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NTD-NRC-94-4337
DCP/NRC0242
Docket No.: STN-52-003

November 7, 1994

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

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of April 29, 1994, July 14, 1994, August 18, 1994, and September 2, 1994. This completes the responses associated with the April 29, and July 14 letters. 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. Attachment B is a listing of the questions associated with your letters of April 29, 1994, and July 14, 1994 and the date of the Westinghouse letters that transmitted the responses.

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

In addition to the RAIs, your letter of September 2, 1994 included the following statement.

> In addition, Westinghouse should consider addressing the materials, fabrication, testing, and inspection of the depleted uranium flywheel in a topical report, either as an appendix to the existing WCAPs, or a separate topical report. The flywheel design for the AP600 is a significant departure from previous designs. It is using an unproven material, a depleted uranium alloy which has not been used in a structural or cyclic loaded design application before. The enclosing of the reactor coolant pump flywheel within the reactor coolant pressure boundary, a possible source of internally generated missiles, is a major concern. A major feature of this design is that the pump is not to be taken apart for inspections or maintenance during its entire service life, another major departure from present designs and maintenance practice.

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Of prime importance in the consideration of reactor coolant pump and flywheel integrity is minimizing the potential for generation of missiles from the flywheel in compliance with the requirements of General Design Criteria 4. The AP600 approach is to demonstrate that fragments from a postulated flywheel fracture do not penetrate the surrounding pressure boundary and thus do not become missiles. See the response to RAI 250.11R1 and the topical report WCAP-13734, "Structural Analysis Summary for the AP600 Reactor Coolant Pump High Inertia Flywheel," for additional information on the analysis of the retention of flywheel fragments. Since this information shows that the fragments from a postulated worst case flywheel fracture would be retained by the reactor coolant pump housing, with a large margin, the flywheel does not represent a source of internally generated missiles.

General Design Criteria 4 requires that "...structures, systems, and components shall be appropriately protected against dynamic effects, including the effect of missiles, ...". The AP600 canned motor reactor coolant pump is in full compliance with this requirement. The AP600 approach to meeting this requirement provides for safe operation with more certainty than the Regulatory Guide 1.14 approach of providing for flywheel integrity by testing and inspection.

Disassembly of the reactor coolant pump for inspections or maintenance during its service life is not required to provide safe operation of the pump since inspections of the flywheel are not relied on to prevent missiles from a postulated flywheel fracture.

A topical report specifically dedicated to the materials, fabrication, testing, and inspection of the depleted uranium flywheel is therefore not required to demonstrate the safe operation of the reactor coolant pump. As noted in Subsection 5.4.1 of the SSAR and the response to RAI 251.2, the AP600 canned motor reactor coolant pump uses a fundamentally different approach to demonstrate safe operation of the flywheel than the approach for previous reactor coolant pump designs. This approach does not rely on the integrity of the flywheel to assure safe operation.

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

Brian A. McIntyre for

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

/nja

Enclosure

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

NTD-NRC-94-4337
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED NOVEMBER 11, 1994

RAI No.	Issue
210.066R01	SSAR Sect 3.9.3.2.2, max stress level for valves
210.084	Definition of passive failure
210.113	Justification for use of uranium flywheel
210.114	Special requirements for uranium flywheel
210.115	Experience base for uranium rotating components
210.116	Flywheels exposed to corrosive medium
210.117	Corrosion resistance to uranium flywheel
210.119	Fracture toughness of depleted uranium
210.120	Specifications for uranium castings/forgings
210.122	Flywheel enclosure stresses
210.123	ASME Code for rotating/high cycle application
210.124	Structural analysis of RCP shaft & bearing support
210.125	RCP mean time between failures
210.126	RCP flywheel inspection criteria
210.128	RCP flywheel retention of fracture toughness
210.129	Retention of RCP flywheel dimensions
210.130	RCP flywheel enclosure & weld inspection methods
210.131	RCP flywheel enclosure leak tightness monitoring
210.132	Personnel protection from uranium toxicity
210.133	RCP flywheel ultrasonic testing procedures
210.134	RCP location interference with SG tube examination
210.135	RCP flywheel enclosure leak tightness assurance
210.136	Pipe size used to determine maximum RCP overspeed
210.137	Procedure for RCP repair
210.138	RCP flywheel enclosure leak exposure hazard
210.139	Introduction of uranium corrosion products in RC

