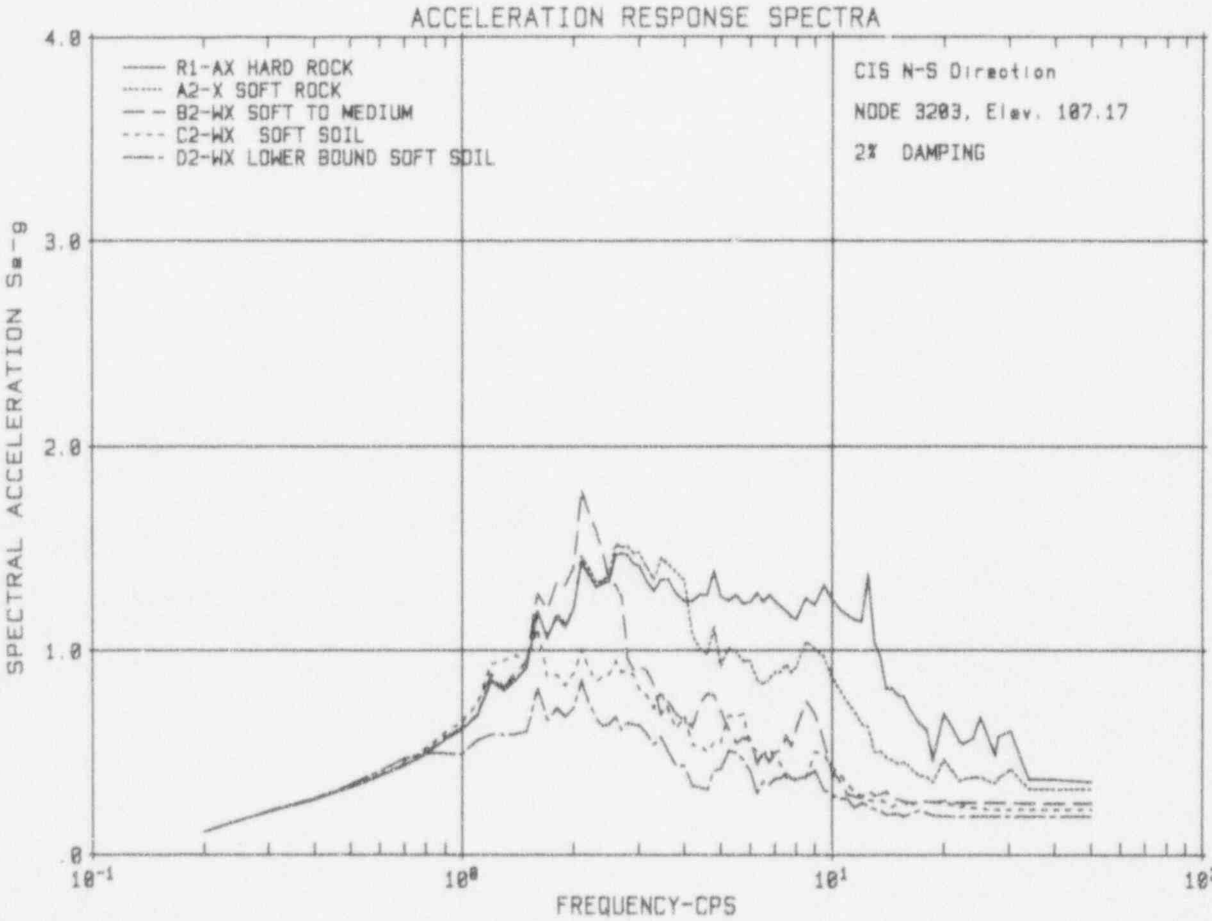
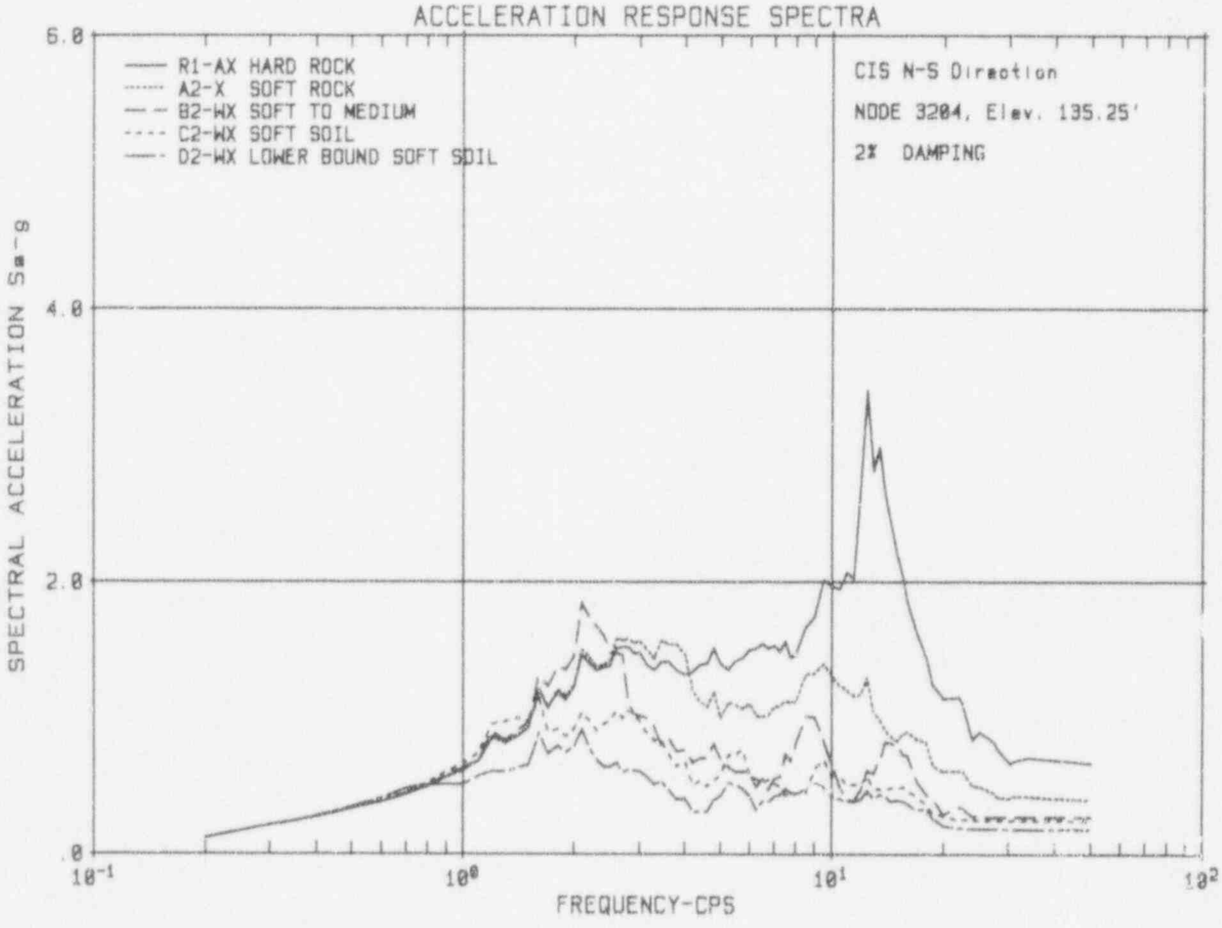


Figure 230.42-11
2D SASSI Analysis, N-S Direction
Lower Bound Soft Soil Profile



Question 230.79

From the review of Figures 3.7.1-14 and 3.7.1-15 of the SSAR, it appears that the soil shear degradation curves for the typical soil used in the analysis and design are based on the soil shear degradation model recommended by H.B. Seed and I.M. Idriss in 1970. A comparison of the shear degradation curves presented in Figures 3.7.1- 14 and 3.7.1-15 in the SSAR with the current published industry results, such as the results published by I.M. Idriss and Geomatrix in 1990, shows that the Seed-Idriss 1970 curves overestimated the soil strain degradation. The staff anticipates that the use of the Seed-Idriss 1970 curves in the SSI analyses of the NI structures will underestimate the seismic structural responses. Provide the basis for using the Seed-Idriss 1970 curves in the SSI analyses.

Response: (Revision 1)

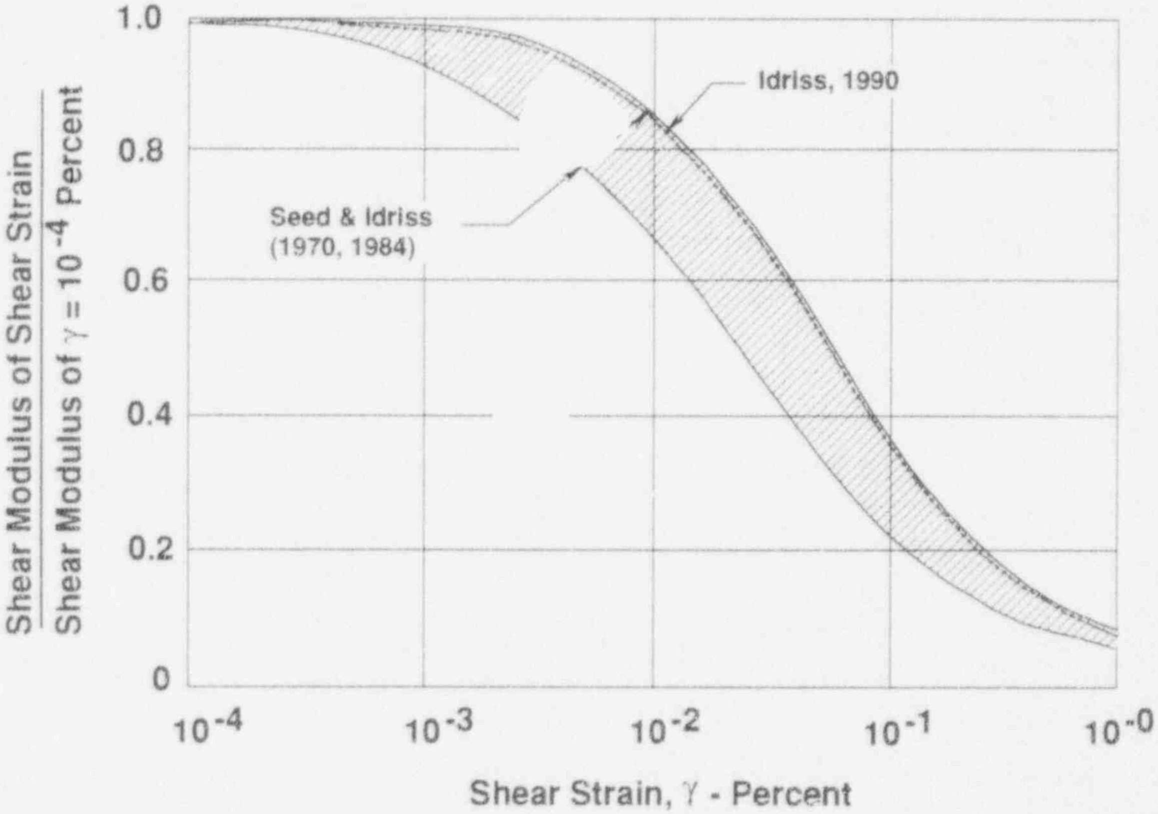
~~The subject of this RAI was discussed during a meeting among NRC staff and consultants and Westinghouse and Bechtel on seismic analyses on April 14, 1994 and will be discussed further during a meeting scheduled at the end of May. A written response to this RAI will be prepared following the May meeting.~~

The range of soil degradation curves proposed by Seed & Idriss in 1970 and confirmed in the 1984 study are shown in Figures 230.79-1 and 230.79-2. The average curves corresponding to this range with limiting soil material damping value of 15 percent were used in the seismic SSI analysis of the NI with the foundation soil sites. The soil degradation curves reported by Idriss (1990) are also shown in these figures. In order to assess the impact of the new soil degradation curves on seismic SSI responses, the case of soft-to-medium soil case was re-analyzed in the EW direction using the Idriss 1990 soil degradation curves and the 2D SSI model of the nuclear island. The results of 2D analyses are compared with the 3D design responses in Figures 230.79-3 through 230.79-13. As shown in these figures, and depending on the nodal location, some small variations are observed with respect to the frequency and amplitude of the response. These differences are small and are generally covered by the 3D enveloping results. In light of the results of this parametric study and considering the fact that the design responses are the envelope of all soil and rock cases, re-analysis of SSI cases using Idriss 1990 curves are not considered warranted.

In relation to the above RAI, a concern was raised by the NRC team during the July 1994 audit regarding the adequacy of the range of soil degradation curves for sites with clay contents. In order to address this concern, the site-specific soil degradation curves for the Savannah River Site (marked as Geomatrix and Bechtel) and the Lotung site are compared with the range of Seed & Idriss curves in Figures 230.79-14 and 230.79-15. The soil materials at these two sites can be characterized as silty sand and clayey sand. As can be observed from these figures, the range in Seed & Idriss' study adequately covers the soil degradation curves of these sites. On the other hand, the strain-dependent properties of pure clay materials with high plasticity fall above the range in Seed & Idriss' study. However, sites with pure high plastic clay would not meet the AP600 interface requirements for minimum shear wave velocity characteristics and for bearing capacity.