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ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED NOVEMBER 11, 1994

RAI No.	Issue
210.140	Consequences of RCP flywheel fracture
210.141	Weld repair to cast or forged uranium flywheel
220.050R01	Factor of safety for sliding & overturning
220.087R01	Energy component for embedment effect
220.093R01	Containment severe accident loading
231.026R01	Properties in SSAR Table 2A-6
281.022	Basis for elemental iodinr decontamination factor
281.023	Basis for particulate iodine removal coefficients
440.253	PRHR drawing illustrating baffle configuration
471.023	Sensitivity of airborne radiation monitors
471.025	Safety related radiation monitors
471.026	Radiation monitor local alarm module
480.015R01	WGOTHIC validation with test data
480.032R01	Prediction of hydrogen distribution
952.101	PCCS interior velocities using WGOTHIC code
952.103	Water distribution model for hot conditions
952.104	Water distribution model support of DBA anal.

ATTACHMENT B

CROSS REFERENCE OF WESTINGHOUSE RAI RESPONSE TRANSMITTALS TO NRC LETTERS OF APRIL 29, 1994, AND JULY 14, 1994

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
210.029	SSAR Section 3.2.2.5	04/29/94	06/27/94
210.030	Acceptability of EPRI NP-6628	04/29/94	05/19/94
210.031	Requested revision to RAI 210.9	04/29/94	06/27/94
210.032	SSAR Section 3.7.3.9	04/29/94	07/08/94
210.033	Acceptable code verification methods	04/29/94	06/30/94
210.034	SSAR Table 3.2-3	04/29/94	06/27/94
210.035	Addition of QA requirements to SSAR Table 3.2-3	04/29/94	06/16/94
210.036	Component cooling water quality group	04/29/94	06/27/94
210.037	Consistency of SSAR Table 3.2-3 & Sect 5.4.7.1.2	04/29/94	06/16/94
210.038	PRHR support safety class	04/29/94	06/27/94
210.039	Class A PRHR HX/Class C IRWST interface	04/29/94	06/27/94
210.040	SSAR Section 3.6.2.1.1.4	04/29/94	06/27/94
210.041	SSAR Section 3.6.2.3.1	04/29/94	07/08/94
210.042	SSAR sections 3.6.2.3.2, 3.9.3.4, 3.10.1.3	04/29/94	06/30/94
210.043	SSAR section 3.6.2.3.4.2	04/29/94	06/27/94
210.044	SSAR section 3.6.2.4 & 3.6.2.4.2	04/29/94	07/08/94
210.045	SSAR section 3.6.2.4.2	04/29/94	07/08/94
210.046	SSAR section 3.7.3.1	04/29/94	05/19/94
210.047	SSAR section 3.7.3.4	04/29/94	06/27/94
210.048	SSAR section 3.7.3.5	04/29/94	07/25/94
210.049	SSAR section 3.7.3.8.2.1	04/29/94	07/08/94
210.050	SSAR section 3.7.3.8.2.1	04/29/94	06/27/94
210.051	SSAR section 3.7.3.9	04/29/94	07/08/94
210.052	SSAR section 3.9.1.1	04/29/94	06/30/94
210.053	SSAR section 3.9.2.1	04/29/94	06/27/94
210.054	SSAR section 3.9.2.1.2	04/29/94	06/27/94
210.055	SSAR sections 3.9.2.1.2 & 14.2.8.2.18	04/29/94	06/27/94
210.056	SSAR section 3.9.2.1	04/29/94	06/27/94
210.057	SSAR sections 14.2, 1.9.2.1.1, 3.9.2.1.2	04/29/94	06/27/94
210.058	SSAR section 14.2.8.1.77	04/29/94	06/27/94
210.059	SSAR sections 3.9.1.1 & 3.9.3.1.2	04/29/94	07/27/94
210.060	Elimination of OBE, SSAR scetion 3.7	04/29/94	06/30/94
210.061	Low pressure side design critreia	04/29/94	06/16/94
210.062	Loading combinations for level D condition	04/29/94	06/30/94
210.063	Method of combination of dynamic responses	04/29/94	07/25/94
210.064	Use of elastic-plastic method of analysis	04/29/94	07/25/94
210.065	SSAR section 3.9.3.1.7	04/29/94	06/30/94
210.066	SSAR sect 3.9.3.2.2, max stress level for valves	04/29/94	07/22/94
210.067	Design/requirements of pressure-relieving devices	04/29/94	07/08/94
210.068	Stress criteria for active component supports	04/29/94	07/25/94
210.069	Section 3.9.3.4.3, snubber operability	04/29/94	07/25/94
210.070	SSAR section 3.9.5.2.4	04/29/94	06/30/94
210.071	Sheet 53 of SSAT Table 3.2-3	04/29/94	06/16/94
210.072	SSAR sections 3.9.7.1 & 3.9.7.3	04/29/94	06/27/94
210.073	ASME Class 1,2,3 components procurement specs	04/29/94	06/30/94
210.074	Exception to position C.7.b in RG 1.124	04/29/94	06/16/94
210.075	Exception to position C.6.b in RG 1.130	04/29/94	06/16/94
210.076	SSAR section 3.6.2	04/29/94	07/27/94
210.077	SSAR section 3.6.2	04/29/94	06/30/94
210.078	WCAP-13054, SSAR Tables 3.9-5, 3.9-6, 3.9-7, 3.9-8	04/29/94	07/08/94
210.079	WCAP-13054, SSAR Tables 3.9-5, 3.9-6, 3.9-7, 3.9-8	04/29/94	07/27/94
210.080	SSAR section 3.9.3	04/29/94	06/30/94
210.081	Dynamic qualification by experience	04/29/94	08/03/94
210.082	Dynamic loads for electrical equipment	04/29/94	06/27/94
210.083	Superimposition of accident load onto seismic load	04/29/94	06/27/94
210.084	Definition of passive failure	04/29/94	11/07/94
210.085	Impact of postulated pipe break on valve dics	04/29/94	07/22/94
210.086	Auditable records - seismic/dynamic qualification	04/29/94	06/27/94
210.087	Valve leakage under seismic loading	04/29/94	07/22/94
210.088	Seismic qualification report	04/29/94	07/25/94
210.089	Advers seismic interaction of nonsafety SSCs	04/29/94	07/22/94
210.090	COL applicant process for seismic interaction	04/29/94	07/29/94
210.091	SRP section 3.9.3	04/29/94	06/16/94

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
210.092	Determination of worst case orientation	04/29/94	06/27/94
210.093	Aging by analysis	04/29/94	06/27/94
210.094	Functionality of CRDM for seismic & LOCA loads	04/29/94	06/27/94
210.095	SSAR section 3.9.2.5	04/29/94	06/30/94
210.096	SSAR section 3.9.5	04/29/94	06/27/94
210.097	SSAR section 3.9.5, basis of deflection allowables	04/29/94	06/30/94
210.098	Thermal stratification in design of RV & internals	04/29/94	07/22/94
210.099	SSAR figures 3.9-5 & 3.9-6	04/29/94	06/27/94
210.100	Flow-induced vibrations of reactor internals	04/29/94	06/30/94
210.101	Key dimensions of reactor vessel & supports	04/29/94	06/27/94
210.102	Preoperational vibration test program	04/29/94	06/30/94
210.103	Stress limits for core support structures	04/29/94	06/30/94
210.104	Exceptions to positions C.1 & C.2 of RG 1.20	04/29/94	06/30/94
210.105	Section 4 of Section 3.9.2 of SRP	04/29/94	06/30/94
210.106	ASME fatigue design curve margin for 60-yr plant	04/29/94	06/27/94
210.107	SSAR section 3.9.3.4	04/29/94	06/30/94
210.108	Justification for use of ASME 1989 Addenda	04/29/94	06/02/94
210.109	List of ASME Code cases used in design	04/29/94	06/30/94
210.110	Inservice testing of pumps and valves	04/29/94	06/30/94
281.020	Post Accident Sampling System compliance	07/14/94	09/01/94
281.021	Hydrogen analysis in containment atmosphere	07/14/94	09/01/94
281.022	Basis for elemental iodinr decontamination factor	07/14/94	11/07/94
281.023	Basis for particulate iodine removal coefficients	07/14/94	11/07/94