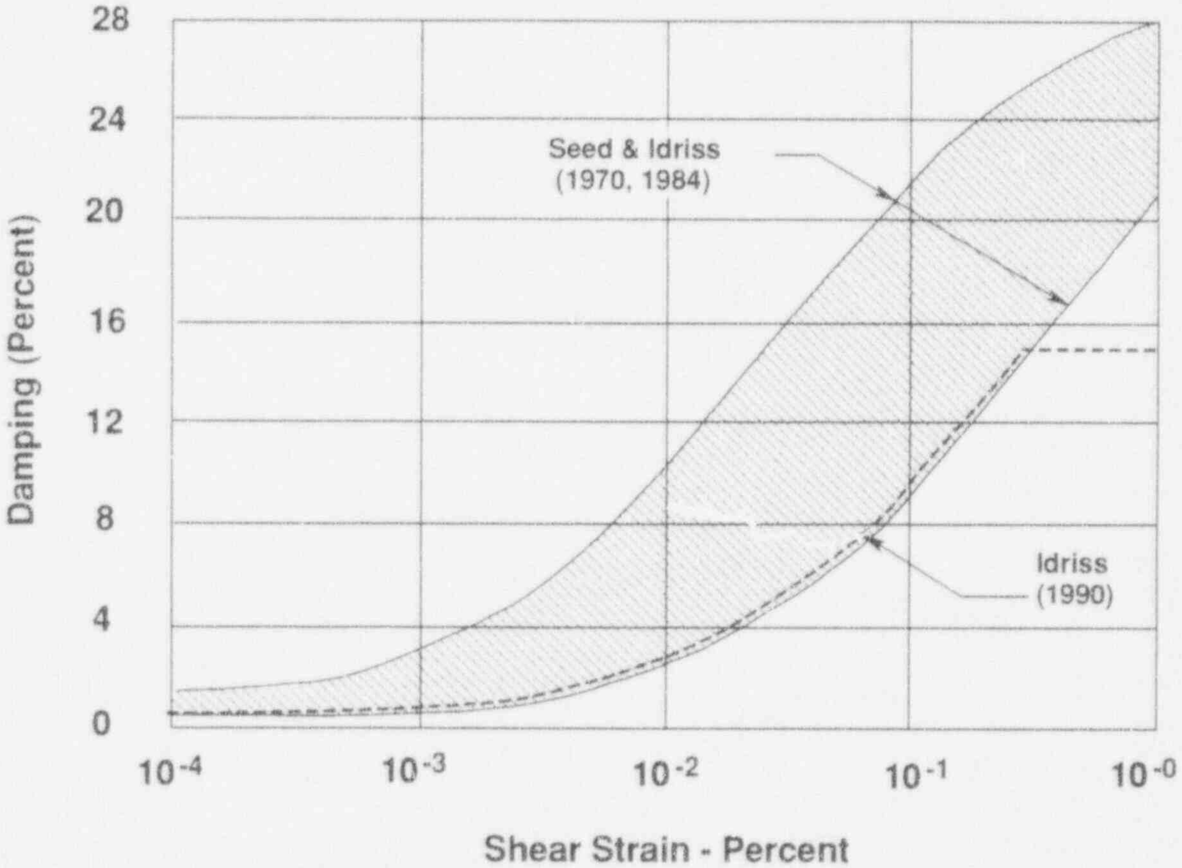
SSAR Revision: NONE

Figure 230.79-1
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation



Variation of Shear Modulus with Shear Strain for Sands

Figure 230.79-2
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation



Damping Ratios

Figure 230.79-3
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

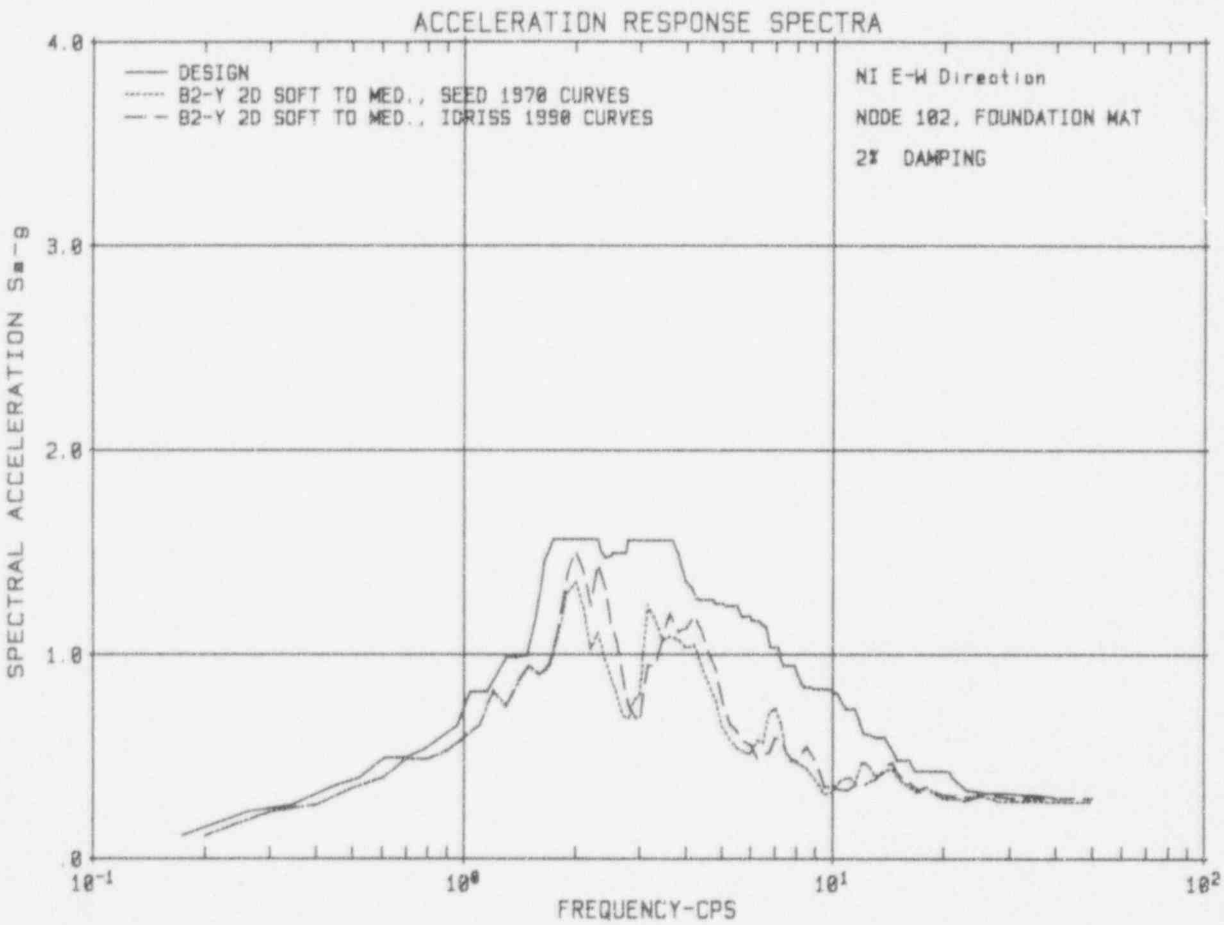


Figure 230.79-4
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

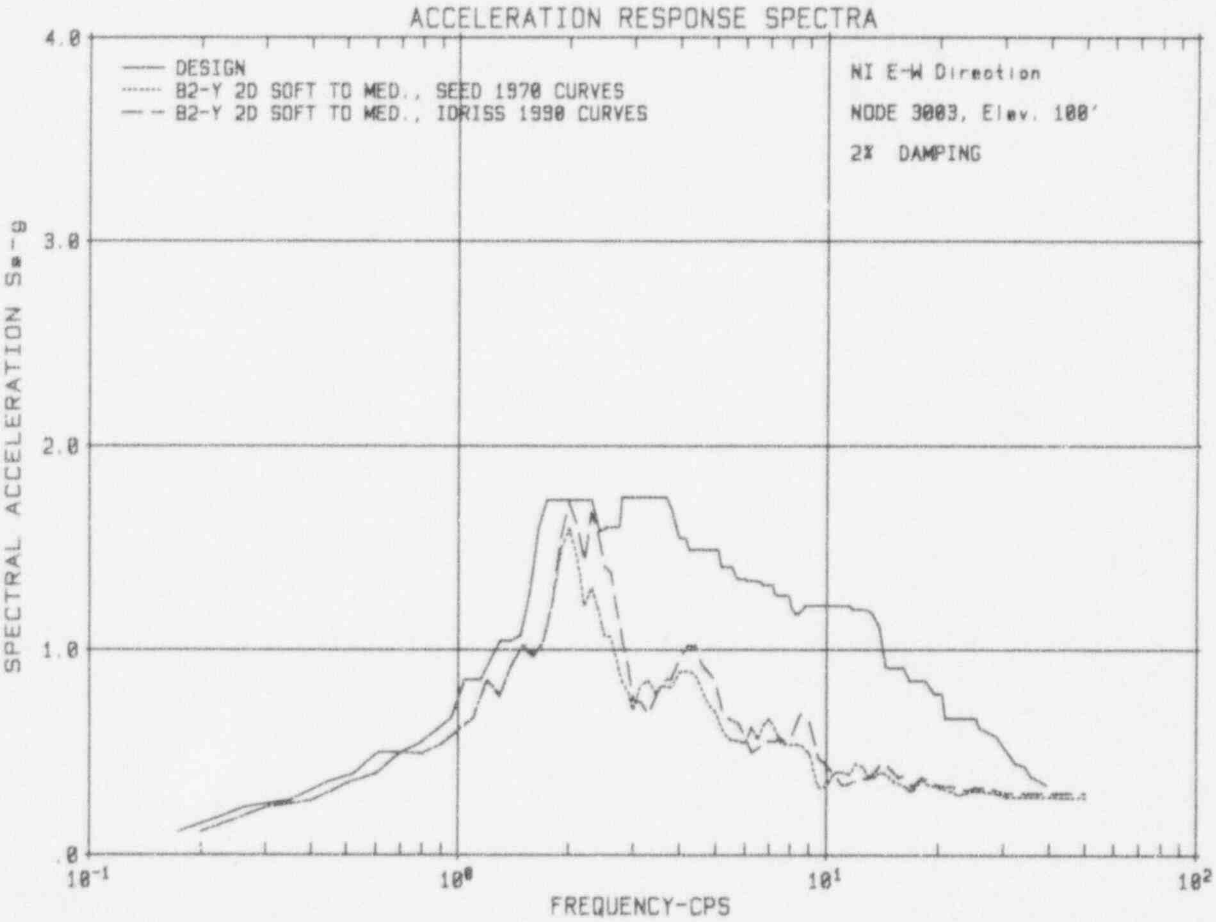


Figure 230.79-5
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

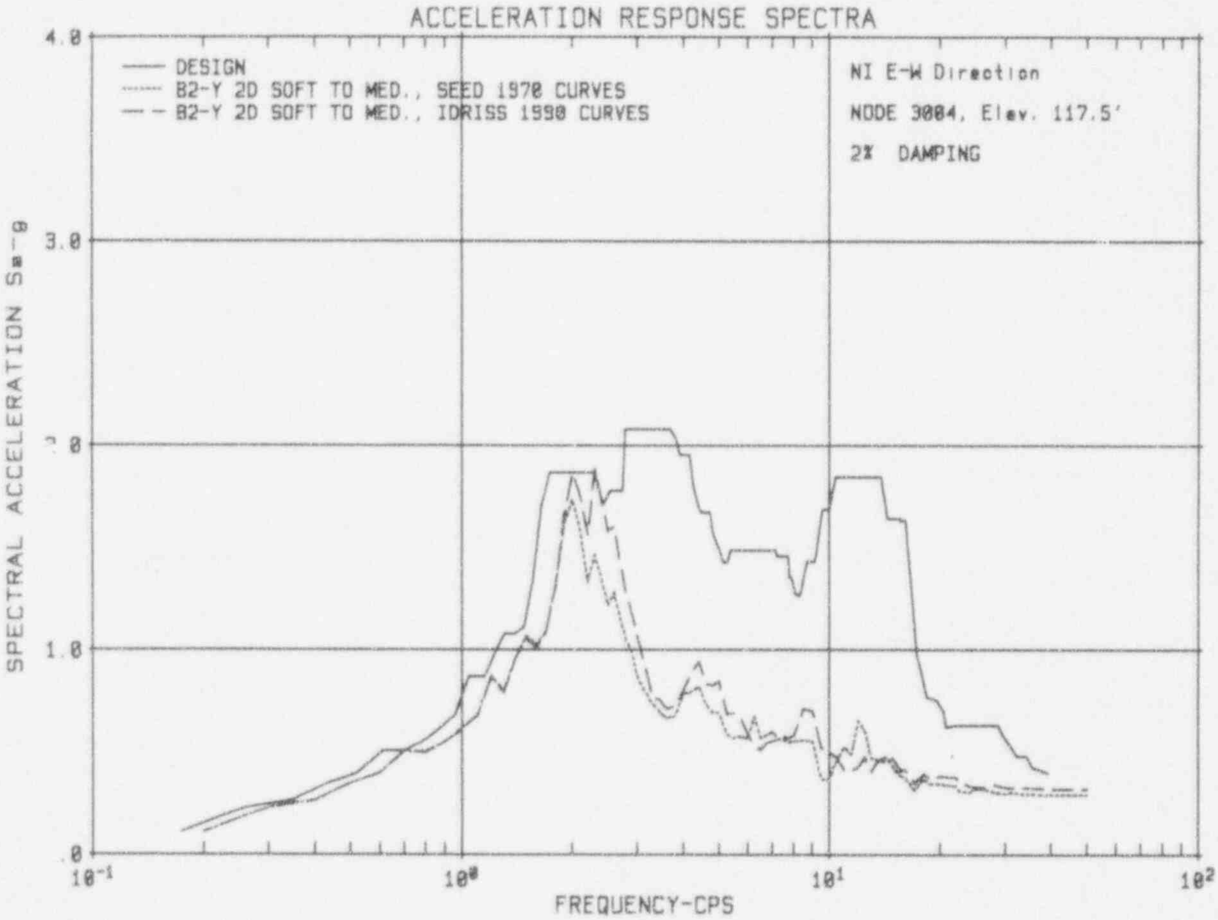


Figure 230.79-6
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

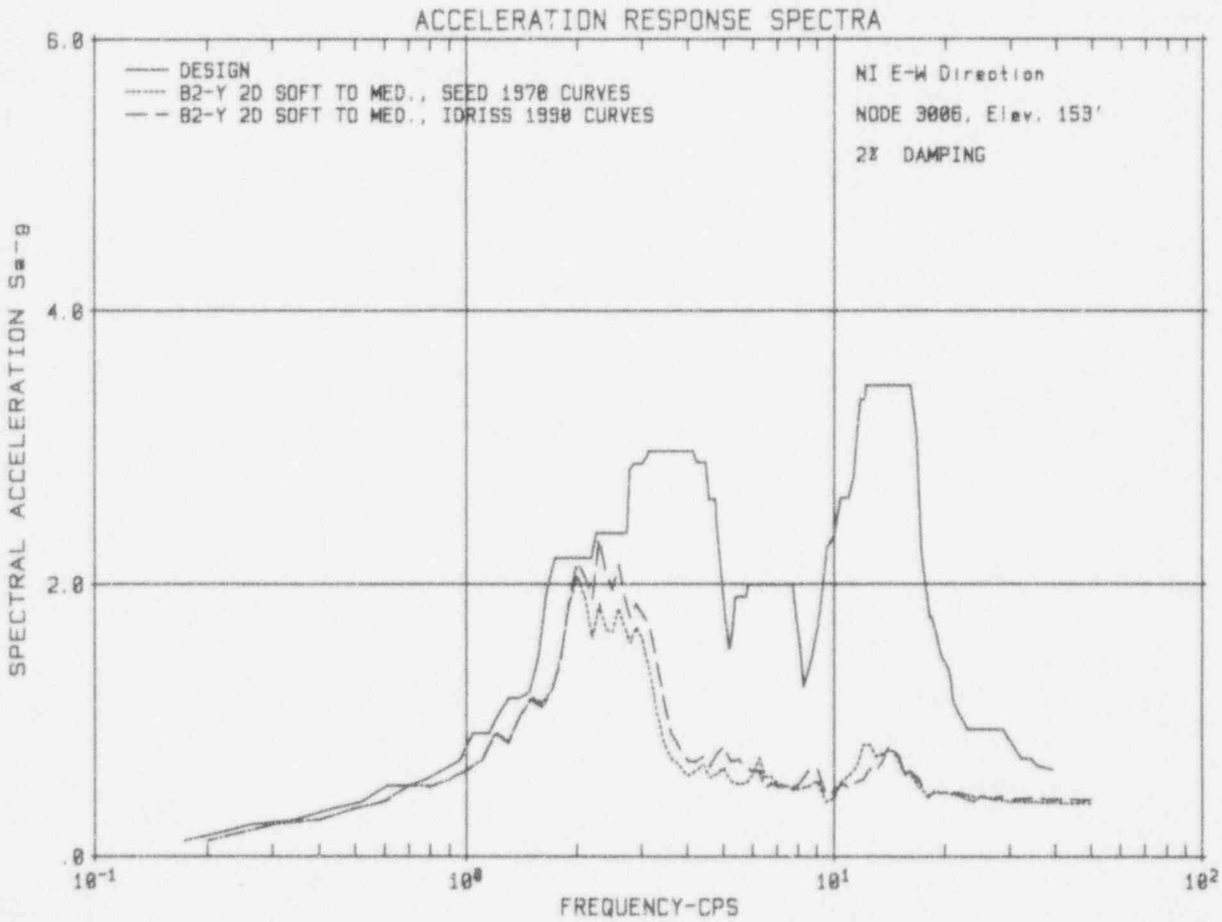


Figure 230.79-7
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

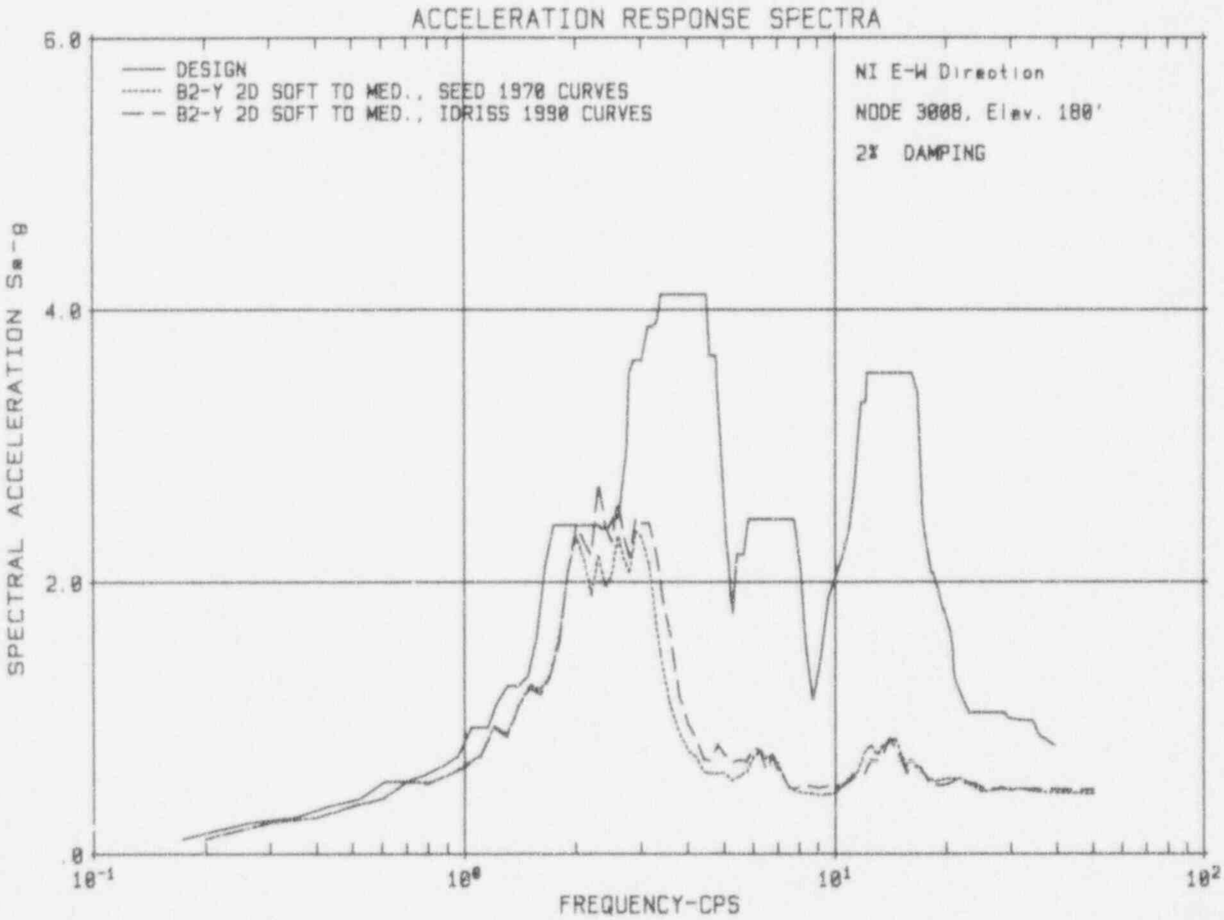


Figure 230.79-8
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

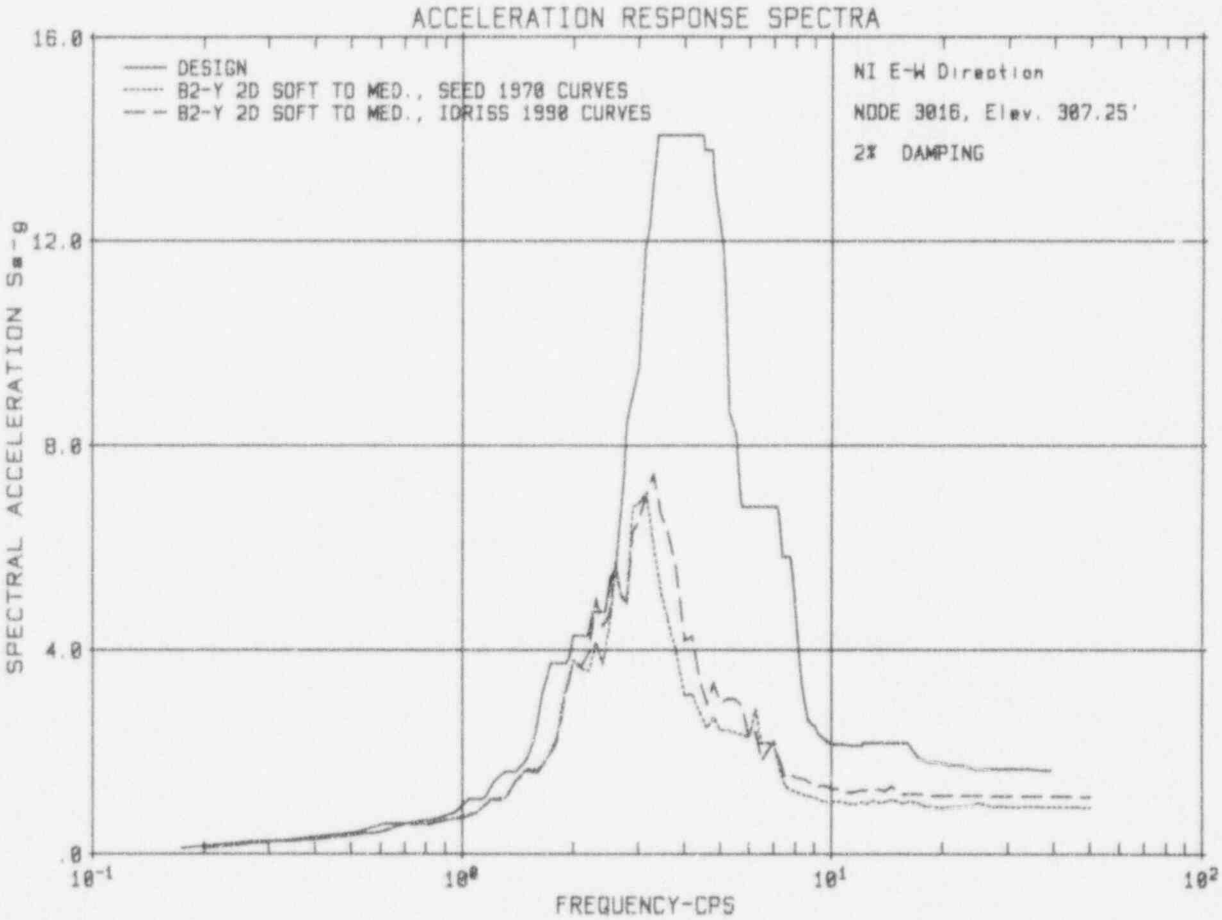


Figure 230.79-9
2D SASSI Analysis, E-W Directic..
Soil Degradation Curve Evaluation