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Question 210.66

As stated in Section 3.9.3.2.2 of the SSAR, active valves are those whose operability is relied upon to perform a safety-related function during transients or events considered in the respective operating condition categories. This section references Tables 3.9-9 and 3.9-10 of the SSAR for stress limits used for active Class 1, and Class 2 and 3 valves, respectively. These tables provide no special stress limits for active valves. Note b in Table 3.9-9 states that valve operability is demonstrated by testing. The staff does not agree with the criteria for active valves that allows the calculated stresses to approach Service Level D limits. To provide further assurance of operability, in addition to testing, the staff's position is that the calculated maximum stress in the valves under all Service Levels be less than 1.10 times the allowable yield strength of the applicable material. This will help to insure that the deformation resulting from these loads will be small enough to allow operability. Revise Section 3.9.3.2.2 and Tables 3.9-9 and 3.9-10 of the SSAR to reflect this staff position.

Response: (Revision 1)

Tables 3.9-9 and 3.9-10 will be revised to address the staff's position with these changes, SSAR Section 3.9.3.2.2 does not have to be revised. Response revision 1 clarifies that operability for active check valves may be shown by analysis

SSAR Revision:

Revise Note b to table 3.9-9 to read:

b. Class 1 valve service Level D criteria for ~~active valves and~~ inactive valves is based on the criteria in ASME III, Appendix F, F-1420 for verification of pressure boundary integrity. ~~Valve operability is demonstrated by testing.~~

Add Note g to table 3.9-9 to read:

g. For active valves, pressure integrity verification will be based on using the ASME Code allowables one level less than the service loading condition. For example, the evaluation of Level D loading, Level C allowables will be used. Valve operability is demonstrated by testing. Check valve operability may be shown by analysis. See Subsection 3.9.3.2.2 for an outline of test requirements.

Add Note e to Table 3.9-10 to read:

e. For active valves, pressure integrity verification will be based on using the ASME Code allowables one level less than the service loading condition. For example, for the evaluation of Level D loading, Level C allowables will be used. Valve operability is demonstrated by testing. Check valve operability may be shown by analysis. See Subsection 3.9.3.2.2 for an outline of test requirements.

Revise Table 3.9-9 as follows:

Table 3.9-9

Stress Criteria for ASME Code Section III Class 1 Components[a] and Supports and Class CS Core Supports

Design/Service Level	Vessels/Tanks Pumps	Piping	Core Supports	Valves, Disk & Seats	Components Supports [c][d]
Design and service level A	ASME Code, Section III NB-3221, 3222	ASME Code, Section III NB-3652, 3653	ASME Code, Section III NG-3221, 3222, 3231, 3232	ASME Code, Section III NB-3520, 3525	ASME Code, Section III Subsection NF NF-3221, 3222 NF-3231.1(a) NF-3240
Service level B (Upset)	ASME Code, Section III NB-3223	ASME Code, Section III NB-3654	ASME Code, Section III NG-3223, 3233	ASME Code, Section III NB-3525	ASME Code, Section III Subsection NF NF-3223, 3231.1(a) NF-3240
Service level C (Emergency	ASME Code, Section III NB-3224	ASME Code, Section III NB-3655	ASME Code, Section III NG-3224, 3324	ASME Code, Section III NB-3526	ASME Code, Section III Subsection NF NF-3224, 3231.1 (b) NF-3240
Service level D (Faulted)	ASME Code, Section III (see § 3.9.1.4) NB-3225 (No active Class 1 pumps used)	ASME Code, Section III (see § 3.9.1.4) NB-3656	ASME Code, Section III (see § 3.9.1) NG-3225, 3335	(b) (g)	ASME Code, Section III Subsection NF, (see § 3.9.1) NF-3225, 3231.1(c) NF-3240

Revise Table 3.9-10 as follows:

Table 3.9-10

Stress Criteria for ASME Code Section III
Class 2 and 3 Components and Supports

Design/Service Level	Vessels/Tanks	Piping	Pumps	Valves, Disks, Seats	Component Supports
Design and service level A	ASME Code Section III NC-3217 NC/ND-3310, 3320	ASME Code Section III NC/ND-3652, 3653	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3510	ASME Code Section III NF-3321 NF-3231 NF-3260
Service level B (Upset)	ASME Code Section III NC/ND-3310, 3320	ASME Code Section III NC/ND-3653	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	ASME Code Section III NF-3321 NF-3231 NF-3260
Service level C (Emergency)	ASME Code Section III NC/ND-3310, 3320	ASME Code Section III NC-3654	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	ASME Code Section III NF-3321 NF-3231 NF-3260
Service level D (Faulted)	ASME Code Section III NC/D-3310, 3320	ASME Code Section III NC3655	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520 (e)	ASME Code Section III NF3321 NF 3231 NF-3260