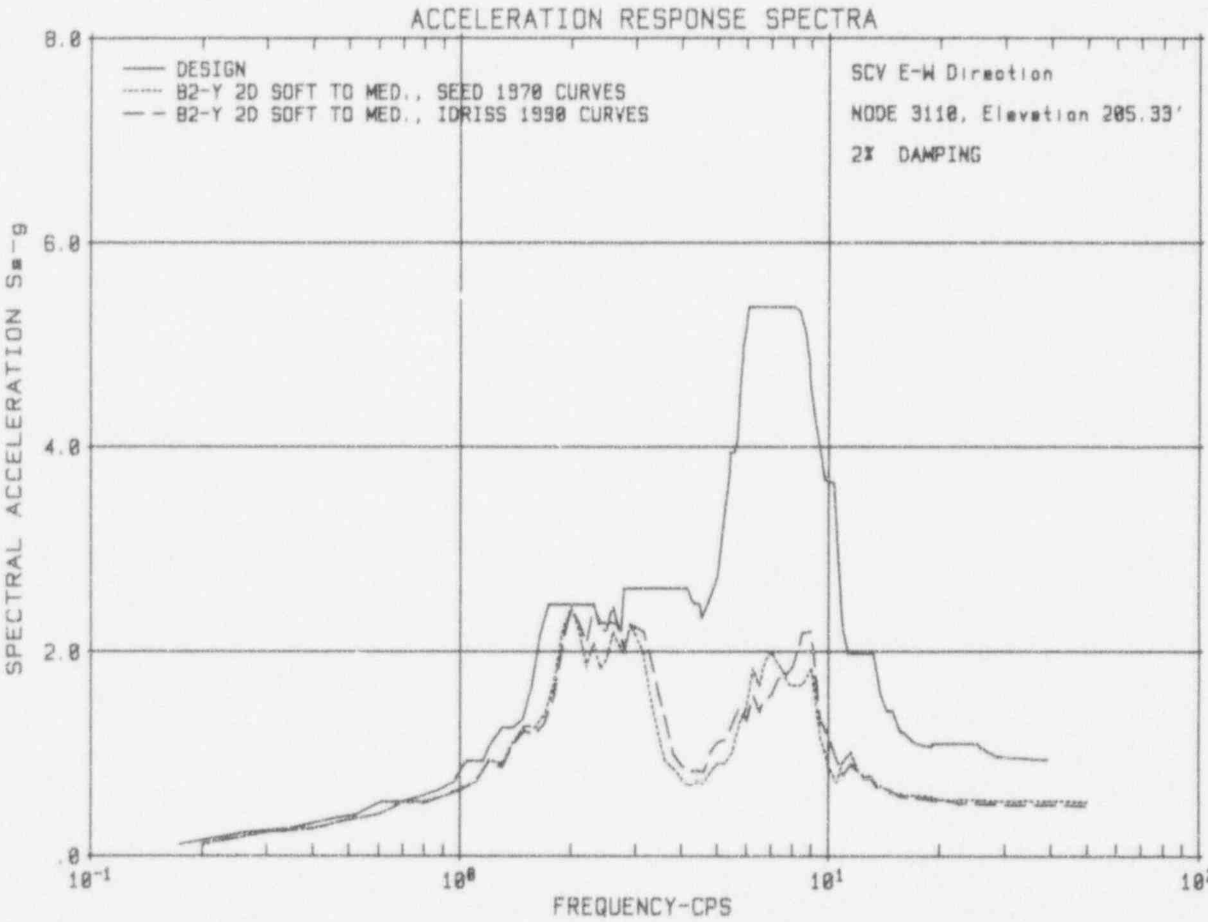


Figure 230.79-10
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

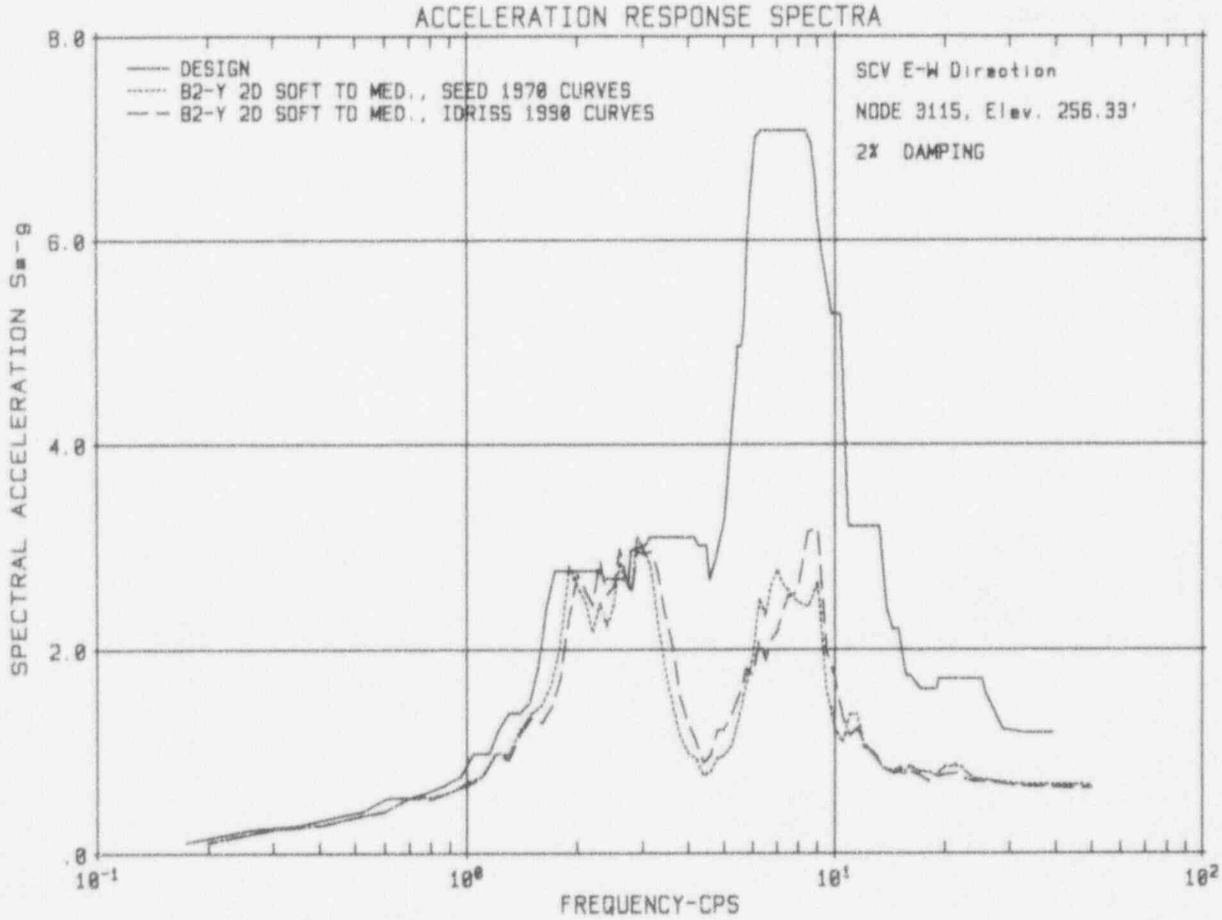


Figure 230.79-11
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

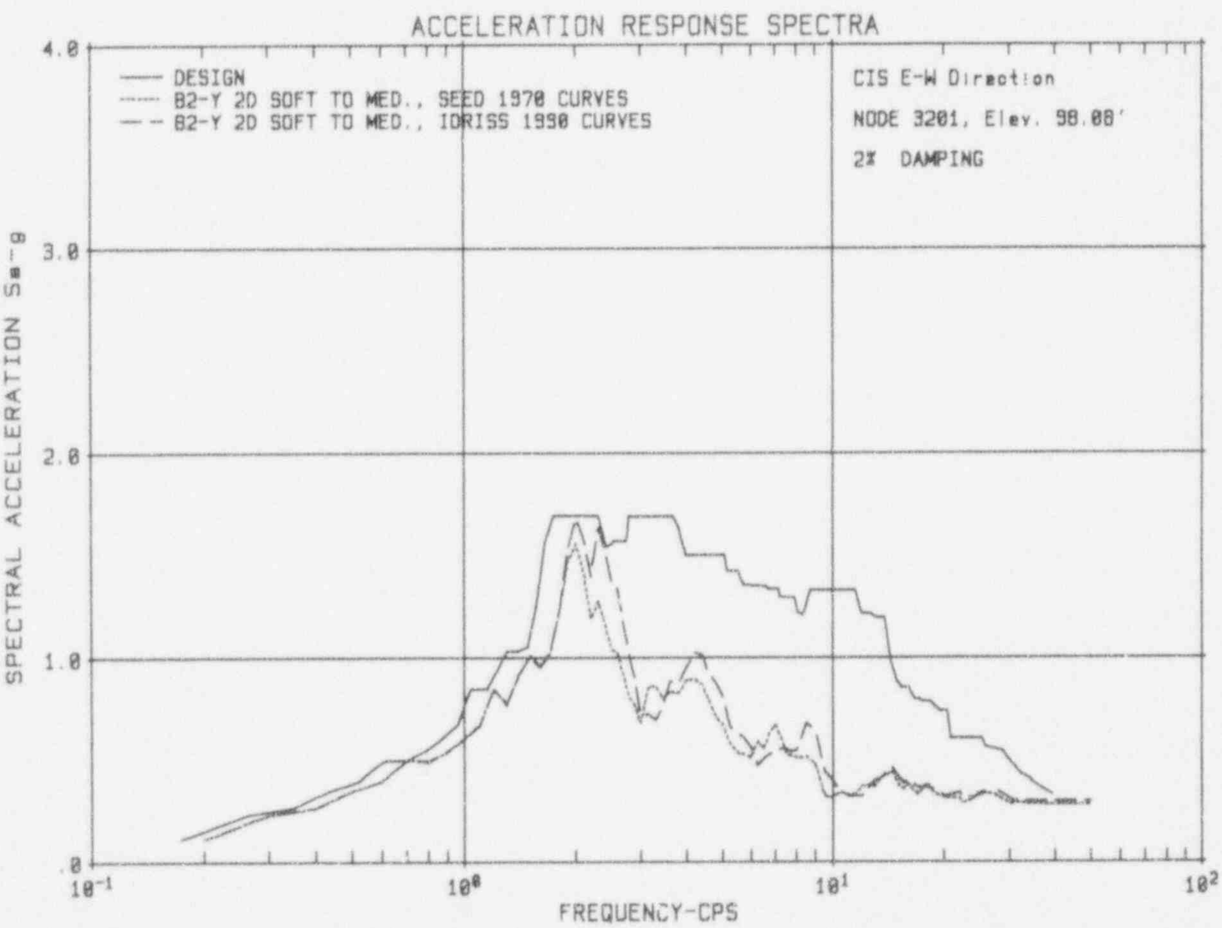


Figure 230.79-12
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

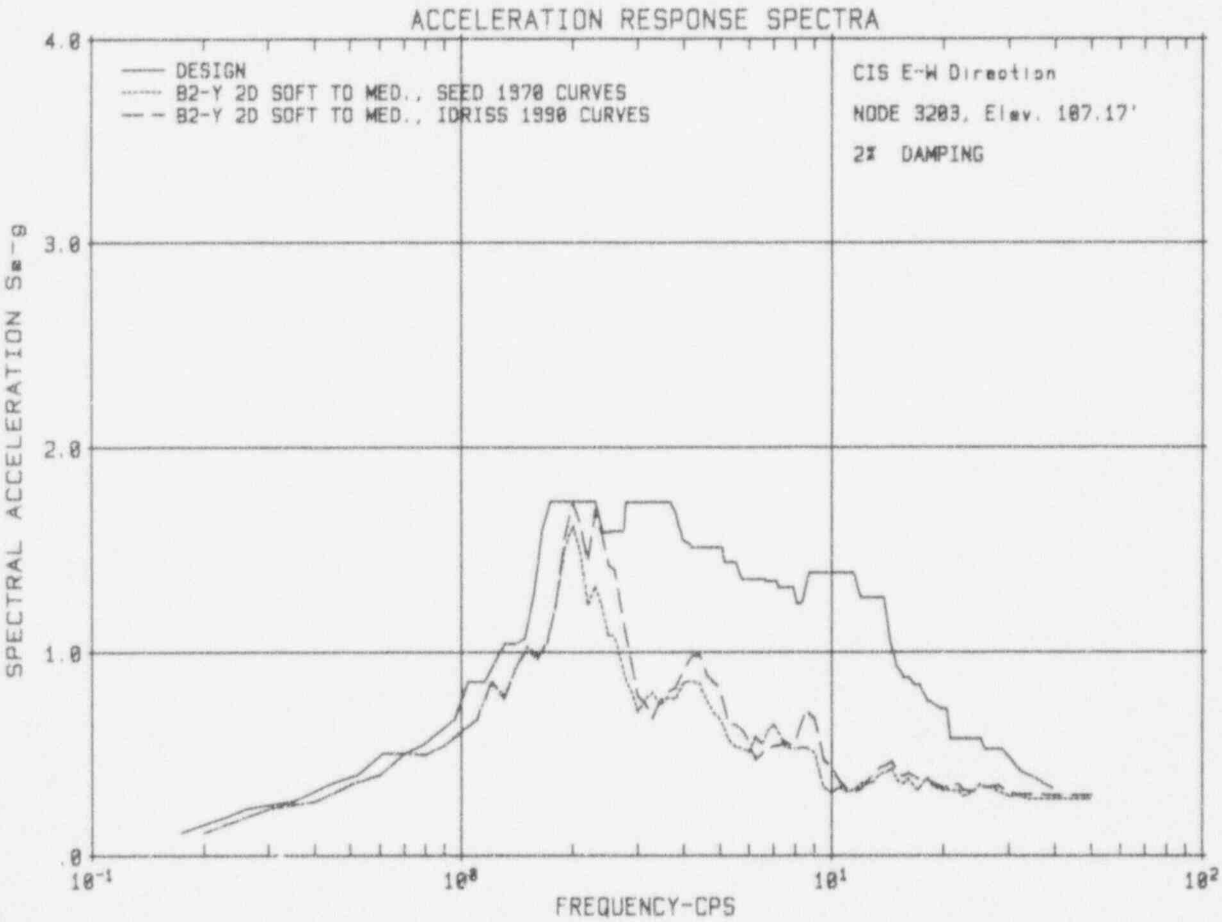


Figure 230.79-13
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation

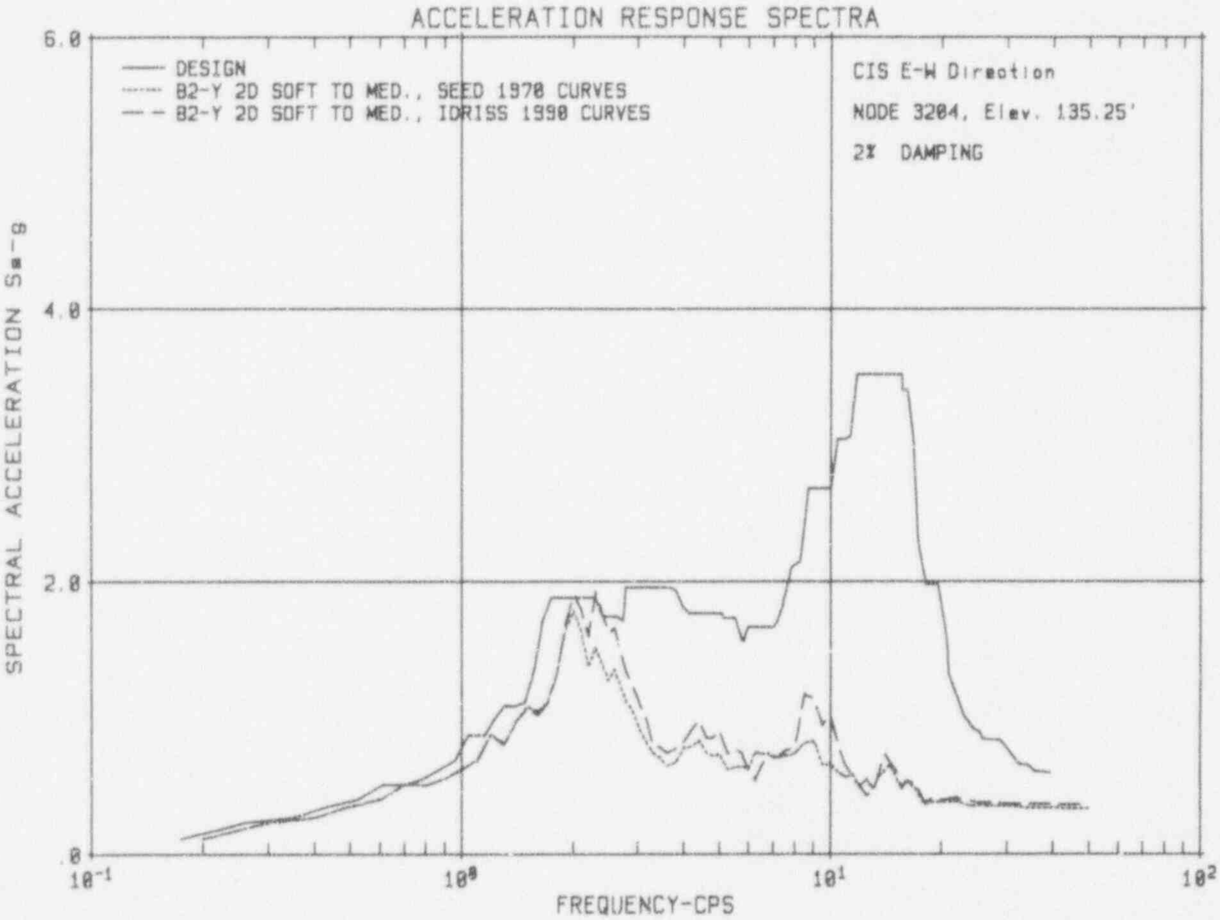
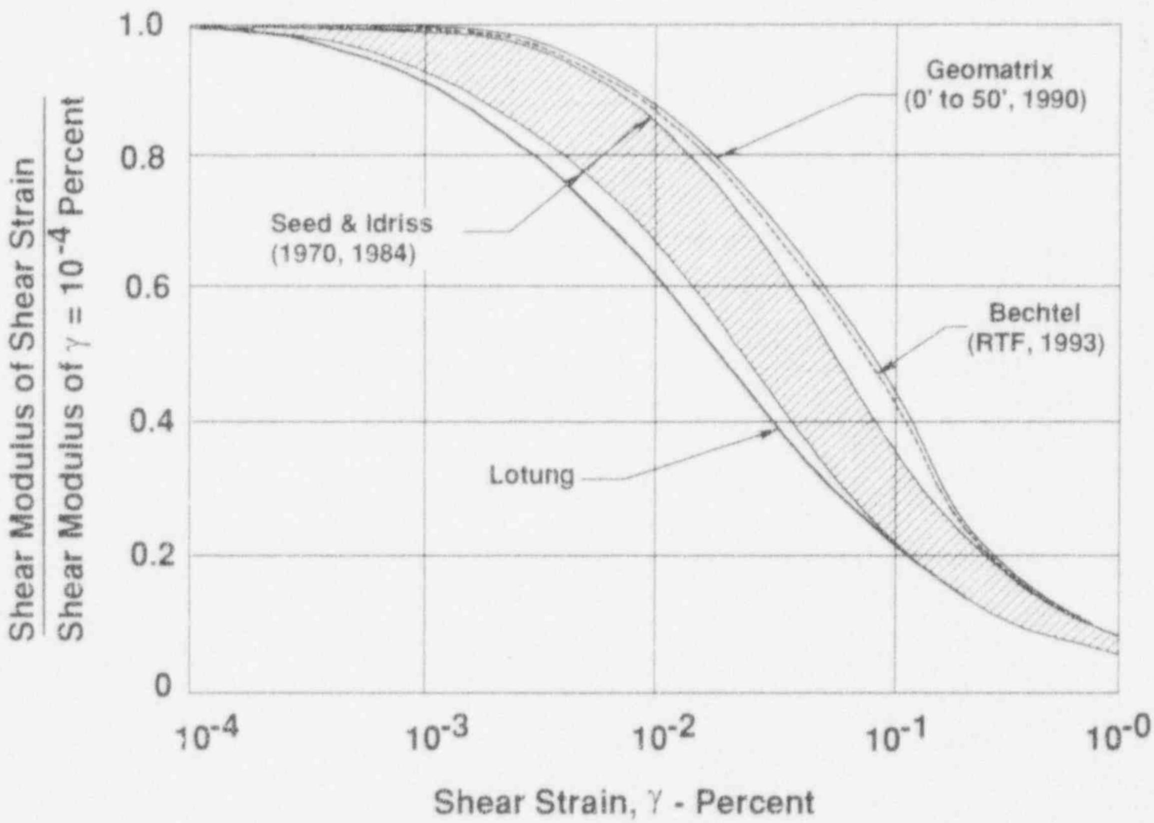
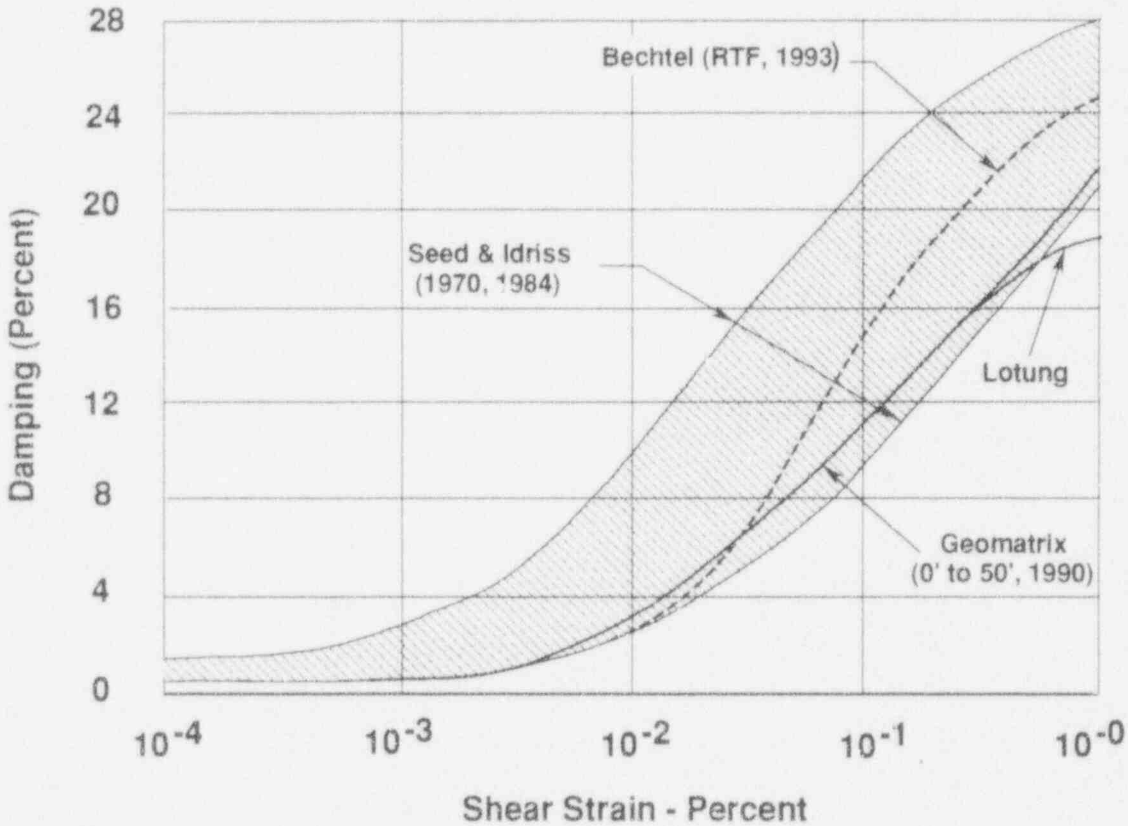


Figure 230.79-14
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation



Variation of Shear Modulus with Shear Strain for Sands

Figure 230.79-15
2D SASSI Analysis, E-W Direction
Soil Degradation Curve Evaluation



Damping Ratios For Sands

Question 231.29

It appears that the Poisson ratio values selected for soils above the water table may not be consistent with values normally expected for silty sands of densities high enough to support a shear wave velocity of 1000 fps. Evaluate and discuss the effect of the assumed Poisson ratio values on the SSI responses.

Response: (Revision 1)

~~The effect of the assumed Poisson's ratio values on the SSI responses will be submitted by July, 1994.~~

The case of soft-to-medium soil profile with the water table and base rock at 120 ft depth was analyzed using the 2D SSI model of the nuclear island in the NS direction. For this soil profile, a constant Poisson's ratio of 0.25 was used for all soil layers. The results of the SSI analysis for this case were compared with the results of the same soil case with the previously used Poisson's ratio of 0.35 in Figures 231.29-1 through 231.29-11. As shown in these figures, the effect of change in the Poisson's ratio on the SSI responses is relatively insignificant. It should be also noted that the design soil profiles which were identified from a series of 2D SSI analysis cases in the SSAR define the water table at grade level for all soil cases. For these profiles, the Poisson's ratios were adjusted to maintain the P-wave velocity of water.

SSAR Revision: NONE

Figure 231.29-1
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

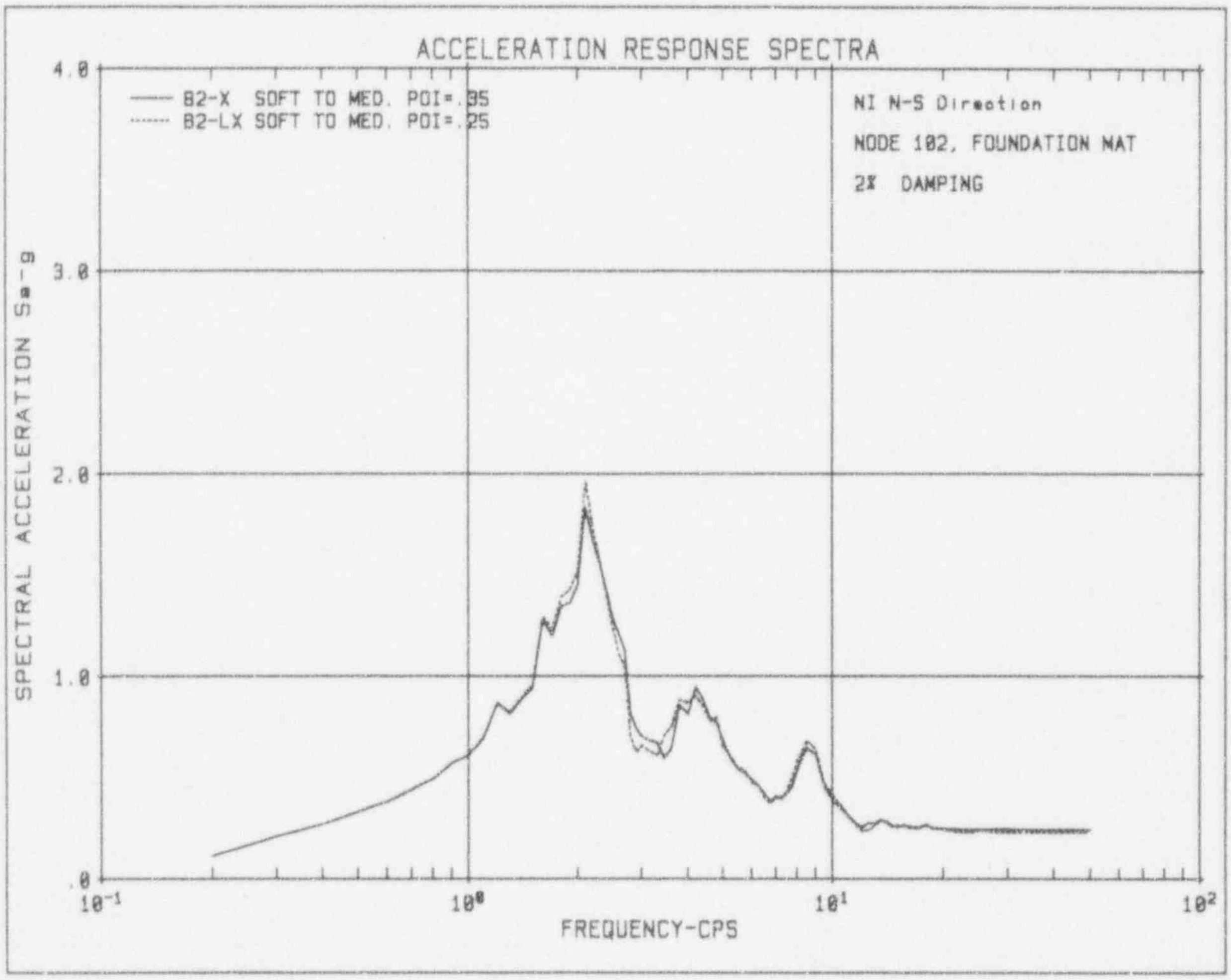


Figure 231.29-2
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

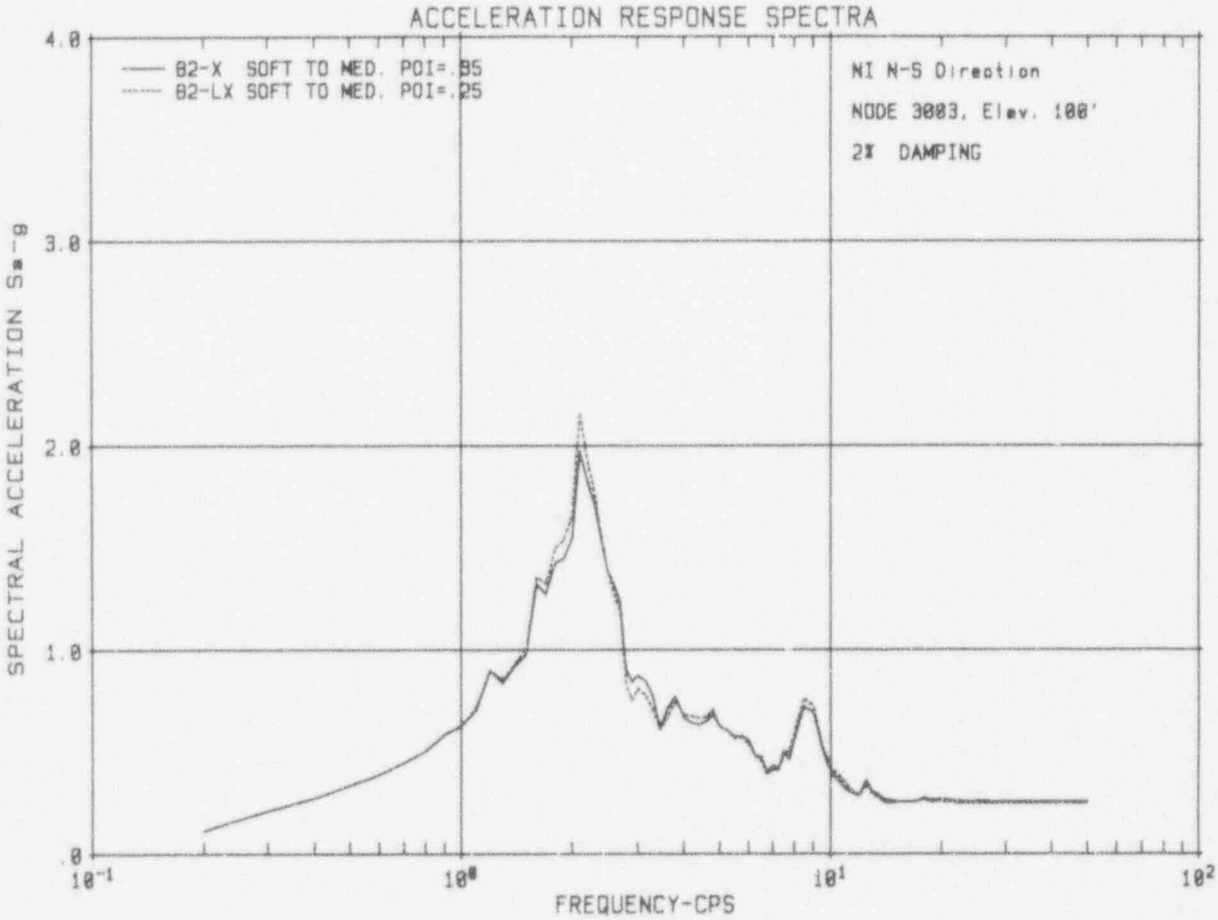


Figure 231.29-3
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

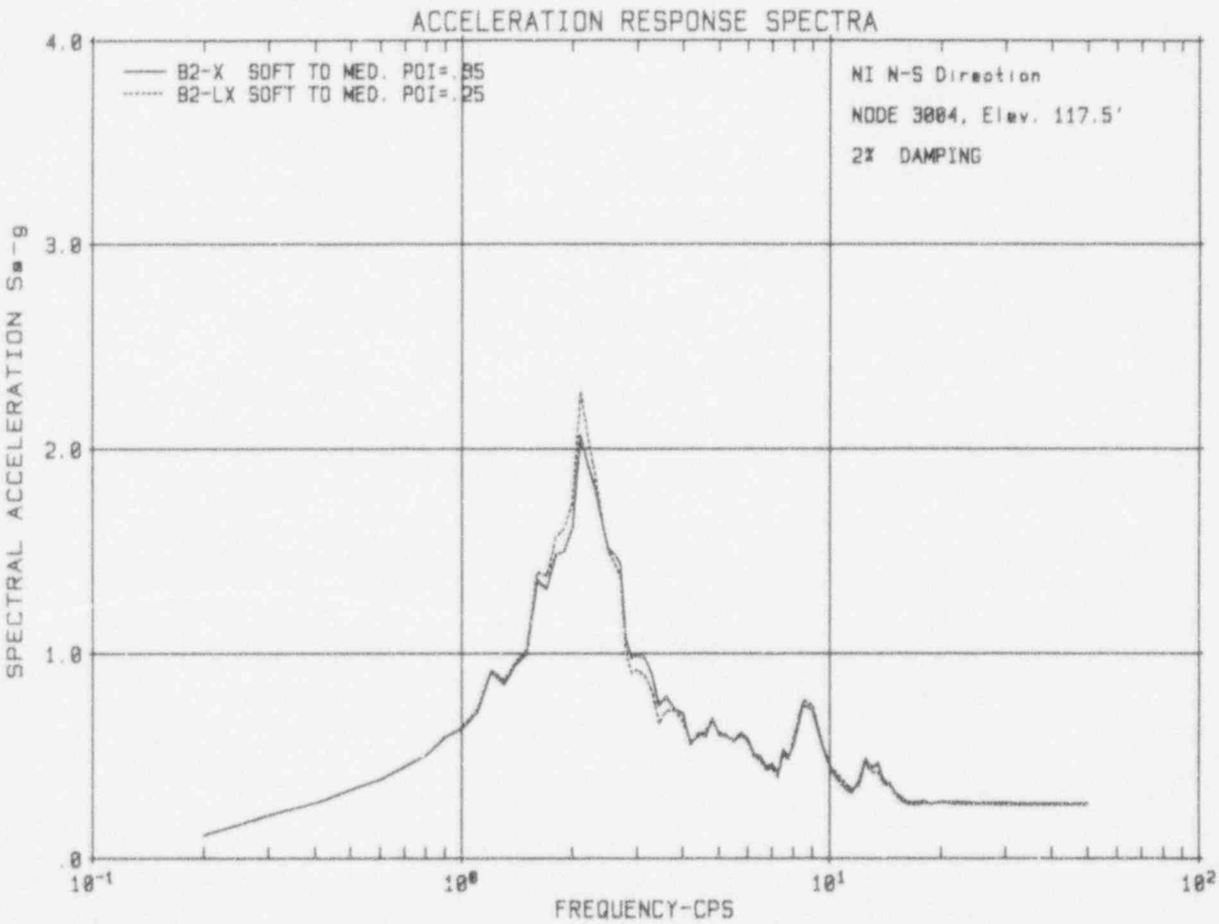


Figure 231.29-4
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

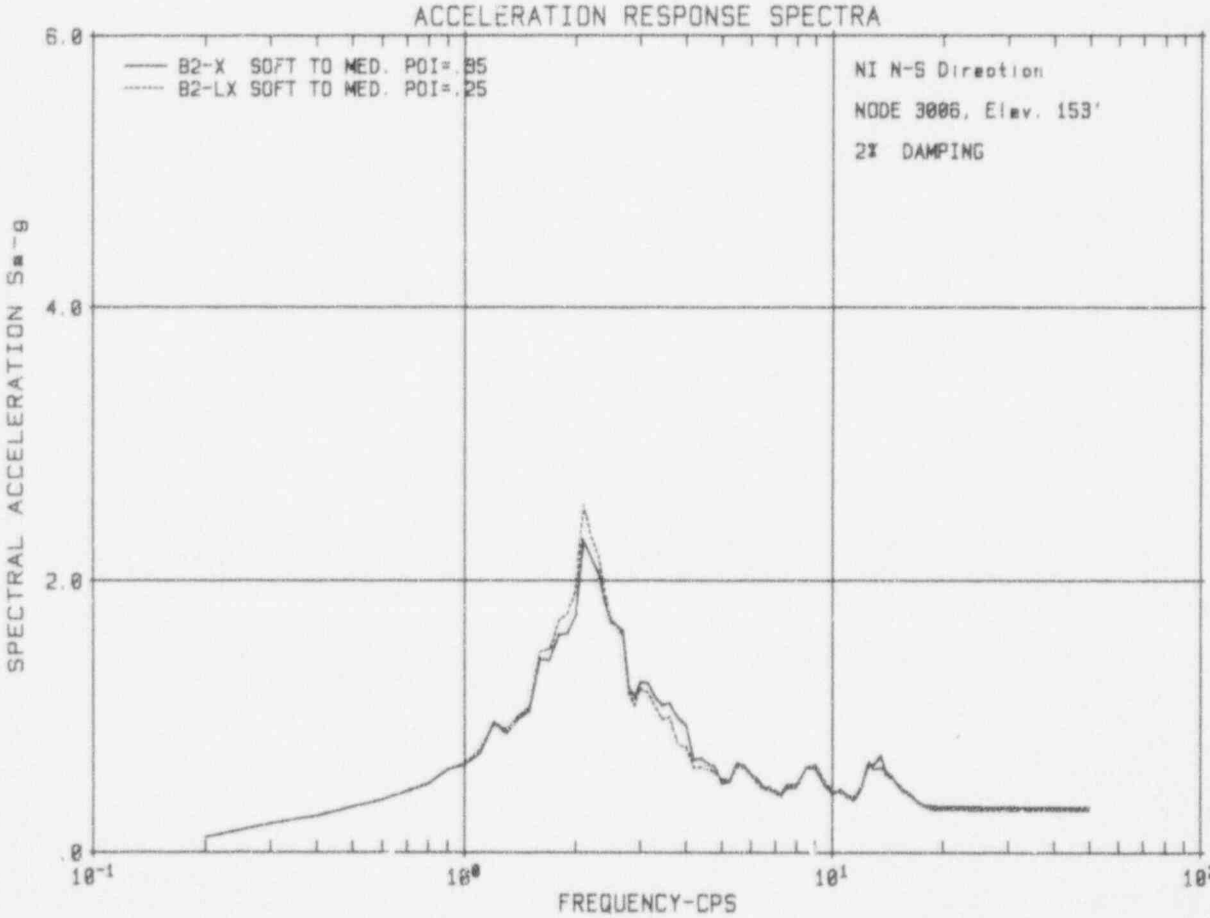


Figure 231.29-5
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

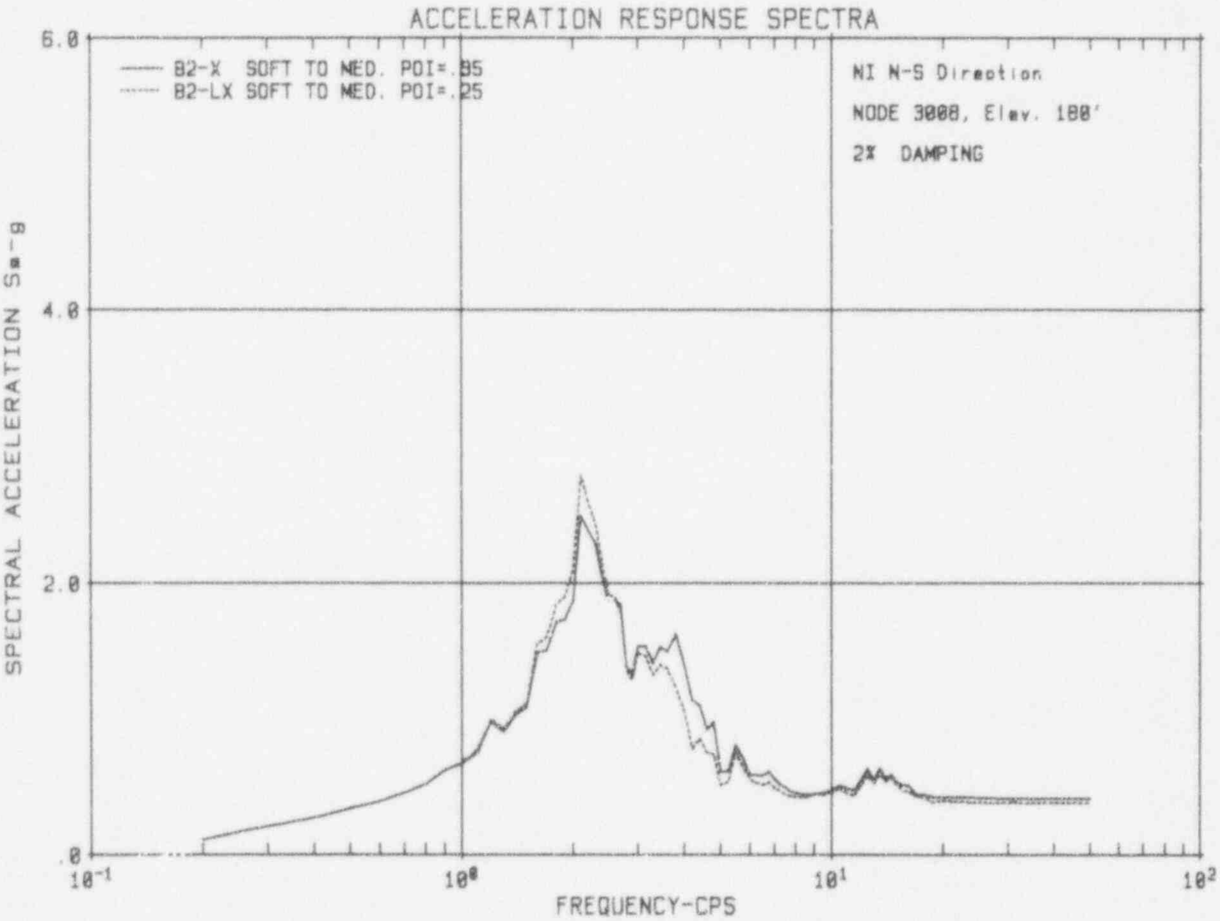


Figure 231.29-6
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

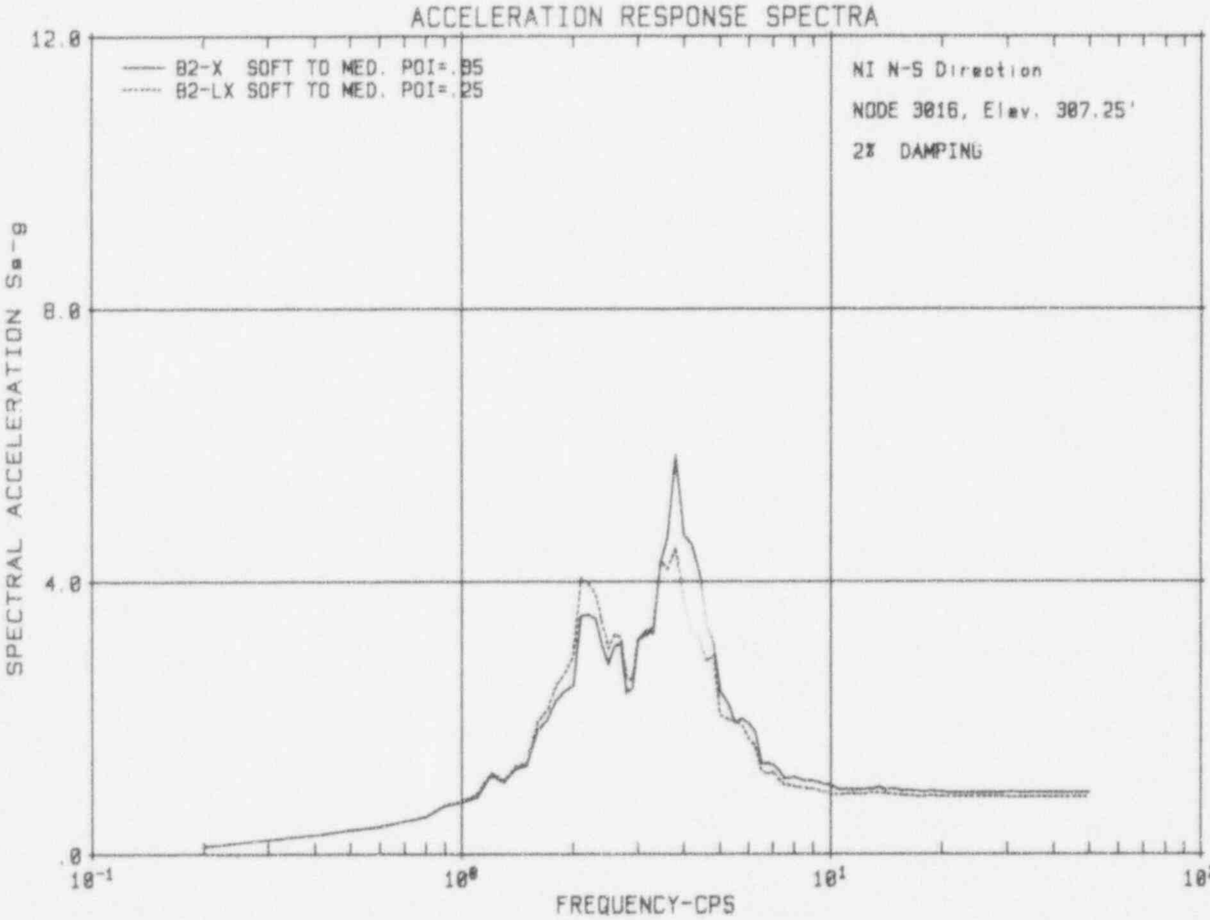


Figure 231.29-7
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

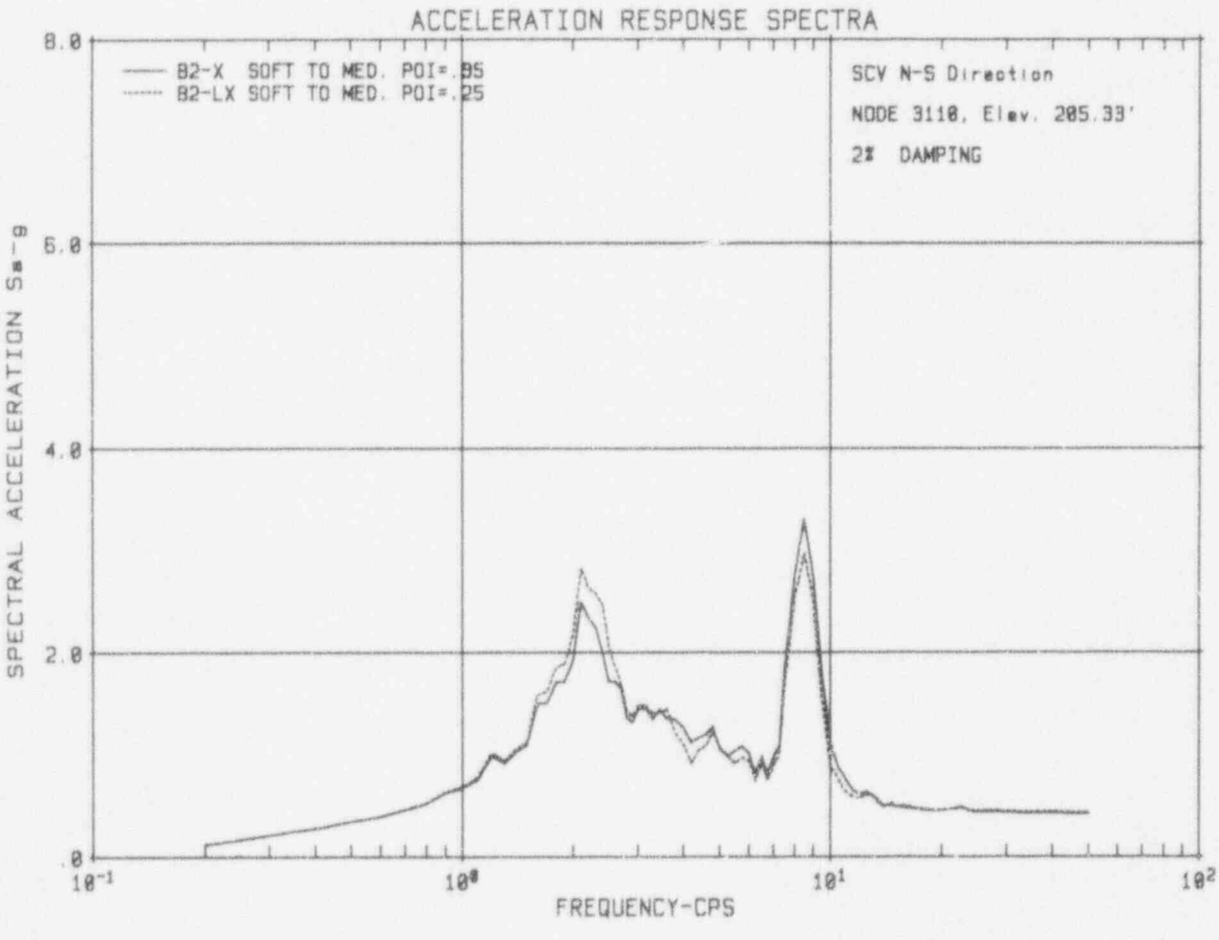


Figure 231.29-8
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

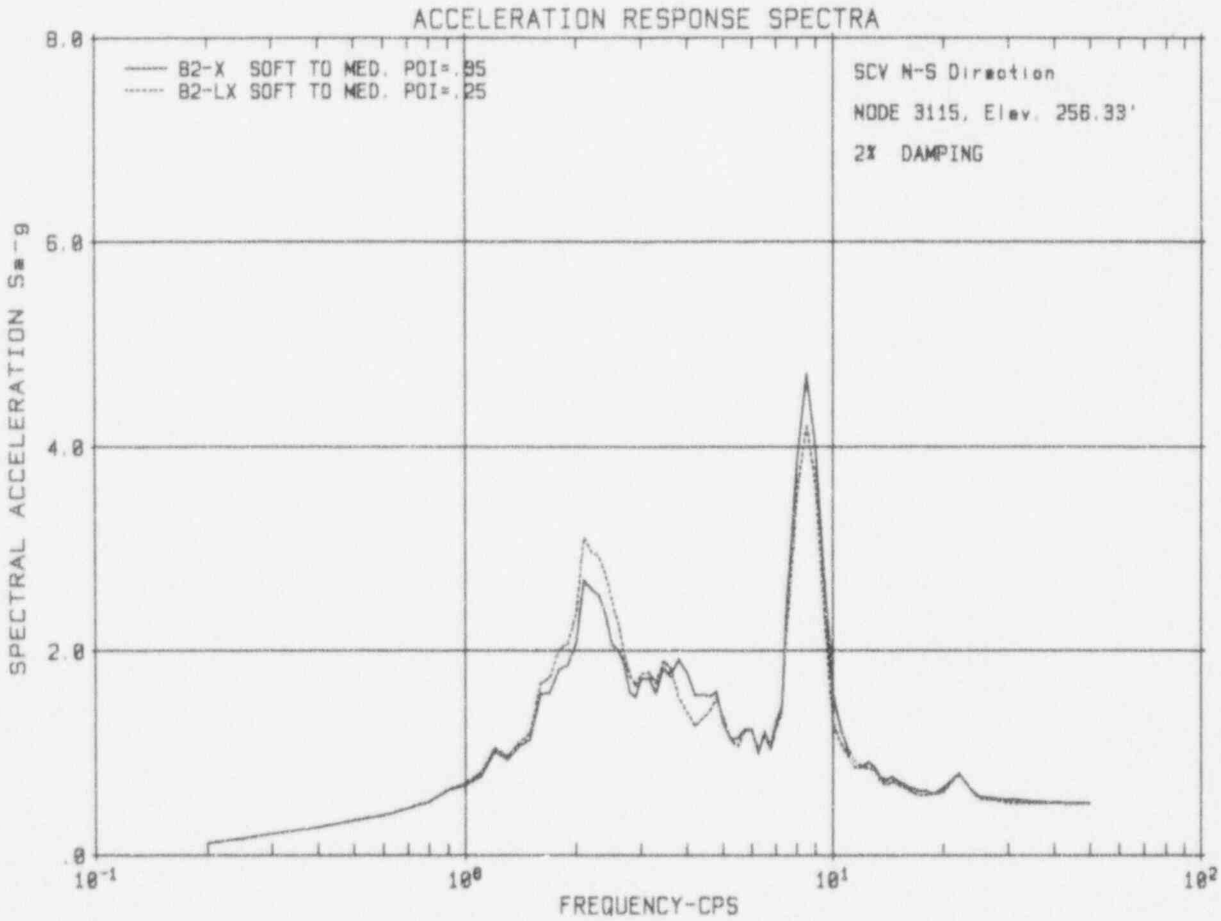


Figure 231.29-9
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

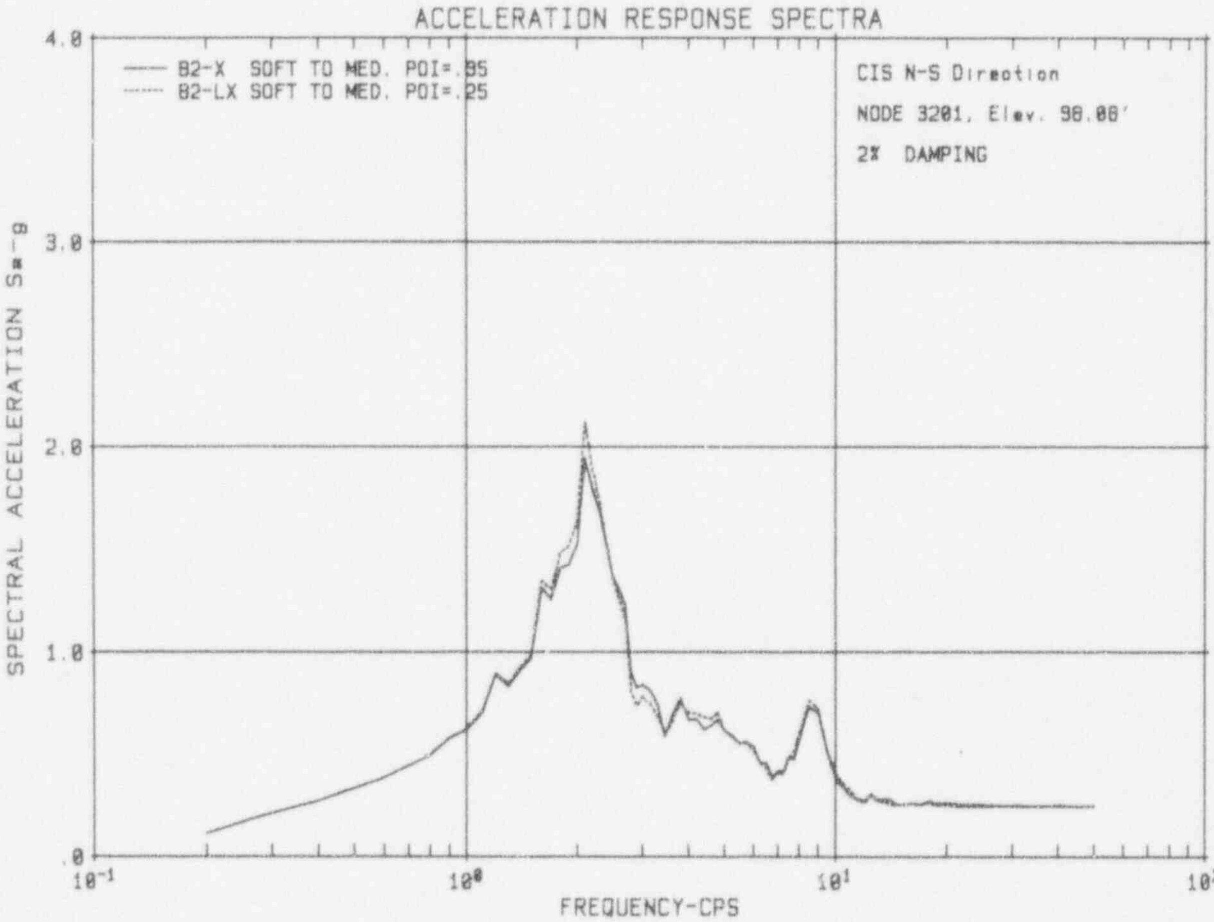


Figure 231.29-10
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation

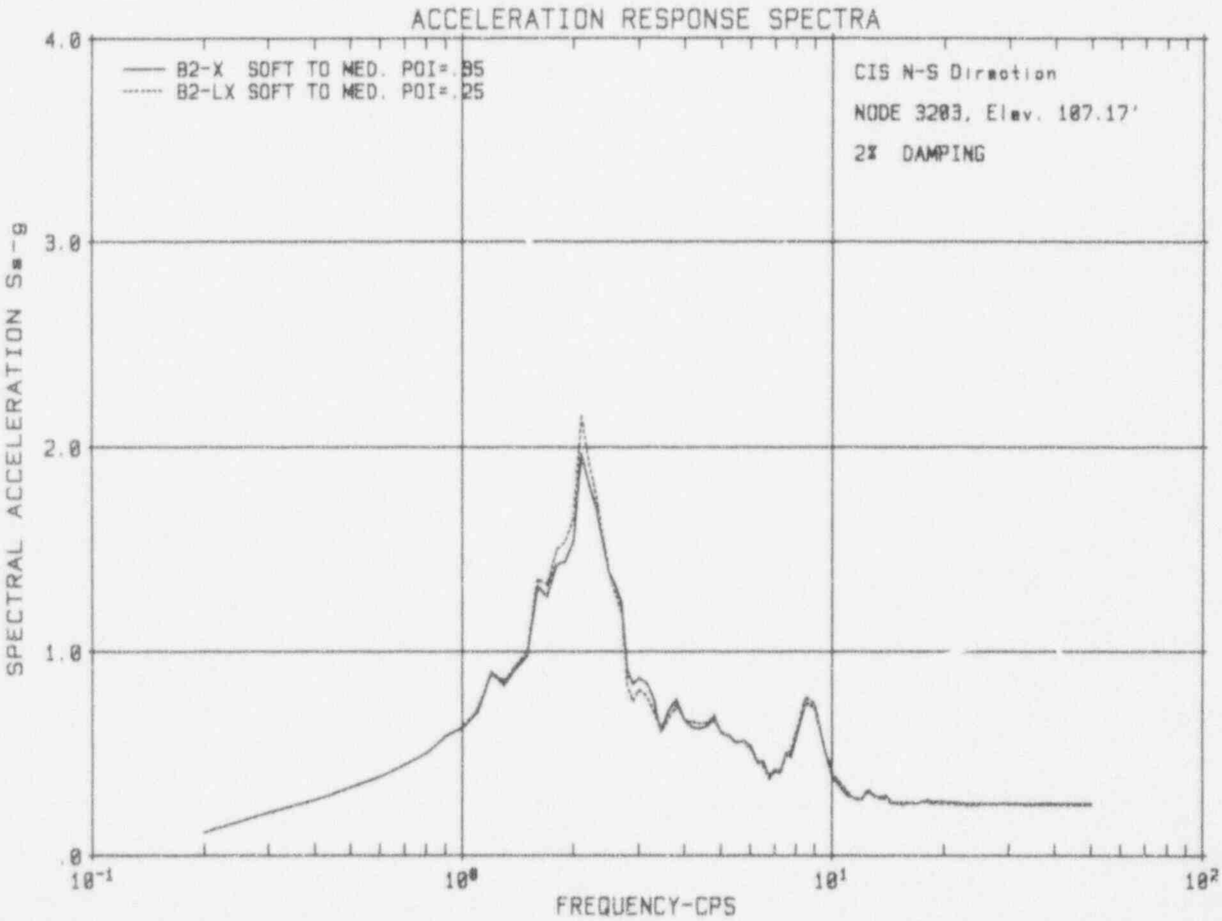
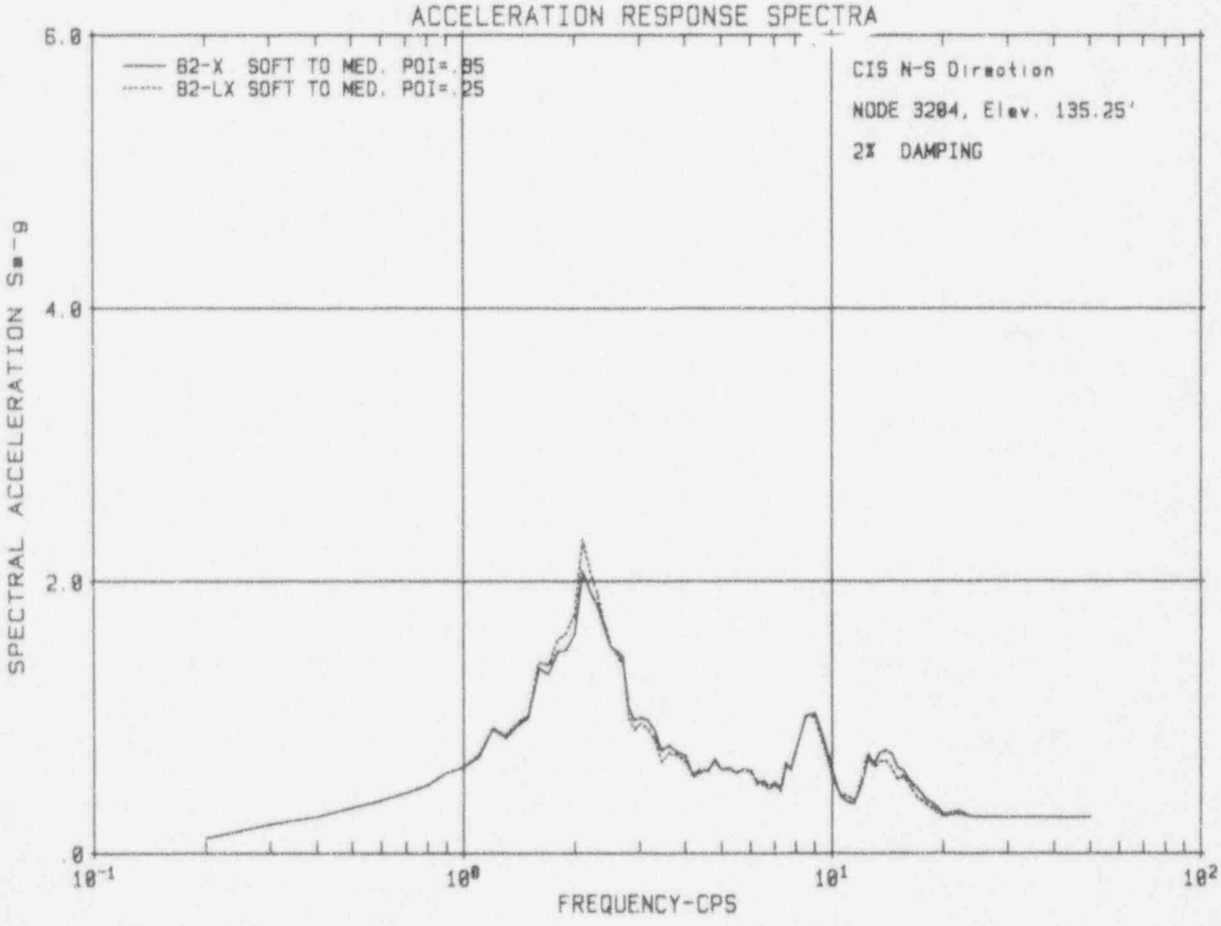


Figure 231.29-11
2D SASSI Analysis, N-S Direction
Poisson Ratio Evaluation



Question 410.145

Section 10.3.2.2.1 of the SSAR states that the main steam lines between the steam generator and the containment penetration are designed to meet the leak-before-break (LBB) criteria. The application of LBB in current PWRs is only for the reactor coolant system, which has a reactor coolant pressure boundary leakage detection system in accordance with RG 1.45. In order to apply LBB to the main steam lines, it has to have a steam line leak detection that is comparable to the reactor coolant pressure boundary leakage detection. Describe the main steam leak detection systems, instrumentation, acceptable leak criteria, and the requirements to be included in the plant technical specifications.