Question 210.84

In the exception to Section 4 of Section 3.9.1 of the SRP described in Revision 1 to WCAP-13054, the last sentence states that a check valve which changes position in response to a pipe rupture event need not meet the criteria for active valves. This does not appear to agree with the staff position in Section B, "Definition of Passive Failure," of SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. The staff recommends that, for passive plant designs, check valves be redefined as active except for those whose proper function can be demonstrated and documented. Revise the exception in WCAP-13054, and any other applicable section in the SSAR to agree with this staff position.

Response:

Safety-related check valves that are required to change position during mitigation of an accident or to achieve safe shutdown are defined as active components. These active check valves will be qualified to operate with the loads they may see after an accident. They may be qualified for these loads by either test or analysis. They will be analyzed together with the piping system to demonstrate that they meet the appropriate ASME code. Refer to SSAR Section 3.9, table 3.9-12 and to RAI 210.66.

The last sentence of the exception to Section 4 of Section 3.9.1 in WCAP-13054 will be deleted because there no longer is an exception.

> The criteria for active valves and supports are not changed. ~~A check valve which changes position in response to a pipe rupture event need not meet the criteria for active valves.~~

SSAR Revision:

See Response Revision 1 for RAI 210.66 for SSAR revisions addressing valve qualification.

Question 210.113

The staff considers depleted uranium to be an unproven material for use in nuclear plants as it has not had proven service in light-water reactor (LWR) plants.

a. Justify the use of uranium for the flywheel and the concept of enclosure (v. no enclosure) as specified in the AP600 standard safety analysis report (SSAR). Describe the basis for the acceptability of this design over the alternative materials considered. Include the advantages of using depleted uranium, considering there is no previous experience with this material for flywheel applications. Discuss the disadvantages of other proven alternative materials considered.

b. Justify the concept of enclosure for the flywheel, rotorbars, etc. Include similar uses and their histories (positive and negative).

Response:

As noted in Subsection 5.4.1 of the SSAR and the response to RAI 251.2, the AP600 canned motor reactor coolant pump uses a different approach to demonstrate safe operation than the approach for previous reactor coolant pump designs. This approach does not rely on the integrity of the flywheel. The integrity of the depleted uranium alloy flywheel is not important to the safety evaluation.

a. Several materials were considered for the reactor coolant pump flywheel. This discussion gives an explanation for the rejection of the most obvious materials. Steel, which is known as the conventional reactor coolant pump flywheel material, was removed from consideration due to the relative low density of steel. To obtain the required rotating inertia from a flywheel made of steel, the flywheel size could not be practically incorporated within a canned motor design. Other potential materials such as tungsten, tantalum, gold, and platinum were dismissed from consideration due to characteristics that could not compete with uranium. These characteristics included material costs and material availability, difficulties in working with the material, and inferior material properties. Uranium offers excellent density and material characteristics for a flywheel application. Due to these characteristics it is possible to design a flywheel that is compact enough to fit within the canned motor pump confines and structurally sound enough to provide flywheel integrity. The chemical behavior of uranium in reactor coolant is well known, in the unlikely event that the jacket should fail and allow contact with the reactor coolant.

b. The concept of a chromium-nickel-iron alloy enclosure to protect the flywheel from the corrosive reactor coolant was chosen because it offers the most complete and effective means of protecting the flywheel. Westinghouse has extensive experience using chromium-nickel-iron alloy enclosures in reactor coolant pumps to protect components susceptible to the corrosive properties of the reactor coolant system. Please refer to the response to RAI 210.117 for a discussion on alternative means to provide corrosion resistance to the flywheel.

SSAR Revision: NONE

Question 210.114

Discuss any special requirements that need to be imposed on the reactor coolant pump flywheel because of the use of uranium besides requiring a protective enclosure.

Response:

The use of an uranium flywheel will not require any special requirements during normal operation, inspection, and maintenance of the reactor coolant pump.

SSAR Revision: NONE

Question 210.115

Discuss any previous experience or prototypes of rotating machinery with uranium rotating structural components.