Response:

Main steam line leak detection is discussed in SSAR Subsection 10.3.3 which refers to Subsection 3.6.3 for a discussion of the leak-before-break application and criteria applicable to the main steam supply system. Subsection 3.6.3.3 states: "As noted in Subsection 5.2.5, the rated capability of the leak detection system for the primary coolant inside containment is 0.5 gpm in one hour. This system also detects leakage of 0.5 gpm from the main steam and feedwater lines inside containment.

Main steam line leak detection inside containment is provided by the following monitored parameters that are indicated in the main control room:

- Containment sump level monitor
- Containment air cooler condensate flow monitor
- Containment humidity
- Containment atmosphere temperature

The leak detection procedure is to set an alarm setpoint at 0.5 gpm in one hour on the sump level monitor. As explained in SSAR Subsection 5.2.5, "The sensitivity of the [containment sump] level sensors allows detection of leakage as low as 0.5 gpm within one hour." When the alarm actuates, the operator reviews other monitors (e.g., containment temperature and humidity, air cooler condensate flow) to determine the source of the leakage. Appropriate actions are then taken according to the technical specifications.

Technical Specification B.3.4.7.b, "RCS Operational Leakage," supports leak-before-break for piping 4" and greater; it requires that if leakage greater than 0.5 gpm is detected and cannot be corrected in 4 hours, the plant must enter LCO 3.0.3 immediately. That technical specification was written specifically for reactor coolant system piping, but the requirements are appropriate for detection of main steam line leakage inside containment. A similar technical specification, written specifically to support leak-before-break for secondary side leakage, will be added, requiring that if secondary side leakage greater than 0.5 gpm is detected and cannot be corrected in 4 hours, the plant must enter LCO 3.0.3 immediately.

SSAR Revision:

An additional Technical Specification to support LBB for the Main Steam Line piping, will be added in Rev. 2 of the SSAR.

PRA Revision: NONE

Question 410.160

Section 9.3.1.3 of the SSAR describes the use of safety-grade air accumulators or other devices to provide short-term operation of the safety-related pneumatic valves following loss of air.

a. Explain what are the "**other devices**."

b. Provide a list of all of the safety-related pneumatically operated valves (required to change valve position to achieve safe shutdown and accident mitigation) that are furnished with safety-related backup air accumulators and/or other devices.

c. How will the adequacy and reliability of the safety-related backup accumulators and or other devices be ensured? NUREG-1275, Vol. 2 recommends (1) periodic testing of safety-grade backup accumulator check valves for leakage; (2) monitoring and/or alarming accumulator pressure; and (3) verifying the adequacy of safety-related accumulators.

Response:

a) SSAR section 9.3.1.3 is reworded as shown below. The words "other devices" have been deleted.

b) There are no safety-related air operated valves that have safety-related air accumulators to support their safety-related functions. There are several safety-related valves that have N_2 stored in accumulators or inside the valve operator. The main steam isolation valves (MSIV) and the main feedwater isolation valves (MFIV) store N_2. The automatic depressurization system (ADS) 4th stage valves (if piston operated valves are used) store N_2 in separate accumulators.

Both the main steam isolation valves and main feedwater isolation valves have pneumatic / hydraulic operators. The stored energy for closing is supplied by high pressure nitrogen stored in the valve operator. The valve is maintained in its normally open position by high pressure hydraulic fluid that opposes the N_2 pressure. For emergency closure, redundant safety-related, Class 1E solenoid valves are energized by separate safety-related Class 1E power sources to dump the high pressure hydraulic fluid to a fluid reservoir.

The AP600 design change report transmitted to the NRC on 2/15/94 indicates that piston-operated gate valves are one of the valve types being considered for the automatic depressurization system fourth stage. These automatic depressurization system valves are normally closed "fail as is" type valves. To perform their safety function they are opened by energizing redundant safety-related, Class 1E solenoid valves to align the nitrogen supply stored in separate N_2 accumulators and to close the normally open vent under the piston while the area above the piston is kept vented (see response to RAI 410.162). The safety-related Class 1E solenoid valves are energized by redundant safety-related Class 1E power sources.

c) The adequacy and reliability of the safety-related N_2 supplies (accumulators/operators) will be provided by:

- integrated N_2 supply (accumulator/operator/piping) leak test during each refueling;

- N_2 supply pressure monitoring and alarm;
- verification of the adequacy of the N_2 supply during plant startup testing.

SSAR Revision:

The second paragraph of Subsection 9.3.1.3 will be revised as follows:

The compressed and instrument air system is required for normal operation and startup of the plant. Pneumatically operated valves in the plant, which are essential for safe shutdown and accident mitigation, are either designed to actuate to fail-safe position upon loss of air pressure ~~Any safety related, pneumatically operated valves required to change position to achieve safe shutdown and accident mitigation are provided with safety grade air accumulators or other devices to provide reliable short term operation of these valves following loss of air.~~ or are provided with safety-related air accumulators to provide the air supply for the safety-related function. Such pneumatically operated valves utilize safety-related solenoid valves to control the air supply. The air accumulators ~~and/or devices~~, if required, are included in the system containing the safety-related valves.

Question 410.162

Section 6.3.2.2.7.6 of the proprietary version of the SSAR discusses the use of backup safety-related air accumulators for the fourth stage ADS valves. Much of this type of information is typically found in non-proprietary versions of other SARs. Therefore, revise the SSAR to address and incorporate the following:

a. Include the information provided in the proprietary version of Section 6.3.2.2.7.6, regarding the backup safety-related air accumulators for the fourth stage ADS valves, in the text of the non-proprietary version of the SSAR.

b. Include these backup safety-related accumulators in a non-proprietary figure.

c. Revise the SSAR to include information about (1) leak testing the accumulators, (2) seismic qualification of the accumulators, (3) the ability of the accumulators to open the valves against maximum containment pressure, (4) the capacity of the accumulators, and (5) testing of the accumulators in accordance to RG 1.68.3.

d. IE Bulletin 80-01 concerns the operability of the pneumatic supply for ADS valves for licensees of GE BWR facilities. However, the bulletin may be relevant to the AP600 design regarding the use of backup safety-related air accumulators. Do the AP600 backup safety-related air accumulators conform with IE Bulletin No. 80-01?

Response:

a) Westinghouse is discussing the general issue of proprietary classification with the NRC staff. When those discussions are completed the SSAR will be revised as appropriate.

b) The AP600 design change report transmitted to the NRC on Febuary 15 1994 indicates that piston operated gate valves are one of the valve types being considered for the automatic depressurization system stage four valves. In the event that a piston operated gate valve is used, a backup safety-related N_2 accumulator configuration would be provided as shown in the attached sketch.

c) The design of the backup accumulators would provide:

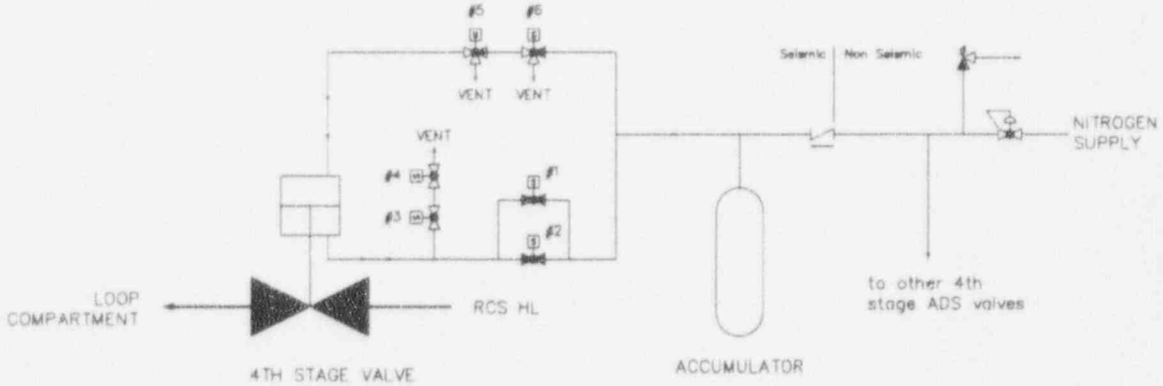
1) The accumulator / line would be leak tested each refueling (pressure decay).
2) The accumulator and the connected lines/valves up to the check isolation valve would be safety-related, seismic I design.
3) The accumulators have a requirement to open/close the automatic depressurization system stage valve two times with the containment at pressures as high as 45 psig. Note that N_2 is supplied at a pressure of about 350 psig.
4) The estimated accumulator capacity is about 200 gallons. Note that this capacity is dependent on the vendor specific valve/operator design.
5) The accumulator would be tested in accordance with the Regulatory Guide 1.68.3.

d) The backup safety-related N_2 accumulators will conform with IE Bulletin No. 80-01, in particular:

- isolation check valve type will be chosen to minimize the potential leakage;
- integrated accumulator/operator leak test will be performed during each refueling;
- the gas accumulator, associated solenoids valves, isolation check valve and piping are Seismic I;
- Technical Specification will be defined to address the automatic depressurization system stage 4th stage operability.

SSAR Revision: NONE

BACKUP SAFETY-RELATED ACCUMULATOR CONFIGURATION



4TH STAGE VALVES	SOLENOID VALVE DIVISION ASSIGNMENT #1	#2	#3	#4	#5	#6
V004A & B	A	B	A	B	A	B
V004C & D	C	D	C	D	C	D

The solenoid valves are shown in their normal positions, which are de-energized. To open the 4th stage ADS valve, solenoids #1, 2, 3 & 4 are energized which aligns the nitrogen supply to under the pistion and keeps the area above the pistion vented. To close the 4th stage ADS valve, solenoids #5 & 6 are energized which aligns the nitrogen supply to over the pistion and keeps the area under the valve vented.

Question 410.196

Include the table provided in the March 18, 1993, response to Q410.27 that lists the safety-related equipment requiring flood protection in the appropriate section of the SSAR. In addition, include the information in the February 9, 1993, response to Q435.56 regarding flood protection for I&C equipment in Section 3.4.1 of the SSAR. Include the caveats regarding information not in the table (i.e., regarding safety-related equipment above the maximum flood level and passive components).

Response:

The environmental qualification of safety-related equipment is addressed in SSAR Section 3.11. SSAR Table 3.11-1 lists all safety-related electrical and mechanical equipment and their environmental zones (room numbers). The data in the response to RAI 410.27 was provided for information and provides a level of detail beyond that necessary for the SSAR. The check valves and relief valves shown in the response to RAI 410.27 are being added to the list of active valves in SSAR table 3.11-1.

SSAR Section 3.4.1.1.2 will be revised to address RAI 435.56.

SSAR Revision:

The last paragraph in SSAR Section 3.4.1.1.2 and Table 3.11-1 will be revised as follows:

> The AP600 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components. As a result, component failures resulting from internal flooding will not prevent safe shutdown of the plant or prevent mitigation of the flooding event. Protection mechanisms are described in Sections 3.6 and 3.11. The protection mechanisms related to minimizing the consequences of internal flooding include the following.
>
> - Structural enclosures
> - Structural barriers
> - Curbs and elevated thresholds
> - Leak detection systems
> - Drain systems
> - Equipment qualification.

Table 3.11-1 (Sheet 18 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
ACTIVE DAMPERS:				
MCR ISOLATION DAMPERS	VBS MD D214	12401	ESF	1 YR
MCR ISOLATION DAMPERS	VBS MD D215	12401	ESF	1 YR
MCR ISOLATION DAMPERS	VBS MD D216	12401	ESF	1 YR
MCR ISOLATION DAMPERS	VBS MD D217	12401	ESF	1 YR
MCR ISOLATION DAMPERS	VBS MD D220	12401	ESF	1 YR
MCR ISOLATION DAMPERS	VBS MD D221	12401	ESF	1 YR
PENETRATIONS:				
PENETRATIONS (MECHANICAL)	SEE TABLE 6.2.3-1			
PENETRATIONS (ELECTRICAL)	SEE FIGURE 3.8.2-4			
ACTIVE VALVES:				
CONTAINMENT ISO-INLET	CCS PL V200	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CCS PL V200-LC	12306	PAMS	2 WKS
LIMIT SWITCH (OPEN)	CCS PL V200-LO	12306	PAMS	2 WKS
MOTOR OPERATOR	CCS PL V200-M	12306	ESF	5 MIN
CONTAINMENT ISO-OUTLET	CCS PL V207	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CCS PL V207-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	CCS PL V207-LO	11300	PAMS	1 YR
MOTOR OPERATOR	CCS PL V207-M	11300	ESF	5 MIN
CONTAINMENT ISO-OUTLET	CCS PL V208	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CCS PL V208-LC	12306	PAMS	2 WKS
LIMIT SWITCH (OPEN)	CCS PL V208-LO	12306	PAMS	2 WKS
MOTOR OPERATOR	CCS PL V208-M	12306	ESF	5 MIN
RCS LETDOWN STOP VALVE	CVS PL V001	11303	ESF	5 MIN
LIMIT SWITCH	CVS PL V001-L	11303	PAMS	1 YR
MOTOR OPERATOR	CVS PL V001-M	11303	ESF	5 MIN
RCS LETDOWN STOP VALVE	CVS PL V002	11303	ESF	5 MIN
LIMIT SWITCH	CVS PL V002-L	11303	PAMS	1 YR
MOTOR OPERATOR	CVS PL V002-M	11303	ESF	5 MIN
DEMIN FLUSH LINE RELIEF	CVS PL V042	11209	ESF	5 MIN
WLS LETDOWN IRC ISOLATION	CVS PL V045	11300	ESF	5 MIN
SOLENOID VALVE	CVS PL V045-S	11300	ESF	5 MIN
LETDOWN FLOW ORC ISO	CVS PL V047	12256	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CVS PL V047-LC	12256	PAMS	2 WKS
LIMIT SWITCH (OPEN)	CVS PL V047-LO	12256	PAMS	2 WKS
LETDOWN LINE RELIEF	CVS PL V056	11300	ESF	5 MIN

Table 3.11-1 (Sheet 19 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
RCS CHARGING STOP VALVE	CVS PL V081	11303	ESF	5 MIN
SOLENOID VALVE	CVS PL V081-S	11303	ESF	5 MIN
MAKEUP LINE CONT ISOLATION	CVS PL V090	12256	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CVS PL V090-LC	12256	PAMS	2 WKS
LIMIT SWITCH (OPEN)	CVS PL V090-LO	12256	PAMS	2 WKS
MOTOR OPERATOR	CVS PL V090-M	12256	ESF	5 MIN
MAKEUP LINE CONT ISOLATION	CVS PL V091	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CVS PL V091-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	CVS PL V091-LO	11300	PAMS	1 YR
MOTOR OPERATOR	CVS PL V091-M	11300	ESF	5 MIN
HYDROGEN ADDITION CONT ISO	CVS PL V092	12256	ESF	5 MIN
LIMIT SWITCH (CLOSED)	CVS PL V092-LC	12256	PAMS	2 WKS
LIMIT SWITCH (OPEN)	CVS PL V092-LO	12256	PAMS	2 WKS
SOLENOID VALVE	CVS PL V092-S	12256	ESF	5 MIN
DEMIN WATER SYS ISOLATION	CVS PL V136A	12255	ESF	5 MIN
LIMIT SWITCH	CVS PL V136A-L	12255	PAMS	2 WKS
MOTOR OPERATOR	CVS PL V136A-M	12255	ESF	5 MIN
DEMIN WATER SYS ISOLATION	CVS PL V136B	12255	ESF	5 MIN
LIMIT SWITCH	CVS PL V136B-L	12255	PAMS	2 WKS
MOTOR OPERATOR	CVS PL V136B-M	12255	ESF	5 MIN
PXS MAKEUP LINE CONT ISO VALVE	CVS PL V171	12454	ESF	5 MIN
SOLENOID VALVE	CVS PL V171-S	12454	ESF	5 MIN
PCCWST ISOLATION VALVE	PCS PL V001A	12701	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PCS PL V001A-LC	12701	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PCS PL V001A-LO	12701	PAMS	2 WKS
SOLENOID VALVE	PCS PL V001A-S	12701	ESF	5 MIN
PCCWST ISOLATION VALVE	PCS PL V001B	12701	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PCS PL V001B-LC	12701	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PCS PL V001B-LO	12701	PAMS	2 WKS
SOLENOID VALVE	PCS PL V001B-S	12701	ESF	5 MIN
PCCWST ISOLATION VALVE	PCS PL V002A	12701	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PCS PL V002A-LC	12701	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PCS PL V002A-LO	12701	PAMS	2 WKS
MOTOR OPERATOR	PCS PL V002A-M	12701	ESF	5 MIN
PCCWST ISOLATION VALVE	PCS PL V002B	12701	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PCS PL V002B-LC	12701	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PCS PL V002B-LO	12701	PAMS	2 WKS
MOTOR OPERATOR	PCS PL V002B-M	12701	ESF	5 MIN
CONT ISOL-AIR SAMPLE LINE	PSS PL V008	11300	ESF	5 MIN
CONT ISOL-AIR SAMPLE LINE	PSS PL V009	11300	ESF	5 MIN

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Table 3.11-1 (Sheet 20 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
CONT ISOL-LIQ SAMPLE LINE	PSS PL V010A	11300	ESF	5 MIN
CONT ISOL-LIQ SAMPLE LINE	PSS PL V010B	11300	ESF	5 MIN
CONT ISOL-LIQ SAMPLE LINE	PSS PL V011	12354	ESF	5 MIN
CONT ISOL-SAMPLE RETURN LINE	PSS PL V023	12354	ESF	5 MIN
CONT ISOL-AIR SAMPLE LINE	PSS PL V046	12454	ESF	5 MIN
CMT A CL INLET ISOLATION	PXS PL V002A	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V002A-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V002A-LO	11300	PAMS	1 YR
SOLENOID VALVE	PXS PL V002A-S	11300	ESF	5 MIN
CMT B CL INLET ISOLATION	PXS PL V002B	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V002B-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V002B-LO	11300	PAMS	1 YR
SOLENOID VALVE	PXS PL V002B-S	11300	ESF	5 MIN
CMT A CL INLET ISOLATION	PXS PL V003A	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V003A-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V003A-LO	11300	PAMS	1 YR
SOLENOID VALVE	PXS PL V003A-S	11300	ESF	5 MIN
CMT B CL INLET ISOLATION	PXS PL V003B	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V003B-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V003B-LO	11300	PAMS	1 YR
SOLENOID VALVE	PXS PL V003B-S	11300	ESF	5 MIN
CMT A PZR LINE ISOLATION	PXS PL V005A	12503	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V005A-LC	12503	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PXS PL V005A-LO	12503	PAMS	2 WKS
MOTOR OPERATOR	PXS PL V005A-M	12503	ESF	5 MIN
CMT B PZR LINE ISOLATION	PXS PL V005B	12503	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V005B-LC	12503	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PXS PL V005B-LO	12503	PAMS	2 WKS
MOTOR OPERATOR	PXS PL V005B-M	12503	ESF	5 MIN
CMT A DISCHARGE ISOLATION	PXS PL V014A	11206	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V014A-LC	11206	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V014A-LO	11206	PAMS	1 YR
SOLENOID VALVE	PXS PL V014A-S	11206	ESF	5 MIN
CMT B DISCHARGE ISOLATION	PXS PL V014B	11207	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V014B-LC	11207	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V014B-LO	11207	PAMS	1 YR
SOLENOID VALVE	PXS PL V014B-S	11207	ESF	5 MIN
CMT A DISCHARGE ISOLATION	PXS PL V015A	11206	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V015A-LC	11206	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V015A-LO	11206	PAMS	1 YR

Table 3.11-1 (Sheet 21 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
SOLENOID VALVE	PXS PL V015A-S	11206	ESF	5 MIN
CMT B DISCHARGE ISOLATION	PXS PL V015B	11207	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V015B-LC	11207	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V015B-LO	11207	PAMS	1 YR
SOLENOID VALVE	PXS PL V015B-S	11207	ESF	5 MIN
CMT A DISCHARGE	PXS PL V016A	11206	ESF	5 MIN
CMT B DISCHARGE	PXS PL V016B	11207	ESF	5 MIN
CMT A DISCHARGE	PXS PL V017A	11206	ESF	5 MIN
CMT B DISCHARGE	PXS PL V017B	11207	ESF	5 MIN
ACCUM A DISCHARGE ISOL	PXS PL V027A	11206	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V027A-LC	11206	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V027A-LO	11206	PAMS	1 YR
MOTOR OPERATOR	PXS PL V027A-M	11206	ESF	5 MIN
ACCUM B DISCHARGE ISOL	PXS PL V027B	11207	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V027B-LC	11207	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V027B-LO	11207	PAMS	1 YR
MOTOR OPERATOR	PXS PL V027B-M	11207	ESF	5 MIN
ACC A DISCHARGE	PXS PL V028A	11206	ESF	5 MIN
ACC B DISCHARGE	PXS PL V028B	11207	ESF	5 MIN
ACC A DISCHARGE	PXS PL V029A	11206	ESF	5 MIN
ACC B DISCHARGE	PXS PL V029B	11207	ESF	5 MIN
CMT A STEAM TRAP BYPASS ISOLATION	PXS PL V030A	11300	ESF	5 MIN
CMT B STEAM TRAP BYPASS ISOLATION	PXS PL V030B	11300	ESF	5 MIN
CMT A STEAM TRAP BYPASS ISOLATION	PXS PL V031A	11300	ESF	5 MIN
CMT B STEAM TRAP BYPASS ISOLATION	PXS PL V031B	11300	ESF	5 MIN
CMT A STEAM TRAP DISCH ISOL	PXS PL V033A	11300	ESF	5 MIN
CMT B STEAM TRAP DISCH ISOL	PXS PL V033B	11300	ESF	5 MIN
ORC NITROGEN SUPPLY CONT ISOLATION	PXS PL V042	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V042-LC	12306	PAMS	2 WKS
LIMIT SWITCH (OPEN)	PXS PL V042-LO	12306	PAMS	2 WKS
SOLENOID VALVE	PXS PL V042-S	12306	ESF	5 MIN
PRHR HX INLET ISOLATION	PXS PL V101	11500	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V101-LC	11500	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V101-LO	11500	PAMS	1 YR
MOTOR OPERATOR	PXS PL V101-M	11500	ESF	5 MIN

AP600

Table 3.11-1 (Sheet 22 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
PRHR HX DISCHARGE ISOL	PXS PL V108A	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V108A-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V108A-LO	11300	PAMS	1 YR
PRHR HX DISCHARGE ISOL	PXS PL V108B	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V108B-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V108B-LO	11300	PAMS	1 YR
RECIRC SUMP A ISOLATION	PXS PL V117A	11206	ESF	5 MIN
LIMIT SWITCH	PXS PL V117A-L	11206	PAMS	1 YR
MOTOR OPERATOR	PXS PL V117A-M	11206	ESF	5 MIN
RECIRC SUMP B ISOLATION	PXS PL V117B	11207	ESF	5 MIN
LIMIT SWITCH	PXS PL V117B-L	11207	PAMS	1 YR
MOTOR OPERATOR	PXS PL V117B-M	11207	ESF	5 MIN
RECIRC SUMP A ISOLATION	PXS PL V118A	11206	ESF	5 MIN
LIMIT SWITCH	PXS PL V118A-L	11206	PAMS	1 YR
MOTOR OPERATOR	PXS PL V118A-M	11206	ESF	5 MIN
RECIRC SUMP B ISOLATION	PXS PL V118B	11207	ESF	5 MIN
LIMIT SWITCH	PXS PL V118B-L	11207	PAMS	1 YR
MOTOR OPERATOR	PXS PL V118B-M	11207	ESF	5 MIN
RECIRC SUMP A	PXS PL V119A	11206	ESF	24 HRS
RECIRC SUMP B	PXS PL V119B	11207	ESF	24 HRS
RECIRC SUMPA	PXS PL V120A	11206	ESF	24 HRS
RECIRC SUMP B	PXS PL V120B	11207	ESF	24 HRS
IRWST/SUMP GRAV INJ A ISOL	PXS PL V121A	11206	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V121A-LC	11206	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V121A-LO	11206	PAMS	1 YR
MOTOR OPERATOR	PXS PL V121A-M	11206	ESF	5 MIN
IRWST/SUMP GRAV INJ B ISOL	PXS PL V121B	11207	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V121B-LC	11207	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V121B-LO	11207	PAMS	1 YR
MOTOR OPERATOR	PXS PL V121B-M	11207	ESF	5 MIN
IRWST INJ A	PXS PL V122A	11206	ESF	24 HRS
IRWST INJ B	PXS PL V122B	11207	ESF	24 HRS
IRWST INJ A	PXS PL V123A	11206	ESF	24 HRS
IRWST INJ B	PXS PL V123B	11207	ESF	24 HRS
IRWST INJ A	PXS PL V124A	11206	ESF	24 HRS
IRWST INJ B	PXS PL V124B	11207	ESF	24 HRS
IRWST INJ A	PXS PL V125A	11206	ESF	24 HRS
IRWST INJ B	PXS PL V125B	11207	ESF	24 HRS
IRWST GUTTER BYPASS A ISOL	PXS PL V130A	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V130A-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V130A-LO	11300	PAMS	1 YR

Table 3.11-1 (Sheet 23 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
SOLENOID VALVE	PXS PL V130A-S	11300	ESF	5 MIN
IRWST GUTTER BYPASS B ISOL	PXS PL V130B	11300	ESF	5 MIN
LIMIT SWITCH (CLOSED)	PXS PL V130B-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	PXS PL V130B-LO	11300	PAMS	1 YR
SOLENOID VALVE	PXS PL V130B-S	11300	ESF	5 MIN
CMT A FILL ISOLATION	PXS PL V230A	11300	ESF	5 MIN
SOLENOID VALVE	PXS PL V230A-S	11300	ESF	5 MIN
CMT B FILL ISOLATION	PXS PL V230B	11300	ESF	5 MIN
SOLENOID VALVE	PXS PL V230B-S	11300	ESF	5 MIN
ACCUM A FILL/DRAIN ISOL	PXS PL V232A	11300	ESF	5 MIN
SOLENOID VALVE	PXS PL V232A-S	11300	ESF	5 MIN
ACCUM B FILL/DRAIN ISOL	PXS PL V232B	11300	ESF	5 MIN
SOLENOID VALVE	PXS PL V232B-S	11300	ESF	5 MIN
PH ADJUST TANK DISCH ISOL	PXS PL V301A	11300	ESF	5 MIN
PH ADJUST TANK DISCH ISOL	PXS PL V301B	11300	ESF	5 MIN
1ST STAGE ADS	RCS PL V001A	11603	ESF	5 MIN
LIMIT SWITCH (CLOSED)	RCS PL V001A-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V001A-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V001A-M	11603	ESF	5 MIN
1ST STAGE ADS	RCS PL V001B	11603	ESF	5 MIN
LIMIT SWITCH (CLOSED)	RCS PL V001B-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V001B-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V001B-M	11603	ESF	5 MIN
1ST STAGE ADS	RCS PL V001C	11603	ESF	5 MIN
LIMIT SWITCH (CLOSED)	RCS PL V001C-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V001C-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V001C-M	11603	ESF	5 MIN
1ST STAGE ADS	RCS PL V001D	11603	ESF	5 MIN
LIMIT SWITCH (CLOSED)	RCS PL V001D-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V001D-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V001D-M	11603	ESF	5 MIN
2ND STAGE ADS	RCS PL V002A	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V002A-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V002A-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V002A-M	11603	ESF	24 HR
2ND STAGE ADS	RCS PL V002B	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V002B-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V002B-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V002B-M	11603	ESF	24 HR
2ND STAGE ADS	RCS PL V002C	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V002C-LC	11603	PAMS	1 YR

Table 3.11-1 (Sheet 24 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
LIMIT SWITCH (OPEN)	RCS PL V002C-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V002C-M	11603	ESF	24 HR
2ND STAGE ADS	RCS PL V002D	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V002D-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V002D-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V002D-M	11603	ESF	24 HR
3RD STAGE ADS	RCS PL V003A	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V003A-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V003A-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V003A-M	11603	ESF	24 HR
3RD STAGE ADS	RCS PL V003B	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V003B-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V003B-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V003B-M	11603	ESF	24 HR
3RD STAGE ADS	RCS PL V003C	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V003C-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V003C-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V003C-M	11603	ESF	24 HR
3RD STAGE ADS	RCS PL V003D	11603	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V003D-LC	11603	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V003D-LO	11603	PAMS	1 YR
MOTOR OPERATOR	RCS PL V003D-M	11603	ESF	24 HR
4TH STAGE ADS	RCS PL V004A	11300	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V004A-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V004A-LO	11300	PAMS	1 YR
4TH STAGE ADS	RCS PL V004B	11300	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V004B-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V004B-LO	11300	PAMS	1 YR
4TH STAGE ADS	RCS PL V004C	11300	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V004C-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V004C-LO	11300	PAMS	1 YR
4TH STAGE ADS	RCS PL V004D	11300	ESF	24 HR
LIMIT SWITCH (CLOSED)	RCS PL V004D-LC	11300	PAMS	1 YR
LIMIT SWITCH (OPEN)	RCS PL V004D-LO	11300	PAMS	1 YR
PRESS EQUALIZATION VALVES	RCS PL V006A	11301	ESF	5 MIN
PRESS EQUALIZATION VALVES	RCS PL V006B	11301	ESF	5 MIN
PRESS EQUALIZATION VALVES	RCS PL V006C	11302	ESF	5 MIN
PRESS EQUALIZATION VALVES	RCS PL V006D	11302	ESF	5 MIN
ADS TEST SOLENOID VALVE	RCS PL V007A	11603	ESF	5 MIN
ADS TEST SOLENOID VALVE	RCS PL V007B	11603	ESF	5 MIN
ADS TEST SOLENOID VALVE	RCS PL V007C	11603	ESF	5 MIN

Table 3.11-1 (Sheet 25 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
ADS TEST SOLENOID VALVE	RCS PL V007D	11603	ESF	5 MIN
RCS HEAD VENT ADS VALVE	RCS PL V152	11300	ESF	5 MIN
RCS HEAD VENT ADS VALVE	RCS PL V153	11300	ESF	5 MIN
RCS INNER SUCTION ISOLATION	RNS PL V001A	11208	ESF	5 MIN
LIMIT SWITCH	RNS PL V001A-L	11208	PAMS	1 YR
MOTOR OPERATOR	RNS PL V001A-M	11208	ESF	5 MIN
RCS INNER SUCTION ISOLATION	RNS PL V001B	11208	ESF	5 MIN
LIMIT SWITCH	RNS PL V001B-L	11208	PAMS	1 YR
MOTOR OPERATOR	RNS PL V001B-M	11208	ESF	5 MIN
RCS OUTER SUCTION ISOL	RNS PL V002A	11208	ESF	5 MIN
LIMIT SWITCH	RNS PL V002A-L	11208	PAMS	1 YR
MOTOR OPERATOR	RNS PL V002A-M	11208	ESF	5 MIN
RCS OUTER SUCTION ISOL	RNS PL V002B	11208	ESF	5 MIN
LIMIT SWITCH	RNS PL V002B-L	11208	PAMS	1 YR
MOTOR OPERATOR	RNS PL V002B-M	11208	ESF	5 MIN
RHR CONTROL/ISOL VALVE	RNS PL V011	12253	ESF	5 MIN
LIMIT SWITCH	RNS PL V011-L	12253	PAMS	2 WKS
MOTOR OPERATOR	RNS PL V011-M	12253	ESF	5 MIN
RNS DISCHARGE CONT ISOL	RNS PL V013	11206	ESF	5 MIN
RNS DISCHARGE RCPB ISOL	RNS PL V015A	11206	ESF	5 MIN
RNS DISCHARGE RCPB ISOL	RNS PL V015B	11207	ESF	5 MIN
RNS DISCHARGE RCPB ISOL	RNS PL V017A	11206	ESF	5 MIN
RNS DISCHARGE RCPB ISOL	RNS PL V107B	11207	ESF	5 MIN
RNS HOT LEG SUCTION RELIEF	RNS PL V021	11206	ESF	5 MIN
RHR PUMP SUCTION HDR ISOL	RNS PL V022	12253	ESF	5 MIN
LIMIT SWITCH	RNS PL V022-L	12253	PAMS	2 WKS
MOTOR OPERATOR	RNS PL V022-M	12253	ESF	5 MIN
IRWST SUCTION LINE ISOL	RNS PL V023	11208	ESF	5 MIN
LIMIT SWITCH	RNS PL V023-L	11208	PAMS	1 YR
MOTOR OPERATOR	RNS PL V023-M	11208	ESF	5 MIN
CONTAINMENT ISOLATION	SFS PL V034	11206	ESF	5 MIN
LIMIT SWITCH	SFS PL V034-L	11206	PAMS	1 YR
MOTOR OPERATOR	SFS PL V034-M	11206	ESF	5 MIN
CONTAINMENT ISOLATION	SFS PL V035	12354	ESF	5 MIN
LIMIT SWITCH	SFS PL V035-L	12354	PAMS	2 WKS
MOTOR OPERATOR	SFS PL V035-M	12354	ESF	5 MIN
CONTAINMENT ISOLATION	SFS PL V038	12354	ESF	5 MIN
LIMIT SWITCH	SFS PL V038-L	12354	PAMS	2 WKS
MOTOR OPERATOR	SFS PL V038-M	12354	ESF	5 MIN
SFS SUCTION THERMAL RELIEF	EFE PL V048	11206	ESF	5 MIN

AP600

Table 3.11-1 (Sheet 26 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
PORV BLOCK VALVE	SGS PL V027A	12406	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V027A-LC	12406	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V027A-LO	12406	PAMS	2 WKS
MOTOR OPERATOR	SGS PL V027A-M	12406	ESF	5 MIN
PORV BLOCK VALVE	SGS PL V027B	12404	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V027B-LC	12404	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V027B-LO	12404	PAMS	2 WKS
MOTOR OPERATOR	SGS PL V027B-M	12404	ESF	5 MIN
STEAM LINE COND DRAIN ISOL	SGS PL V036A	12406	ESF	5 MIN
SOLENOID VALVE	SGS PL V036A-S	12406	ESF	5 MIN
STEAM LINE CONDENSATE ISOL	SGS PL V036B	12404	ESF	5 MIN
SOLENOID VALVE	SGS PL V036B-S	12404	ESF	5 MIN
MAIN STEAM LINE ISOLATION	SGS PL V040A	12406	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V040A-LC	12406	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V040A-LO	12406	PAMS	2 WKS
MAIN STEAM LINE ISOLATION	SGS PL V040B	12404	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V040B-LC	12404	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V040B-LO	12404	PAMS	2 WKS
MAIN FEEDWATER ISOLATION	SGS PL V057A	12406	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V057A-LC	12406	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V057A-LO	12406	PAMS	2 WKS
MAIN FEEDWATER ISOLATION	SGS PL V057B	12404	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V057B-LC	12404	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V057B-LO	12404	PAMS	2 WKS
STARTUP FEEDWATER ISOL	SGS PL V067A	12406	ESF	5 MIN
LIMIT SWITCH	SGS PL V067A-L	12406	PAMS	2 WKS
MOTOR OPERATOR	SGS PL V067A-M	12406	ESF	5 MIN
STARTUP FEEDWATER ISOL	SGS PL V067B	12404	ESF	5 MIN
LIMIT SWITCH	SGS PL V067B-L	12404	PAMS	2 WKS
MOTOR OPERATOR	SGS PL V067B-M	12404	ESF	5 MIN
SG BLOWDOWN ISOLATION	SGS PL V074A	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V074A-LC	12306	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V074A-LO	12306	PAMS	2 WKS
SOLENOID VALVE	SGS PL V074A-S	12306	ESF	5 MIN
SG BLOWDOWN ISOLATION	SGS PL V074B	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V074B-LC	12306	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V074B-LO	12306	PAMS	2 WKS
SOLENOID VALVE	SGS PL V074B-S	12306	ESF	5 MIN
SG SERIES BLOWDOWN ISOL	SGS PL V075A	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V075A-LC	12306	PAMS	2 WKS

Table 3.11-1 (Sheet 27 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
LIMIT SWITCH (OPEN)	SGS PL V075A-LO	12306	PAMS	2 WKS
SOLENOID VALVE	SGS PL V075A-S	12306	ESF	5 MIN
SG SERIES BLOWDOWN ISOL	SGS PL V075B	12306	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V075B-LC	12306	ESF	5 MIN
LIMIT SWITCH (OPEN)	SGS PL V075B-LO	12306	PAMS	2 WKS
SOLENOID VALVE	SGS PL V075B-S	12306	ESF	5 MIN
STEAM LINE COND DRAIN ISOL	SGS PL V086A	12406	ESF	5 MIN
SOLENOID VALVE	SGS PL V086A-S	12406	ESF	5 MIN
STEAM LINE COND DRAIN ISOL	SGS PL V086B	12404	ESF	5 MIN
SOLENOID VALVE	SGS PL V086B-S	12404	ESF	5 MIN
PWR OPERATED RELIEF VALVE	SGS PL V233A	12406	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V233A-LC	12406	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V233A-LO	12406	PAMS	2 WKS
PWR OPERATED RELIEF VALVE	SGS PL V233B	12404	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V233B-LC	12404	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V233B-LO	12404	PAMS	2 WKS
MSIV BYPASS ISOLATION VALVE	SGS PL V240A	12406	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V240A-LC	12406	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V240A-LO	12406	PAMS	2 WKS
MSIV BYPASS ISOLATION VALVE	SGS PL V240B	12404	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V240B-LC	12404	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V240B-LO	12404	PAMS	2 WKS
MAIN FEEDWATER CONT VLV	SGS PL V250A	20400	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V250A-LC	20400	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V250A-LO	20400	PAMS	2 WKS
MAIN FEEDWATER CONT VLV	SGS PL V250B	20400	ESF	5 MIN
LIMIT SWITCH (CLOSED)	SGS PL V250B-LC	20400	PAMS	2 WKS
LIMIT SWITCH (OPEN)	SGS PL V250B-LO	20400	PAMS	2 WKS
STARTUP FEEDWTR CONT VLV	SGS PL V255A	12406	ESF	5 MIN
STARTUP FEEDWTR CONT VLV	SGS PL V255B	12404	ESF	5 MIN
PRESSURE REG VALVE A	VES PL V002A	12401	ESF	5 MIN
PRESSURE REG VALVE B	VES PL V002B	12401	ESF	5 MIN
ACTUATION VALVE A	VES PL V005A	12401	ESF	2 WKS
ACTUATION VALVE B	VES PL V005B	12401	ESF	2 WKS
OUTLET ISOLATION VALVE	WGS PL V051	12155	ESF	5 MIN
RCDT CONTAINMENT ISOL IRC	WLS PL V004	11300	ESF	5 MIN
SOLENOID VALVE	WLS PL V004-S	11300	ESF	5 MIN
RCDT CONTAINMENT ISOL ORC	WLS PL V006	12256	ESF	5 MIN
SUMP CONTAINMENT ISOL IRC	WLS PL V055	11300	ESF	5 MIN
SOLENOID VALVE	WLS PL V055-S	11300	ESF	5 MIN
SUMP CONTAINMENT ISOL ORC	WLS PL V057	12256	ESF	5 MIN

Table 3.11-1 (Sheet 28 of 28)

Safety-Related Electrical and Mechanical Equipment

Description	AP600 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)
SOLENOID VALVE	WLS PL V057-S	12256	ESF	5 MIN
RCDT GAS CONTAINMENT ISOL	WLS PL V067	11300	ESF	5 MIN
SOLENOID VALVE	WLS PL '/067-S	11300	ESF	5 MIN
RCDT GAS CONTAINMENT ISOL	WLS PL V068	12256	ESF	5 MIN
SOLENOID VALVE	WLS PL V068-S	12256	ESF	5 MIN
MISCELLANEOUS:				
NON-ACTIVE VALVES		SEE TABLE 3.2-3		
HEAT EXCHANGERS		SEE TABLE 3.2-3		
TANKS		SEE TABLE 3.2-3		
HYDROGEN RECOMBINER A	VLS MY E01A	11300		1 YR
HYDROGEN RECOMBINER B	VLS MY E01B	11300		1 YR
REMOTE SHUTDOWN WORKSTATION		12303		NOTE 3

Note 1:RT (Reactor trip), ESF (Engineered Safeguards Feature), PAMS (Post Accident Monitoring), ISOL (Isolation)
Note 2:Zones identified by room numbers -- see Section 1.2, Non-proprietary General Arrangement drawings
Note 3:Not required post-accident
Note 4:Only 3 of 16 used for PAMS
Note 5:Reference Table 3D.4-2

Question 410.253

How is the main steam supply system designed to protect against water entrainment in accordance with Position 2 of Section 10.3 of the SRP?

Response:

Water entrainment due to condensation is addressed in the layout of the steamline piping. The layout for the steamline piping inside the auxiliary building provides draining by proper sloping of the lines to enhance condensate collection, and in the use of condensate drains. A 12" condensate pot is provided upstream of the main steam isolation valve to collect condensate in the main steamline.

SSAR Revision:

The fourth paragraph of Subsection 10.3.2.2.1 will be revised as follows:

The main steam lines between the steam generator and the containment penetration are designed to meet the leak before break criteria described in Subsection 3.6.3. The portion of the system between the containment penetration and the anchor downstream of the MSIV is part of the break exclusion zone. For each of these cases piping failures need not be postulated.

The layout of the steam piping provides for the collection and drainage of condensate to avoid water entrainment, by the proper sloping of lines and the use of condensate drain pots.

PRA Revision: NONE

Question 440.61

Provide a description of the design features that minimize shutdown risk. The description should include non-safety-related systems for normal shutdown operations, passive safety-related systems, and the functional capability and availability of the active systems to ensure defense-in-depth, accident mitigation, and core damage prevention capability. The design bases, functions and supporting analyses for each system used to minimize shutdown risk should be discussed.

Response:

Systems that minimize shutdown risk are discussed in Reference 440.61-1. The AP600 PRA shows the effectiveness of these systems in minimizing the risk during shutdown conditions. Descriptions of these systems are provided in the appropriate SSAR sections. The SSAR also includes the design basis for each of these systems. The response to RAIs 100.11, 440.53 and 440.62 provides additional discussion about features of these systems that minimize shutdown risk.

The response to RAIs 440.63 and 440.66 address the analysis of the plant during shutdown conditions.

The response to RAI 440.56 discusses instrumentation available during shutdown conditions.

References:

440.61-1 WCAP-13793, AP600 Systems / Events Matrix, June 1994

SSAR Revision: NONE

PRA Revision: NONE

Question 440.205

In sequence 24 of the LOOP event tree, PF-203, Figure F-9 of the PRA, the offsite power is not recovered in 24 hours and the PRHRS is initially operating. What is the effect of ADS actuation at 24 hours after LOOP? What is the operator expected to do in this regard? Can RNS be used to remove decay heat? Will a transportable generator be available after 24 hours?

Response:

It is very unlikely that loss of both offsite and onsite power will last 24 hours. Should the loss of offsite and onsite power continue for 24 hours, the normal residual heat removal system will not be available due to the unavailability of the ac power. If onsite power becomes available prior to 24 hours, the startup feedwater pumps would be automatically actuated to remove decay heat. The operator can take actions to use the normal residual heat removal system to remove decay heat after the reactor coolant system is cooled down and depressurized to its cut-in conditions.

If ac power is lost for an extended time, it is likely that a generator can be made available within 24 hours to sustain decay heat removal with the passive residual heat removal heat exchanger. If the generator cannot be made available, then the automatic depressurization system is automatically actuated to put the plant in a long term core cooling mode before the 24 hour batteries are exhausted. This automatic depressurization system actuation places the plant in a safe long term shutdown condition using injection from the accumulators, core makeup tanks, IRWST and ultimately from containment recirculation. A generator is not required to support this shutdown cooling mode.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.214

Note 16 in the P&ID PXS M6001 and note 22 of PXS M6002 indicate that freeze seal would be used for direct vessel injection lines, accumulator injection lines, and gravity injection lines. Why is this necessary? What is the plant condition under which a freeze seal is used? How do you account for this in the PRA?

Response:

Freeze seals can be used to temporarily isolate those portions of the lines in the passive core cooling system that connect directly to the primary pressure boundary. They are used to allow maintenance on valves or instruments in lines that interface directly with the reactor coolant system.

The plant conditions under which the freeze seals could be used are as follows:

- Mode 5 (cold shutdown with the reactor coolant system open)

- Mode 6 (refueling)

The PRA models accident scenarios in Mode 5. The PRA models maintenance of equipment; the impact is the same regardless of whether manual valves or freeze seals are used to provide the isolation. Therefore, use of freeze seals has no impact on the PRA.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.218

In a transient with loss of steam generator heat removal, the PRHRS is used to remove decay heat. In such transients, what are the projected RCS pressure, temperature and level? What does the operator need to do to ensure its operation? What instrumentations are available to the operator? During a transient, some RCS inventory may be lost through the pressurizer safety valves. In an overcooling transient, the RCS level could shrink due to thermal contraction. The staff is concerned that there may not be sufficient inventory to support PRHR operation. What is the minimum RCS inventory that can support PRHR operation? What is the elevation of the top of the PRHRS HX? How does it compare with the level in the CMT and the top of the vessel? Can the RCS inventory become lower than the minimum needed in some of the transients? What is the pressurizer level at which the PRHRS HX would start draining?

Response:

Following a transient with loss of steam generator heat removal capability, the passive residual heat removal heat exchanger is automatically actuated to provide the safety-related decay heat removal function. Due to the passive residual heat removal system operation, the reactor coolant system parameters vary in relation to the initiating event, the number of available heat exchangers, the duration of the passive residual heat removal system operation, and the availability of nonsafety-related systems during the transient. SSAR Section 15.2 provides analyses that demonstrate the performance of the safety-related systems. The projected temperature and corresponding pressure is consistent with the passive residual heat removal system functional requirements described in the response to RAI 440.92. The projected reactor coolant system level is consistent with the design criteria requiring that the passive residual heat removal heat exchanger operation, during postulated non-LOCA events with loss of steam generator decay heat removal capability, shall sufficiently cool the reactor coolant system.

The operator has the capability to monitor the passive residual heat removal heat exchanger operation by means of several instruments. SSAR Table 7.5-1 provides the list of variables available to the operator. Passive residual heat removal operation and the reactor coolant system makeup provided by the core makeup tanks are automatically actuated and require no operator action.

Regarding concerns about the reactor coolant system inventory during the passive residual heat removal heat exchanger operation, it should be noted that two separate kind of events can be postulated:

- Reactor coolant system shrinkage due to reactor coolant system cooldown

- The opening of pressurizer safety valves with loss of primary coolant for a reactor coolant system heatup.

Section 4.1 of Reference 440.218-1 provides the analysis of the steam line break (a reactor coolant system cooldown event). The results of this analysis indicate that the core makeup tanks provide sufficient reactor makeup to support passive residual heat removal system operation with reactor coolant system temperature as low as 212°F without credit for accumulator injection and without automatic depressurization system actuation.

Section 15.2.7 of the SSAR provides the analysis of the loss of main feedwater event (a reactor coolant system heatup event). In this event the pressurizer safety valves open such that some reactor coolant inventory is lost. The results of this analysis show that there is sufficient reactor coolant system inventory to maintain pressurizer level which is more than enough to support passive residual heat removal system operation.

The passive residual heat removal system will remain water filled until the reactor coolant system level drops to the hot leg. There is no minimum reactor coolant system inventory required to support the passive residual heat removal system operation since the passive residual heat removal heat exchanger will operate with water/steam or just steam inputs. However, for non-LOCA events the reactor coolant system conditions are such that the passive residual heat removal system will operate with water flow from the hot leg.

The elevations requested are as follows:

- Top PRHR heat exchanger (top tubes)	130.67 ft
- Top core makeup tank (CMT)	127.63 ft
- CMT water elevation to actuate ADS (67% of CMT volume)	120.16 ft
- Top reactor vessel	113.78 ft
- Pressurizer bottom	120.60 ft

Refer to the responses to RAIs 440.103 and 440.126 respectively, for information relevant to IRWST inventory during the passive residual heat removal system operation and the effect of the pressurizer heaters on the passive residual heat removal system operation.

References:

440.218-1 AP600 Design Change Description Report, February 15, 1994.

SSAR Revision: NONE

Question 440.220

How is the PRHRS HX tube integrity verified? Do you need to perform eddy current tests similar to that for steam generators? How and during what plant conditions are these tests done? How do such tests affect the availability of the IRWST?