Response:

The flywheel design has been prototype manufactured and tested. Testing has been performed up to speeds of 1785 RPM and has included:

- Friction Dynometer testing (Drag Loss)
- Flywheel as a thrust and radial bearing
- Modifications to improve drag loss

SSAR Revision: NONE

Question 210.116

Discuss any previous experience or prototypes of rotating machinery with a flywheel susceptible to corrosion in a welded enclosure exposed to a corrosive medium.

Response:

Westinghouse manufactures canned motor pumps for other applications, where the pump design is required to have a minimum specified inertia. For these cases, the rotor provides the rotating inertia and acts as a "flywheel." This design includes a rotor lamination of electrical steel that is susceptible to corrosion. The rotor can serves as a welded enclosure to separate the corrosion susceptible materials from the corrosive environment. The welds used in the AP600 flywheel assembly construction are similar to the proven welds used in rotor construction.

SSAR Revision: NONE

Question 210.117

Have other means of providing corrosion resistance to the uranium flywheel, such as weld overlay (Zr, Ta, Ti) or plasma spray coatings been considered?

Response:

Other means for providing corrosion resistance to the uranium flywheel were considered. From these evaluations it was decided that a nickel-chromium-iron alloy enclosure would be the most effective and desirable option. There are concerns in how a plasma spray coating or a corrosive resistant plating would hold up in the reactor coolant pump application. There is broad experience with seal welded nickel-chromium-iron alloy and it has proven to hold up well when used as a reactor coolant pump can.

SSAR Revision: NONE

Question 210.119

a. How do you ensure adequate fracture toughness of the depleted uranium castings/forgings for the 60-year life of the plant? The Deel and Burian report covers only one casting, and that one data point can not justify predicting a correlation between fracture performance and a given C_v (See Q210.121b).

b. What elements (alloy and residual) will adversely affect the fracture toughness of uranium?

c. Will the alloying elements, including residual elements, cause a degradation of properties, particularly fracture toughness, at the temperatures at which this material will be used over the life of the component?

d. Will thermal cycling over the life of the plant cause a significant change in properties of the uranium alloy chosen?

Response:

a. Adequate fracture toughness of the uranium alloy casting, as well as the appropriate range of other important material properties, is provided by control of alloying elements, contaminants, casting practices, and the heat treatment process. Control of these parameters will provide for a material that is consistent with the 10 ft. lb. Charpy test data for this material that is the basis for the fracture toughness value of 50 ksi$\sqrt{in}$. The test data came from Reference 210.119-1. Testing performed by a casting supplier shows that this level can be consistently achieved.

b. The controlled elements are listed in the SSAR Table 5.4-2 and will be specified as follows:

Carbon	150	ppm Max.
Iron	75	ppm Max.
Silicon	75	ppm Max.
Copper	20	ppm Max.
Aluminum	20	ppm Max.

c. The flywheel operates in the motor bearing region, below the thermal barrier, and does not see the pumped fluid temperature. The bearing water temperature will be around 150°F for normal operating conditions. Due to this low temperature, no degradation of material properties is expected.

d. As described above, the service temperature of the flywheel is sufficiently low that thermal aging of the flywheel is not expected to occur. According to Reference 210.119-2, a temperature of 350°C (662°F) is needed for thermal aging.

References:

210.119-1. Deel and Burian, Battelle BMI-2032, "The Mechanical Properties of Depleted Uranium," prepared for US DOE, July 16, 1979.

210.119-2. "Metals Handbook," Desk Edition, American Society for Metals, 1985.

SSAR Revision: NONE

Question 210.120

Provide:

a. the specifications for the uranium in both part forms (casting and forging).

b. the specified alloy control, residual element control, grain morphology, and NDE requirements.

Response:

a. Please refer to RAI 210.119 response part b. for a discussion on the uranium material specifications.

b. The alloy control, grain, and NDE is controlled by melting and casting practice and by specifications in the reactor coolant pump design specification and material procurement document.

SSAR Revision: NONE

Question 210.122

Demonstrate the adequacy of the flywheel enclosure's use of stresses higher than those recommended by the Standard Review Plan guidelines for normal and design speeds.

Response:

As noted in Subsection 5.4.1 of the SSAR and the response to RAI 251.2, the AP600 canned motor reactor coolant pump uses a different approach to demonstrate safe operation than the approach for previous reactor coolant pump designs. This approach does not rely on the integrity of the flywheel. Therefore, the stresses in the flywheel enclosure are not important to the safety evaluation.