Response:

The design of the passive residual heat removal heat exchanger tube to tube sheet joint is the same as in the steam generator. The inspection of the passive residual heat removal heat exchanger will be performed using the same techniques. These inspections will be performed when the passive residual heat removal heat exchanger is not required to be available by the technical specifications and the reactor coolant system is depressurized during modes 5 and 6.

Access to the tubes for inspection is through the channel head manway. Since the channel heads are located outside the IRWST, the IRWST is unaffected by this inspection.

SSAR Revision: NONE

Question 440.232

Section A.3 of the PRA defines a small-break LOCA as a break from 3/4 to 4 inches, while Section 15.6.5.4B.3.1 of SSAR defines a small-break LOCA as a break size of one square foot or less. Provide the following information:

a. Clarify the inconsistency in the definition of LOCA sizes.

b. What is the size or the LOCA that a CVS pump is capable of mitigating? How long can this pump perform its mitigating functions? Given such a LOCA, what is the expected plant and operator response to bring the plant to a safe condition?

c. For how long will the CVS be able to inject borated water during an RCS leak or a very small-break LOCA? How much inventory is available? What measures assure that this inventory will be available?

Response:

a. In the AP600 PRA, LOCAs are classified according to the break size, the location of the break, and the effect on mitigating systems. The break size was defined on the basis of plant systems required to mitigate the accident. The break sizes that separate the LOCA categories are 10-inch diameter between large/medium, 4-inch diameter between medium/small, 3/4-inch diameter between small/very small, and 3/8-inch diameter between very small RCS leak. More details of the LOCA classification is provided in Chapter 7 and Appendix A of the AP600 PRA Report. The RCS leak was addressed in reference 440.232-1 and will be incorporated in Revision 2 of the PRA.

The design basis LOCA classification in the SSAR Sections 15.6.5.4A and 4B is based on ANSI 18-2 which divides plant conditions into different categories according to anticipated frequency of occurrence and potential radiological consequences to the public. Based on reference 440.232-2, the RCS design basis breaks were categorized in the SSAR as follows:

Large break is defined as a rupture of the reactor coolant system with a total cross-section area equal to or greater than one square foot. This event is considered as a Condition IV event (limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a design basis accident for design engineered safeguards.

Small break is defined as a rupture of the reactor coolant system with a total cross-section area less than one square foot in which normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered as a Condition III event because it is an infrequent fault that may occur during the life of plant.

b. One chemical and volume control system makeup pump is capable of supplying sufficient makeup flow to compensate for the leakage from breaks up to 3/8-inch diameter (RCS leak). The chemical and volume control system can maintain the RCS pressure and presssurizer level and permit the operator to perform a

controlled shutdown using the startup feedwater pumps. The normal residual heat removal system is later used to continue the cooldown so the RCS can be completely depressurized and thus reduce the leakage through the break. The PRA evaluation of RCS leak is described in reference 440.232-1.

For breaks sizes from 3/8 to 3/4-inches in diameter (very small LOCA), the chemical and volume control system, in conjunction with the core makeup tanks and the passive residual heat removal system, can provide sufficient RCS makeup for leakage and accommodate RCS cooldown. The normal residual heat removal system is later used to continue the cooldown so that the RCS can be completely depressurized and thus reduce leakage through the break. For small LOCAs, up to two inches, one CVS pump can provide sufficient makeup to prevent core uncovery. The PRA does not credit CVS operation for mitigating break sizes greater than 3/4-inch diameter because of consideration of limited CVS makeup water supplies. The event tree model for the very small LOCA event is provided as Figure F-18 in Appendix F, Section F.2.19 of the AP600 PRA Report.

The CVS has sufficient inventory to provide RCS makeup for breaks up to 3/4-inch for 5 to 6 hours without operator action. As the operator proceeds to promptly reduce the RCS pressure for shutdown, the amount of inventory being lost through the break will diminish. This reduces the demand for makeup and the CVS pumps will deliver makeup at a reduced rate. The PRA does not model operator actions that would reduce the RCS pressure and allow CVS operation for 24 hours. This will be evaluated in Revision 2 of the PRA.

The expected plant and operator responses following RCS leaks and very small LOCAs are described in reference 440.232-3.

c. The CVS makeup pumps draw suction from the boric acid tank (BAT). (See SSAR Table 9.3.6-2 for the BAT capacity). The inventory in the BAT will be administratively controlled. In addition to the BAT, the CVS has the capability to take suction from the spent fuel pit. The CVS connection to the spent fuel pit is located approximately two feet below the normal water level and two feet above the spent fuel pit pump suction connection. This arrangement precludes a loss of cooling or inadvertent draining of the spent fuel pit due to CVS operation. The inventory in the spent fuel pit will be administratively controlled. Conservatively assuming a reduced inventory in the BAT, the BAT and the spent fuel pit maintain sufficient inventory to provide suction to one CVS makeup pump to supply RCS makeup for 5 to 6 hours at 100 gpm. This does not take credit for operator actions to replenish the inventory in the BAT. In addition, the AP600 RTNSS evaluation (reference 440.232-4) indicates that the CVS is not RTNSS significant.

References:

440.232-1 ET-NRC-93-3990, "AP600 Reactor Coolant System Leak PRA Evaluation," October 1993.

440-232-2 WCAP-8340, "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies," July 1974.

440.232-3 WCAP-13793, "AP600 System / Event Matrix," June 1994.

440-232-4 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," September 1993.

SSAR Revision: NONE

PRA Revision: NONE

Question 460.5

Provide the following informatio· ·egarding solid radwastes (Section 11.4):

a. Estimates of solid waste volumes expected to be shipped annually for wet solid wastes and dry solid wastes separately.

b. A discussion of compliance with Position III.1 of BTP ETSB 11-3 regarding the storage capacity for accumulated filter sludges.

c. A discussion of compliance with Position III.2 of BTP ETSB 11-3 regarding storage volume for solidified wastes (both wet and dry solid wastes) available in the plant.

Response: (Revision 1)

a. Estimates of solid waste volumes expected to be generated and shipped annually are separately provided below in Tables 1.A. and 2.A for wet solid wastes and dry solid wastes. These volumes are consistent with SSAR Table 11.4-4. In Table 11.4-4 the volume of wastes to be shipped accounts for volume reduction of some wastes and addition of the shipping/disposal container. The processing and packaging factors used in SSAR Table 11.4-4 are conservative and may have over-estimated the annual disposal quantities. Below each estimate of annual disposal quantities are the details of the inputs to each category of generation (Tables 1.B and 2.B). These waste generation estimates are based on an 18 month refueling cycle. For a 24 month refueling cycle, the annual waste generation is less.

Table 1.A - Annual Wet Solid Waste Generation and Disposal Quantities Summary

Wet Solid Wastes	Waste Generation Volume, ft^3/yr	Expected ~~Shipping or~~ Disposal Volume, ft^3/yr
Spent Charcoal and Ion Exchange Resins	250	314.8
Mixed Liquid Wastes	15	17.0
Chemical Wastes	350	19.8
Total Wet Solid Wastes	615	351.6

The generation of these wet solid wastes is summarized as follows:

Table 1.B - Annual Wet Solid Waste Generation Quantities Details

Waste Source	Waste Generation Rate	Annualized Waste Generation	Comments
WLS Deep Bed Filter Charcoal	10 ft^3 / 6 months	20 ft^3	
WLS Deep Bed Filter Zeolite	40 ft^3 / year	40 ft^3	
3 x WLS Ion Exchanger Resin	30 ft^3 ea. / year	90 ft^3	
2 x CVS Ion Mixed Bed Exchanger Resin	50 ft^3 / 18 months	33.3 ft^3	one vessel discharged each refueling
1 x CVS Cation Bed Exchanger Resin	50 ft^3 / 36 months	16.7 ft^3	
2 x SFS Ion Exchanger Resin	75 ft^3 / 18 months	50 ft^3	one vessel discharged each refueling
Subtotal Wet Solid Waste (Spent Resins and Charcoal)		250 ft^3	
Mixed Liquid Wastes	4 gal./month for 17 months + 100 gallons during refueling outage	15 ft^3	mainly contaminated lubricating oil
Chemical Wastes	200 gal./month for 17 months + 500 gallons during refueling outage	350 ft^3	Chemical laboratory and decontamination wastes
Total Wet Solid Wastes		615 ft^3	

Table 2.A - Annual Dry Solid Waste Generation and Disposal Quantities Summary

Dry Solid Wastes	Waste Generation Volume, ft^3/yr	Expected Disposal Volume, ft^3/yr
Spent Filter Cartridges	3.6	24.5 ~~4.7~~
WGS Charcoal	8.7	9.8
Compactible DAW	4480	953.0 ~~991.3~~
Non-Compactible DAW	225	363.4
Mixed Solid Wastes	5	7.5
Total Dry Solid Wastes	4722	1358.2 ~~1376.7~~

The generation of these dry solid wastes is summarized as follows:

Table 2.B - Annual Dry Solid Waste Generation Quantities Details

Waste Source	Waste Generation Rate	Annualized Waste Generation	Comments
WSS Resin Fines Filter	1 x 0.5 ft^3 cartridge / year	0.5 ft^3	6" dia. x 30" pleated fiberglas, 25 microns @ 99% efficiency
2 x WLS Filter	4 x 0.3 ft^3 cartridges / 18 months	0.8 ft^3	6" dia. x 18"
CVS Makeup Filter	1 x 0.3 ft^3 cartridge / 18 months	0.2 ft^3	6" dia. x 18"
CVS Reactor Coolant Filter	4 x 0.5 ft^3 cartridges / 18 months	1.3 ft^3	6" dia. x 30"
SFS Filters	4 x 0.3 ft^3 cartridges / 18 months	0.8 ft^3	6" dia. x 18"
WGS Guard Bed Charcoal	8 ft^3 / 36 months	2.7 ft^3	assumes abnormal charcoal replacement
WGS Delay Bed Charcoal	60 ft3 / 10 years	6 ft^3	assumes abnormal charcoal replacement
High Activity Dry Compactible Waste	2 ft^3/month for 17 months + 40 ft^3 during refueling outage	50 ft^3	

Low Activity Dry Compactible Waste	60 ft^3/month for 17 months + 5000 ft^3 during refueling outage	4000 ft^3	
Compactible HVAC Filters	See Note 1	430 ft^3	
High Activity Dry Non-Compactible Waste	1 ft^3/month for 17 months + 20 ft^3 during refueling outage	25 ft^3	
Low Activity Dry Non-Compactible Waste	6 ft^3/month for 17 months + 200 ft^3 during refueling outage	200 ft^3	
Mixed Solid Wastes	0.2 ft^3/month for 17 months + 4 ft^3 during refueling outage	5 ft^3	
Total Dry Solid Wastes		4722 ft^3	

Note 1: Waste HVAC filters are generated as follows -

Containment purge system (VFS) 13.7 ft^3
36.7 ft^3 granular charcoal / 60 months
4 x 12" x 24" x 24" HEPA / 60 month
2 x 12" x 24" x 24" Pre-filter / 60 months
2 x 12" x 24" x 24" Post-filter / 60 months

Rad. Chem. Lab. Supply & Recirculation (VAS) 42.7 ft^3
4 x 12" x 24" x 24" HEPA / 18 month
4 x 12" x 24" x 24" Pre-filter / 6 month

Rad. Chem. Lab. Exhaust (VAS) 21.3 ft^3
2 x 12" x 24" x 24" HEPA / 18 month
2 x 12" x 24" x 24" Pre-filter / 6 month

Radwaste Building (VRS) 240 ft^3
60 x 12" x 24" x 24" HEPA / yr

Annex Building (VHS) 112 ft^3
28 x 12" x 24" x 24" HEPA / yr

429.7 ft^3

Under normal conditions, there are no wastes generated from the secondary-plant cycle. If radioactivity is detected in the steam generator blowdown and reaches a predetermined level (due to primary and secondary leakage), the blowdown is diverted to the ~~condensate polishers~~ to the liquid radwaste system for processing as described in SSAR Subsections 10.4.8 and 11.2.2.1.5. Thus, blowdown demineralizer ~~condensate polishing~~ resins are not a normal source of radwaste requiring shipping and disposal. Should plant operation continue with leakage from the primary to the secondary side utilizing the blowdown demineralizers ~~condensate polishers~~, a shipping (disposal) volume of up to 300 ft^3/month could be produced by the steam generator blowdown ~~condensate polishing~~ system, as indicated in SSAR Subsection 11.4.2.1.

b. The AP600 ~~plant~~ does not include filters that generate sludge-type wastes. Also, tanks and sumps are designed to minimize the formation of sludge deposits, and the particulate matter that can cause sludge deposits is transported to and removed by the cartridge filters in the liquid radwaste system. Therefore, there are no storage provisions for accumulating sludges.

c. Branch Technical Position ETSB 11-3 specifies in Position III.2. that storage areas for solidified wastes should be capable of accommodating at least 30 days of waste generation at normal generation rates and that these storage areas should be indoors. The storage durations of the storage areas ~~in the Radwaste Building~~ are evaluated for four general types of waste and container categories discussed below.

<u>Spent Ion Exchange Resins and Filter Charcoal in High-Integrity Containers (HICs)</u>

Although the shipping or disposal volume is nearly independent of container size (based on equal filling efficiency, e.g., 90%), the storage duration for the filled HICs is dependent on the number of containers which is indirectly proportional to container size. To be conservative, it is therefore assumed that the spent resins and filter bed charcoal will be dewatered in HICs that will fit into a Type B shipping cask (i.e., the SEG 3-82B, formally HN-200). Normally 250 ft^3/yr of spent resin and charcoal is expected to be generated (SSAR Table 11.4-4), with an activity of 950 curies (SSAR Table 11.4-6). This resin can be mixed to produce a uniform specific activity of 3.8 Ci/ft^3. A 158 ~~70~~-ft^3 HIC filled to 90% would contain about 540 ~~240~~ Curies, well within the cask's capability. About two ~~four~~ of these HICs are required per year. The three onsite storage casks can each hold one of these HICs, and the resulting storage duration is about one and a half years ~~9 months~~. The ~~two~~ spent resin container fill stations may also be used for storage until it is necessary to begin filling another HIC. The ~~two~~ fill stations and two of the onsite storage casks (reserving one onsite storage cask for high-activity filter drums) provide about one and a half years ~~12 months~~ of storage. These storage times are in addition to the pre-packaging storage times provided by the spent resin tanks as described in SSAR Subsection 11.4.2.2.1.

<u>High Activity Filter Cartridges in Drums</u>

As indicated in SSAR Tabie 11.4-4, packaging of the CVS reactor coolant filter cartridges is expected to normally generate 3 ~~0.35~~ drums of waste per year. This is based on a generation rate of 4 ~~2 ft³ of~~ filter cartridges every 18 months. ~~Based on a drum volumetric loading of 50 percent, about one half of a drum is filled every 18 months (2 ft³/7.55 ft³ x 0.5), and about one drum is produced every 36 months.~~ The high-

high-activity filter storage tube module may be used to store all filter cartridges normally generated every 18 months. Thus, after a drum is filled with high-activity filters, encapsulated, and sealed, it may remain in the processing cask ~~for about 17 months before~~ **until** it is necessary to begin to [illegible]age the filter cartridges stored in the storage tube module to clear space for the next batch of spent filters. Therefore, a storage duration of about 17 months is available for high-activity filters using only the processing cask. One of the onsite storage casks could also be used for high activity filter drum storage if necessary.

<u>Other Wastes in Drums</u>

Based on SSAR Table 11.4-4, about 11 drums are produced each year containing wastes other than high activity filter and mixed wastes. The Radwaste Building (Proprietary Figure 1.2-29) has a packaged waste storage room that may be used to store both drums and boxes. Using two storage locations for palletized drums stacked three high, 24 drums can be stored. This provides about 28 months of storage for the normal expected generation rate. Stacking only two pallets high provides about 18 months of storage. Without stacking about 9 months of storage is available for the normal generation rate.

Mixed wastes are accumulated in drums and are sent to **an off site processing facility** ~~special prefabricated storage building~~ at an expected rate of about three drums per year.

<u>Wastes in Boxes</u>

Based on SSAR Table 11.4-4, about **12** ~~12.7~~ boxes are generated per year. **Ten** ~~Fourteen~~ box storage locations are available in the packaged waste storage room. Without stacking and with stacking two and three high, about 1, 2, and 3 years of storage are provided, respectively.

Maximum truck loading is expected to be 28 boxes. ~~This is equivalent to the 14~~ **Ten** storage locations **can accumulate a truck load when** stacked **three** ~~two~~ high. At the normally expected generation rate, it takes **2 years** ~~26 months~~ to produce a truck load.

In summary, indoor storage is provided for all categories of packaged wastes well in excess of 30 days, based on normally expected waste generation rates.

SSAR Revision: **Table 11.4-4 is to be revised as follows:**

Table 11.4-4 (Sheet 1 of 3)

Expected Annual Radwaste Volumes

Source	Form	Influent Vol. (ft^3)	Vol. Reduction	Container (1)			
				Type	% Full	Quantity	Disposal Vol. (ft^3)
WLS Deep Bed Filter	Spent Activated Charcoal	20	-	HIC (2)	90	0.14	25.2
WLS Deep Bed Filter	Spent Zeolytic Resin	40	-	HIC	90	0.28	50.4
WLS Ion Exchanger #1	Spent Borated Cation Bed H+Form Resin	30	-	HIC	90	0.21	37.8
WLS Ion Exchanger #2	Spent Borated Mixed Bed H+OH Form Resin	30	-	HIC	90	0.21	37.8
WLS Ion Exchanger #3	Spent Borated Mixed Bed H+OH Form Resin	30	-	HIC	90	0.21	37.8
CVS Mixed Bed Ion Exchangers	Spent Borated Mixed Bed Li^7OH Form Resin	33.3	-	HIC	90	0.23	41.9
CVS Cation Bed Ion Exchanger	Spent Borated Cation Bed H+Form Resin	16.7	-	HIC	90	0.12	21.0
SFS Ion Exchangers	Spent Borated Mixed Bed H+/OH Resin	50	-	HIC	90	0.35	62.9
WSS Resin Fines Filter	Spent Filter Cartridge	0.5	4:1	Drum	90	0.02	0.1
WLS Filters	Spent Filter Cartridge	0.8	4:1	Drum	90	0.03	0.2
CVS Makeup Filter	Spent Filter Cartridge	0.2	4:1	Drum	90	0.01	0.07
CVS Reactor Coolant Filters	Spent Filter Cartridge	1.3	-	Drum	Note 4 ~~50~~	3 ~~0.35~~	22.5 ~~2.7~~
SFS Filters	Spent Filter Cartridge	0.8	-	Drum	50	0.21	1.6
WGS Guard Bed	Activated Charcoal	2.7	-	Drum	90	0.4	3.0
WGS Delay Bed	Activated Charcoal	6	-	Drum	90	0.9	6.8
RCA SCAs (2)	High Activity Compactible DAW (2)	50	4:1	Drum	90	1.9	14.2

Table 11.4-4 (Sheet 2 of 3)

Expected Annual Radwaste Volumes

Source	Form	Influent Vol. (ft^3)	Vol. Reduction	Container (1)			
				Type	% Full	Quantity	Disposal Vol.(ft^3)
RCA SCAs	Other Compactible DAW	4000	6:1	Box	90	8.2	847.7
HVAC Exhaust Filters	Pre, HEPA, and Charcoal Filters	430 ~~611~~	6:1	Box	90	0.88 ~~1.26~~	91.1 ~~129.4~~
RCA SCAs	Higher Activity Non-Compactible DAW	25	-	Drum	70	4.9	36.4
RCA SCAs	Lower Activity Non-Compactible DAW	200	-	Box	70	3.2	327.0
Lubricants and Cleaning Agents	Mixed Wastes - Solids	5	-	Drum	70	1.0	7.5
	Mixed Wastes - Liquids	15	-	Drum	90	2.3	17.0
Radiochemistry Laboratory	Chemical Wastes	350	(3)	Drum	90	2.6	19.8
Totals		5337 ~~5518~~				1.8 HICs 17.3 ~~14.6~~ Drums 12.3 ~~12.7~~ Boxes	1710 ~~1729~~

(1) Container Parameters

Drum Internal Volume = 7.35 ft^3
Drum Disposal Volume = 7.5 ft^3

HIC Internal Volume = 158 ft^3
HIC Disposal Volume = 179 ft^3

Box Internal Volume = 90 ft^3
Box Disposal Volume = 103 ft^3

(2) Acrynoms

CVS = Chemical and Volume Control System
DAW = Dry Active Waste
HEPA = High Efficiency Particulate Air
HIC = High Integrity Container
HVAC = Heating, Ventilating and Air Conditioning

Table 11.4-4 (Sheet 3 of 3)

Expected Annual Radwaste Volumes

RCA	=	Radiologically Controlled Area
SCA	=	Surface Contamination Area
SFS	=	Spent Fuel Cooling System
WGS	=	Gaseous Waste Management System
WLS	=	Liquid Waste Management System
WSS	=	Solid Waste Management System

(3) Mobile System Concentration = 25:1 and 80% Waste Loading Using Aquaset/Petroset.

(4) One filter cartridge encapsulated in each drum

Question 460.7

Table 2 of Section 11.5 of the SRP, "Process and Effluent Monitoring Instrumentation and Sampling Systems," includes a service water system effluent monitor. The staff notes that AP600 design includes an upstream provision for this monitor in the form of component cooling water system monitor. The staff does not consider an upstream provision as an adequate basis for eliminating a downstream provision for this monitor. Therefore, include a service water system monitor or justify its elimination (Sections 11.5).

Response: (Revision 1)

The Service Water System water inventory is integrated with the Circulating Water System and the cooling tower basin. The Service Water System does not have a dedicated effluent discharge pathway but shares a common cooling tower blowdown system connection with the circulating water system as shown in Figure 10.4.5-1. As shown in Table 11.5.1, the normally nonradioactive systems which provide potential leak pathways into the service water system include radiation monitors with nominal minimum detectable concentrations of 1.0E-8 uCi/cc. Further dilution in the service water or circulating water systems or in the cooling water blowdown system, as described in Section 11.2.3.3, would reduce the concentration to approximately 3E-11 which is below the minimum detectable concentration for a continuous-type monitor. Therefore, a radiation monitor in the cooling tower blowdown (service water effluent) will not provide additional information or control of effluent releases.

As discussed in Section 11.5.3, the primary means of quantitatively evaluating the isotopic activities in effluent paths is a program of sampling and laboratory measurements. Table 9.3.4-2, which indicates grab sample locations, identifies a grab sample for the cooling tower blowdown as well as the cooling tower basin and service water basin. The grab sample allows for measuring of the activities in the effluent path.

SSAR Revision: NONE

NRC Disposition

Clarify which are the normally non-radioactive leakages that feed into the SWS and which are identified in Table 11.5-1 of the SSAR. Provide this information in the SSAR.

Disposition Response:

The service water system (SWS) configuration is described in SSAR Section 9.2.1, Revision 1.

The only potential radioactive inleakage path to the SWS is from the normally non-radioactive component cooling water system (CCS). The interface occurs at the CCS heat exchangers, which are cooled by the SWS. The CCS has a radiation monitor which is identified in SSAR Table 11.5-1. The design of the SWS includes a radiation monitor with a high alarm to monitor the SWS cooling tower blowdown flow (see also response to RAI 410.110). The radiation monitor is shown on SSAR Figure 9.2.1-1. The SSAR will be revised.

SSAR Revision:

Revise 9.2.1.2.1 to add a new paragraph to follow the third paragraph and read as follows:

A small portion of the service water system flow is normally diverted to waste water through a blowdown flow path upstream of the cooling tower. Blowdown maintains the water chemistry of the service water system.

Revise 9.2.1.2.2 under the heading of **Service Water System Valves** to add a new paragraph to follow the fourth paragraph as follows:

An air-operated isolation valve is provided in the cooling tower blowdown line. This valve allows the operator to regulate the blowdown flow. The valve also provides automatic isolation of blowdown upon loss of offsite power. The valve fails closed upon loss of control air or electrical power.

Revise 9.2.1.5 to add a new paragraph to follow the sixth paragraph and read as follows:

A radiation monitor with a high alarm is provided to monitor the cooling tower blowdown flow for radioactive leakage into the service water system from the component cooling water heat exchangers.

Revise Table 11.5-1 to add a radiation monitor listing as indicated below:

Detector	Type	Service	Isotopes	Nominal Range
SWS-JE-RE001	γ	Service Water Blowdown	Cs-137	1.0E-7 to 1.0E-2 μCi/cc

Revise 11.5.2.3.1 to add a new heading and text description, to follow at the end of the subsection, to read as follows:

Service Water Blowdown Radiation Monitor

The service water blowdown radiation monitor (detector SWS-JE-RE001) measures the concentration of radioactive materials in the blowdown flow from the service water system to the waste water system.

The service water blowdown radiation monitor initiates an alarm in the main control room if the concentration of radioactive materials exceeds a predetermined setpoint. Following the alarm, the operator can manually isolate the blowdown flow.

The range and principal isotopes are listed in Table 11.5-1. The detector is a gamma sensitive, sodium iodide, thallium activated, gain stabilized scintillator that views the liquid sample volume.

Question 460.11

Provide the following information regarding the solid radwaste management sy tem (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: (Revision 1)

a. References to resin solidification and encapsulation will be deleted from Section 11.4, except as related to space reserved for future or optional solidification facilities. ~~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 ~~auxiliary and~~ radwaste building~~s~~ that process or house radioactive solid wastes are located in areas ~~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~~

Sloped floors and floor drains are provided to collect and thereby ~~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 can be ~~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 location ~~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 auxiliary and radwaste buildings are monitored by ~~an~~ exhaust radiation monitors, as described in SSAR Subsection 11.5.2.3.2. The spent resin waste container fill station~~s and the high-activity filter processing cask~~ includes provisions for smear survey and decontamination of the external surfaces of the waste containers after filling, as described in SSAR Subsection~~s~~ 11.4.2.3.1 ~~and 11.4.2.3.2, respectively~~. ~~Boxes and drums~~ Drums and boxes containing filters and lower-activity dry wastes are also surveyed and decontaminated~~,~~ ~~as described in SSAR Subsection 11.4.2.3.3~~. ~~A~~ Mobile or portable equipment may be used for 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. Liquid wastes generated from solid radwaste system operations are discharged directly to the liquid radwaste system (WLS) or are collected by the radioactive waste drain system (WRS) and directed to the WLS for subsequent processing and monitored discharge.

c. ~~The portable grouting unit~~ Encapsulation of spent filters is by mobile or portable equipment. ~~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 t~~ 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 uses mobile or portable processing systems for wet and dry solid wastes, allowing the most efficient volume reduction methods and equipment to be used. ~~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 s~~ Solidification of ~~y~~ 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. Reference to resin solidification will be deleted from the SSAR .

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 AP600 1750 ft^3/yr ~~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 AP600 1750 ft^3/yr ~~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~~.

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

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 1.03×10^{-5} m^3/year per MWe-Hr. This is about 10 percent less than the AP600 1750 ft^3/yr ~~URD~~ goal. ~~Also Calvert Cliffs, which has had a program for solid radioactive waste reduction in place since 1985[2], shipped about 2×10^{-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.~~

~~These few examples demonstrate that w~~ With aggressive solid radioactive waste programs, current nuclear power plants are being operated within the AP600 1750 ft^3/yr ~~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~~ minimize their solid wastes ~~ation 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]

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

~~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 three ~~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 (3 4) the packaged waste storage room. ~~The performance of these storage areas has been described in the responses to Q460.5e. 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) is a ~~are two~~ cells with thick concrete walls (See SSAR Figure 1.2-29). ~~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 the ~~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 onsite storage casks are described in SSAR Section 11.4.2.2.6 and Table 11.4-12.
~~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 for stacking waste boxes or pallets of drums ~~of at least 18 feet~~. The mobile systems facility crane ~~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 area ~~room~~ periodically accessed by personnel. Mobile racks for hanging lead blankets or ~~and~~ shield panels on casters may be used ~~are available~~ for flexible response to changing conditions in the storage area ~~room~~.

Although the ALWR Utility Requirements Document (URD) is not a regulatory requirement, ~~T~~ 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 COL applicant.~~

SSAR Revision:

Revise SSAR Section 11.4.1.3 as follows:

11.4.1.3 Functional Design Basis

The solid waste management system is designed to meet the following objectives:

- Provide for the transfer and holdup of spent radioactive ion exchange resins and deep bed filtration media from the ~~various~~ ion exchangers and filters in the liquid waste processing, chemical and volume control, and spent fuel cooling systems

- Provide the means to mix, sample, and transfer spent resins and filtration media to high integrity containers or liners for dewatering or solidification by mobile processing equipment, as required

- Provide the means, by mobile processing equipment, to change out, transport, sample, hold up, and package high-activity filter cartridges from liquid systems in a manner that minimizes radiation exposure of personnel and spread of contamination

- Provide the means, by mobile processing equipment, to change out, transport, dry, sample, hold up, compact, and package lower-activity filter cartridges from liquid systems

- Provide the means to compact and package spent filters from the plant heating, ventilation, and air-conditioning systems

- Provide the means to segregate solid wastes by radioactivity level and to store and process the wastes as appropriate, including segregation and compaction, by mobile processing equipment, to achieve efficient volume reduction

- Provide the means to characterize and accumulate nonradioactive hazardous and radioactive hazardous (mixed) wastes for subsequent processing

- Provide the means to segregate clean wastes originating in the radiologically controlled area (RCA) and to verify that they are nonradioactive

- Provide the means to survey, decontaminate, and weigh packaged wastes and to store them for at least six months in the event of delay or disruption of offsite shipping
- Provide the capability, **using both permanent and mobile processing equipment**, to limit the packaged volume of primary low level waste to 1750 cubic feet per year when augmented by efficient operating techniques to minimize the generation of radioactive and nonradioactive wastes in the radiologically controlled area
- Provide the means to package radioactive solid wastes for offsite shipment and disposal according to applicable regulations, including Department of Transportation regulation, 49 CFR 173 (Reference 1) and NRC regulation, 10 CFR 71 (Reference 2).
- **Provide the means to direct liquid radwaste from solid radwaste processing to the liquid radwaste system for subsequent processing and monitored discharge**

Delete SSAR Section 11.4.2.2.15:

~~11.4.2.2.15 Portable Grouting Unit~~

~~The portable grouting unit provides for mixing grout and pumping it into filter disposal containers for waste encapsulation when required by waste form criteria.~~

Revise SSAR Section 11.4.2.3.1as follows:

11.4.2.3.1 Spent Resin Processing Operations

Demineralized water is used to transfer spent resins from the ~~various~~ ion exchangers ~~and bed filters~~. Although a bypass allows spent media to be transferred directly to a waste disposal container, the normal mode of system operation is to transfer the spent media to the spent resin tanks. Before the transfer operation, it is verified that the spent resin tank selected and aligned as a receiver has the capacity to accept the bed. It is also verified that the resin mixing pump is aligned to discharge excess transfer water through the resin fines filter to the liquid waste processing system. Either spent resin tank can be designated as a high-activity tank to receive resins exceeding a preset limit ~~(for example, more than 100 R/hr)~~.

During the transfer operation the tank level is monitored and the resin mixing pump is operated, if required, to limit tank water level. The operator stops the transfer when the CCTV camera viewing the sight flow glass indicates ~~on a control panel monitor~~ that the sluice water is clear and the transfer line is ~~therefore~~ flushed of resins.

After the bed transfer, the tank solids level can be checked by operating the resin mixing pump to lower the water level below the solids level. The solids level can be determined by the bottoming out of the ultrasonic surface detector recording.

Between bed transfer operations the water level in the spent resins tank is maintained above the solids level. Demineralized water is supplied for water level adjustment as well as a backup water source for flushing resin handling lines after resin recirculation and waste disposal container filling operations.

The solids bed can be agitated and mixed at any time by using compressed air or by operating the resin mixing pump in the resin mixing mode. In the resin mixing mode, water is drawn from the spent resin tank via resin retention screens. The water is returned via tank mixing eductors that generate a resin slurry recirculation in the tank equivalent to about four times the resin mixing pump capacity. The solids bed is locally fluidized during this operation.

Before transferring spent media to a waste disposal container, the disposal container is placed into ~~one of~~ the ~~two~~ waste container fill station using the radwaste crane. The waste container fill station~~s are~~ is a shielded enclosure~~s~~ for the spent resin disposal containers during filling operations. ~~The cells collect any gaseous or liquid leakage and allow external smear survey and decontamination of the waste containers. The hoses for waste container filling, dewatering, and venting are confined within the cells.~~

~~The dewatering standpipe from the container underdrain is connected to a flexible hose from the fill head bottom dewatering connection. The television camera and cable and the fill, dewatering, and vent hoses are attached to the fill head. The fill head is moved by its handling device over the container and sealed to the container port. The radwaste crane is then used to replace the fill station shield cover.~~

~~The resin mixing mode, as previously described, is established to locally fluidize and mix the solids bed in the spent resin tank before container filling.~~ The resin transfer pump is then started in the recirculation mode. A resin slurry is drawn from the spent resin tank and returned to the same tank. A representative resin sample may be obtained during recirculation or container filling modes ~~by operating the sampling device for a preset period of time~~.

The container fill valve for the appropriate fill station is opened to initiate the filling operation. ~~The fill valve opens only if a limit switch on the fill head indicates that the fill head is installed on a waste container. The resin dewatering pump is started to dewater the resin as it accumulates in the container. The resin dewatering pump discharges the water to the recirculation line.~~ The transfer water flows back to the spent resin tank, thereby preserving the water inventory in the system and retaining any resin fines or dislodged crud within the system.

The resin mixing pump can be stopped at any time during the filling operation. When the solids level nears the top of the container~~, as detected by level sensors and observed by a television camera,~~ the fill valve is closed and cycled to top off the container. Excessive water or solids level automatically closes the fill valve.

When the filling operation is complete, the line- flushing sequence controller is manually initiated to automatically operate the pumps and valves to flush the resin transfer lines to the waste container or back to the spent resin tank. The resin mixing pump supplies filtered flush water from the spent resin tank. ~~The dewatering pump is operated periodically over the next few days until no further dewatering flow is detected by the pump discharge pressure indicator and/or audible indications from the pump.~~

For dewatered resin shipping and disposal, the waste container is an NRC-approved, high integrity container designed and verified to dewater the solids bed to less than 0.5 percent free water. For solidified resin shipping and disposal, the waste container is a carbon steel liner. The resin is solidified ~~encapsulated by filling the container with an NRC approved binder (for example, vinyl ester styrene)~~, using vendor ~~contractor~~-supplied equipment ~~as further described in Subsection 11.4.2.4.1.~~

~~When the filling and dewatering operations are complete, the fill head is remotely disconnected from the waste container and moved aside by its handling device. The dewatering hose is cut inside the waste container by using~~

~~a reach rod tool through a port in the shield cover. The container lid is then installed through the small opening in the shield cover by using a remotely operated capper handled by a jib crane.~~

The top and sides of the container are then smear- surveyed through small ports in the shield cover. If contamination is detected, which should be an infrequent occurrence, demineralized water is sprayed on container surfaces through a set of fixed cleaning nozzles.

The cleaning water drains to the radioactive waste drain system where the water is directed ~~a nearby sump and is pumped~~ to the liquid radwaste processing system ~~on automatic level control~~.

~~When the waste container is verified to meet the surface contamination limits for shipping, the high bay area is cleared of personnel and the shield doors are closed. The transfer operation is controlled from the console in the radwaste building control room while monitoring operations on closed circuit television. The shield cover is removed by using the radwaste crane under remote control. The power grapple of the radwaste crane is then connected to the container's sling, and the container is lifted and transferred to an onsite storage cask. The remote reading dynamometer on the radwaste crane weighs the container during the transfer process. The radwaste crane is also used to replace the cover on the storage cask and to move and stack the storage casks.~~

~~The radwaste crane is also operated by remote control to transfer waste containers from the onsite storage casks to trailer mounted shipping casks. The radwaste crane handles the shipping cask cover and can also handle the shipping cask if this should prove necessary for trailer or cask maintenance.~~

Revise SSAR Section 11.4.2.4.1 as follows:

11.4.2.4.1 Portable and Mobile Radwaste Capabilities

Portable or mobile systems can be located in the mobile systems facility ~~or~~ for processing and packaging chemical wastes. Chemical wastes are ~~normally~~ processed in the radwaste building by a mobile solidification system when a batch accumulates in the chemical waste tank.

The spent resin processing system provides ~~includes~~ connections to ~~in~~ the ~~radwaste building~~ mobile systems facility in the auxiliary building train bay to allow spent resins to be delivered to a portable or mobile system for packaging by dewatering or solidification, using radwaste systems provided and operated by contractors.

Connections permit ~~convenient~~ tie-in of resin transfer and dewatering lines that may be required to an optional facility for which space is reserved adjacent to the radwaste building. ~~This optional facility could be required for certain plants where regional low level disposal facilities require solidification of spent resins. However, solidification can be performed in the radwaste building's high bay or mobile systems facility by using a portable unit to mix a qualified binding agent and to pump it into the spent resin waste container. For this process, the waste container can be inside a waste container fill station or in an onsite storage cask.~~

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 contenis, 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 (Revision 2)

Table 9.3.3-2 of SSAR Subsection 9.3.3 will be updated to include the missing ~~all~~ local sample points in the ~~primary and~~ secondary systems. ~~The revised table will be available in July 1993~~. SSAR Tables 9.3.3-1 and 9.3.3-2 include tritium as one of the radioisotopes which is analyzed in both primary and secondary systems sampling.

Sampling is performed to measure those water chemistry and radioactivity characteristics which must be monitored. The ranges and accuracy of analysis will be appropriate for the water chemistry characteristics being measured. These monitoring frequencies are selected so there is sufficient time to detect chemistry or radioactivity changes before any adverse affects occur.

SSAR Tables 11.5-1 and 11.5-2 list the radiation detectors in the AP600. Tables 460.15-1 and 460.15-2 identify other SSAR items which corresponds to the SRP Tables 1 and 2 sampling and monitoring provision items.

Table 460.15-1 - Gaseous Radiological Monitoring and Sampling

Number	Item	SSAR Reference
1	Waste Gas Holdup System	Section 11.5.2.3.2 Table 9.3.3-2, items 25, 26, 27, 28 Table 11.5-1
2	Condenser Evacuation System	Section 11.5.2.3.3 Table 9.3.3-2, item 29 Table 11.5-1
3	Vent & Stack Release Pt. System	Section 11.5.2.3.3 Table 11.5-1
4	Containment Purge System	Section 11.5.2.3.2 Table 11.5-1
5	Aux. Bldg. Ventilation System	Section 11.5.2.3.2 Table 11.5-1
6	Fuel Storage Area Vent System	Section 11.5.2.3.2 Table 11.5-1
7	Radwaste Area Vent. System	Section 11.5.2.3.2 Table 11.5-2
8	Turb. Gland Seal Cond. Vent System	Included in item 13
9	Mech. Vacuum Pump Exhaust (Hogging) System	Included in item 2
10	Evaporator Vent Systems	N/A
11	Pre-treatment Liquid Radwaste Tank Vent Gas System	Included in item 5
12	Flash Tank and Steam Generator Blowdown Vent System	Flash tank - N/A Steam generator blowdown - Section 11.5.2.3.1 Table 9.3.3-2, item 8 (See also Table 460.15-2 item 16)
13	Turbine Bldg. Vent System	System description in Section 9.4.9.
14	Pressurizer & Boron Recovery Vent Systems	Pressurizer - N/A See note 1. Boron recovery - N/A

Notes: 1. Pressurizer is vented to sampling system, reactor coolant drain tank or containment atmosphere.

Table 460.15-2 - Liquid Radiological Monitoring and Sampling

Number	Item	SSAR Reference
1	Liquid Radwaste (Batch) Effluent System	Section 11.5.2.3.3 Table 9.3.3-2, items 12,, 16, 17, 18, 19, 20, 21, 22 Table 11.5-1
2	Liquid Radwaste (Continuous) Effluent System	N/A
3	Service Water System	System description in Section 9.2.1.
4	Component Cooling Water System	Section 11.5.2.3.1 Table 9.3.3-2, items 13, 14, 15 Table 11.5-1
5	Spent Fuel Pool Treat. Syst.	Table 9.3.3-2, item 9
6	Equip. & Floor Drain Collection and Treatment Systems	Radioactive floor drain system included in item 1. Clean floor drain system - System description in Section 9.3.5. See item 20.
7	Phase Separator Decant & Holding Basin Systems	N/A
8	Chemical & Regeneration Solution Waste Systems	Rad./Chem. lab waste included in item 1. Regeneration chemical waste - N/A
9	Laboratory & Sample System Waste Systems	Included in item 1.
10	Laundry & Decontamination Waste Systems	Laundry - N/A Decontamination waste included in item 1.
11	Resin Slurry, Solidification & Baling Drain Systems	Spent resin - Table 9.3.3-2, item 23 Solidification & Baling - N/A Table 11.5-1
12	Radwaste Liquid Tanks (outside the buildings)	N/A
13	Storm & Underdrain Water Syst.	The storm drain system is site specific.

Number	Item	SSAR Reference
14	Tanks and Sumps inside Reactor Building	Reactor coolant drain tank - Table 9.3.3-2, item 11. Containment sump vents to containment atmosphere.
15	Boron Recovery System Liquid Effluent	N/A
16	Steam Generator Blowdown (Batch) Liquid Effluent System	Section 11.5.2.3.1 Table 11.5-1
17	Steam Generator Blowdown (Continuous) Liquid Effluent System	Section 11.5.2.3.1 Table 9.3.3-2, items 8
18	Secondary Coolant Treat. Waste & Turbine Bldg. Drain Systems	See item 20
19	Ultrasonic Resin Cleanup Waste Systems	N/A
20	Non-Contaminated Waste Water & PWR Turbine Building Clean Drain System	Section 11.5.2.3.3 Table 11.5-1

SSAR Revision: Table 9.3.3-2 is to be revised as shown below:

Table 9.3.3-2 (Sheet 1 of 5)

Local Sample Point Not in the Primary Sampling System
(Normal Plant Operations)

	Sample Point Name	Available Number of Points	Type of Sample*	Process Measurement
Liquid Sample				
1.	CVS Boric Acid Tank	1	Grab	pH, chlorine, fluorine, boron, silica, suspended solids, radioisotopic liquid, oxygen
2.	CVS Boric Acid Batching Tank	1	Grab	Boron, chlorine, fluorine
3.	CVS Letdown	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
4.	Residual Heat Removal Heat Exchanger	2	Grab	Radioisotopic liquid, suspended solids, radioisotopic gas, gross specific activity, strontium, iron, tritium, hydrogen, I-131, conductivity, pH, oxygen, chlorine, fluorine, boron, aluminum, silica, lithium radioisotopic liquid, lithium radioisotopic particulate, magnesium, sulfate, calcium, lithium
5.	PXS IRWST	1	Grab	pH, oxygen, fluorine, boron, conductivity, gross specific activity, sodium, sulfate, silica
6.	Main Steam Line (Outlet SG1)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
7.	Main Steam Line (Outlet SG2)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)

Table 9.3.3-2 (Sheet 2 of 5)

Local Sample Point Not in the Primary Sampling System
(Normal Plant Operations)

	Sample Point Name	Available Number of Points	Type of Sample*	Process Measurement
8.	BDS Steam Generator Blowdown	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
9. ~~8.~~	SFS Loops (Upstream of SFS Pumps)	2	Grab	Conductivity, pH, chloride, silica, corrosion product metals, gross activity, corrosion product activity, fission product activity, iodine-131, tritium, turbidity, boron, corrosion product metals, organic impurities
10. ~~9.~~	PCS Water Storage Tank	1	Grab	Hydrogen peroxide
11. ~~10.~~	RC Drain Tank	1	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. Dissolved gases. State and federal environmental discharge requirement such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite
12. ~~11.~~	WLS Degasifier (downstream of degasifier discharge pump)	1	Grab	Dissolved gases.
13. ~~12.~~	CCS Component Cooling Surge Tank	1	Grab	pH, sodium, chloride silica, corrosion product metals, corrosion inhibitors
14. ~~13.~~	CCS Loops (downstream of CCS pumps)	2	Grab	ph, sodium, chloride, silica, corrosion product metals, gross radioactivity and identification and concentration of principal radionuclide and alpha emitters

Table 9.3.3-2 (Sheet 3 of 5)

Local Sample Point Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample*	Process Measurement
15. ~~14.~~ CCS Hot Leg (upstream of CCS pumps)	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
16. ~~15.~~ WLS Discharge	2	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
17. ~~16.~~ WLS Effluent Holdup Tanks MT05A,B	2 ~~1~~	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emmiters
~~17. WLS Effluent Holdup Tank MT05B~~	~~1~~	~~Grab~~	~~Gross radioactivity and identification and concentration of principal radionuclide and alpha emmiters~~
17. ~~18.~~ WLS Waste Holdup Tanks MT06A,B	2 ~~1~~	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emmiters
~~19. WLS Waste Holdup Tank MT06B~~	~~1~~	~~Grab~~	~~Gross radioactivity and identification and concentration of principal radionuclide and alpha emmiters~~
18. ~~20.~~ WLS Effluent Monitor Tanks MT07A,B	2 1	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. State and federal environmental discharge requirement such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite

~~21.~~	~~WLS Effluent Monitor Tank MT07B~~	~~1~~	~~Grab~~	~~Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. State and federal environmental discharge requirement such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite~~

Table 9.3.3-2 (Sheet 4 of 5)

Local Sample Point Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample*	Process Measurement
19. ~~22.~~ WLS Waste Monitor Tanks MT08A,B	2 1	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. State and federal environmental discharge requirement such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite
~~23. WLS Waste Monitor Tank MT08B~~	~~1~~	~~Grab~~	~~Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. State and federal environmental discharge requirement such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite~~
20. ~~24.~~ WLS Ion Exchanger Pre-filter (downstream) ~~Detergent Waste Tank~~	1	Grab	Suspended solids ~~Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters~~
21. ~~25.~~ WLS Ion Exchanger After-filter (downstream)~~Detergent Waste Monitor Tank~~	1	Grab	Suspended solids ~~Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters. State and federal environmental discharge requirement such as pH, suspended solids, oil and grease, iron, copper, sodium nitrite~~
22. ~~26.~~ WLS Chemical Waste Tank	1	Grab	Gross radioactivity and identification and concentration of principal radionuclide and alpha emitters
23. ~~27.~~ WSS Spent Resin Tank (liquid)	1	Grab	Gross radioactivity, radionuclide concentrations

~~Gaseous Sample~~

~~28.~~	~~Reactor Coolant Drain Tank (overpressure gas space)~~	~~1~~	~~Grab~~	~~% oxygen or % nitrogen~~

Table 9.3.3-2 (Sheet 5 of 5)

Local Sample Point Not in the Primary Sampling System
(Normal Plant Operations)

Sample Point Name	Available Number of Points	Type of Sample*	Process Measurement
Gaseous Sample			
24. ~~29.~~ VES MCR Emergency Air Supply Headers	2	Grab	Air quality, oxygen, carbon monoxide, carbon dioxide, contaminants
25. ~~30.~~ WGS Effluent Discharge to Environment	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)
26. ~~31.~~ WGS Inlet	1	Continuous	Oxygen,hydrogen, moisture
~~32. WGS Inlet~~	~~1~~	~~Grab~~	~~Noble gases, iodine, particulates, tritium~~
27. ~~33.~~ WGS Charcoal ~~Guard~~ Bed Vault ~~Outlet~~	1	Continuous~~Grab~~	Hydrogen ~~Moisture, noble gases, iodine, particulates, tritium~~
28. ~~34.~~ WGS Delay Bed Outlets MV02A,B	2 ~~1~~	Grab	Moisture, noble gases, iodine, particulates, tritium
~~35. WGS Delay Bed Outlet MV02B~~	~~1~~	~~Grab~~	~~Moisture, noble gases, iodine, particulates, tritium~~
29. Condenser Air Removal System	1	Continuous	Radiation monitor (See Section 11.5, Table 11.5-1)

* This column shows methods to obtain a sample for ~~chemical~~ analysis. ~~It does not specify the frequency of sampling nor does it specify actual location of sample collection.~~ "Grab" means that a grab sample is required for the intended ~~chemical~~ analysis. Depending on the sampling condition, this grab sample can be obtained in the laboratory or in the grab sampling unit. "Continuous" means that the required ~~chemical~~ analysis is performed via a probe that monitors the sampling stream continuously.

Question 920.5

The December 9, 1993, response to Q920.1 regarding submittal of the vulnerability analysis indicates that this analysis should be performed by the COL holder. The staff interprets Section 5.2.2.1 of Chapter 9 of the EPRI ALWR Requirements Document for passive plants to mean that the designer should perform an analysis to optimize system design with respect to radiological sabotage protection. Therefore, describe how the design process for the AP600 meets this guidance.

Response:

The plant protection system includes both a physical protection system and a security organization. Major features of the AP600 physical protection system which are not site-specific are within the designers scope. The Combined License applicant is responsible for developing the security plan, and for maintaining a security organization.

The designers' scope of the AP600 physical protection system includes: description of the physical protection system, identification of vital equipment, layout of vital area boundaries and protected area enclosures, layout and arrangement of the CAS and SAS, location of controlled access portals, design of security barriers and security hardened walls, and conceptual design of the vehicular barrier system.

Evaluation of vulnerability to radiological sabotage is performed as part of this design process. The vulnerability evaluation will be used to enhance the plant design as required to improve the physical protection system.

A radiological sabotage vulnerability analysis report will be developed by the Combined License applicant and will consider both the physical protection system and the security organization.

SSAR Revision: NONE

Question 952.91

Provide the following information on the CMT test facility:

a. Final as-built drawings and piping and instrumentation diagrams (P&IDs) that include all recent modifications to the facility.

b. A current description (or drawing) of the steam diffuser nozzle, if still in use.

Response:

a. Up to date, final as-built drawings and piping and instrumentation diagrams for the CMT test facility are provided in reference 952.91-1, which was transmitted to the NRC via Westinghouse letter NTD-NRC-94-4244, dated July 29, 1994.

b. A drawing of the CMT steam diffuser is provided in reference 952.91-1, which was transmitted to the NRC via Westinghouse letter NTD-NRC-94-4244, dated July 29, 1994.

Reference:

952.91-1 WCAP-14132, "AP600 CMT Program - Facility Description Report"

SSAR Revision: NONE
PRA Revision: NONE