The seal welds and the local areas next to the welds are reduced in thickness to provide flexibility and permit radiographic inspection of the welds. The weld and reduced areas may have stresses greater than the guidelines in Standard Review Plan for flywheel stresses at normal and design speeds. However, the stress in the seal welds and flywheel enclosure components for normal and design speeds are within the criteria in Subsection NG of the ASME Code, Section III that is used as a guideline. Subsection NG is used as an alternate to the guidance in the Standard Review Plan and Regulatory Guide 1.14 since the flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The strength of the enclosure is provided primarily by the strength of the parts resisting the centrifugal forces.

SSAR Revision: NONE

Question 210.123

The ASME Code is used for pressure vessels and piping, and at most a low-cycle fatigue environment. Why would the ASME Code be appropriate for rotating machinery, and a high-cycle fatigue application?

Response:

As noted in Subsection 5.4.1 of the SSAR and the response to RAI 251.2, the AP600 canned motor reactor coolant pump uses a different approach to demonstrate safe operation than the approach for previous reactor coolant pump designs. This approach does not rely on the integrity of the flywheel. Therefore, the criteria used for evaluation of the flywheel assembly is not needed for the safety evaluation.

Subsection NG is used as an alternate to the guidance in the Standard Review Plan and Regulatory Guide 1.14 since the flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The loading on the flywheel assembly is not a high cycle fatigue loading because the number of large amplitude stress cycles will correspond to the pump heatup and cooldown cycles and stop and start cycles. The use of Subsection NG criteria developed for core supports is considered appropriate since the flywheel assembly operates in a reactor coolant environment.

SSAR Revision: NONE

Question 210.124

Provide the structural analysis of the shaft and bearing supports taking into account normal operation, anticipated transients, the design basis of a loss-of-coolant accident and a safe-shutdown earthquake.

Response:

The detailed structural analysis of the rotating components and bearings is part of the final analysis design package and is not required to support design certification. This analysis is a post design certification activity and is done as part of the detail design activities. This analysis will account for steady state operation, emergency, design overspeed, and seismic conditions. Preliminary rotor dynamics analysis and seismic analysis show acceptable results for the shaft and bearings supports. The final analysis requirements are established by the design specification.

SSAR Revision: NONE

Question 210.125

a. Provide, for this design, the standard reliability/maintainability information, i.e., mean time between failures, etc.

b. Compare the AP600 RCP with the present RCP design.

Response:

a. The estimated downtime for the AP600 due to unavailability of the canned motor reactor coolant pump is conservatively estimated at 10.5 hours per year. This number was developed by evaluating failures in canned motor pumps and considering the design changes and operating practices developed to address the failures. The AP600 canned motor reactor coolant pump is expected not to have planned inspection or refurbishment. It has no wear components that will require replacement. Mean time between failures has not been developed.

b. The historic number for unavailability of shaft seal pumps in Westinghouse designed two loop plants is 26.3 hours per year.

SSAR Revision: NONE

Question 210.126

In response to Q251.14, it was stated that the welds connecting the pieces of the enclosure are seal welds. The ASME Code does not credit seal welds, and zero joint efficiency should be used for such joints. The uranium flywheel and its Inconel 600 enclosure are parts in rotating machinery. The factor that controls rotating machinery design is usually high cycle fatigue, and in a water environment, corrosion fatigue.

a. Demonstrate that the inspection criteria for the uranium flywheel and its welded Inconel 600 enclosure, based upon the ASME Code, are appropriate for the 60-year design life.

b. The seal weld is essentially a partial penetration butt weld. Describe the inspections and testing to be used to assure a 60-year life. Specify the acceptance criteria to be used and the relationship of the criteria to life performance.

Response:

The flywheel enclosure welds are not pressure boundary welds and therefore are not subjected to the ASME Code Section XI pressure boundary weld inspection criteria. The cyclic loading on the flywheel and enclosure is due to the heatup and cooldown cycles and pump start and stop cycles. The number of these cycles is much lower than the number of stress cycles experienced by the shaft and bearings due to rotation.

a. The welds and materials that are being used on the flywheel enclosure are designed, selected, and tested to meet ASME Code, Section III criteria. These welds and materials are similar to the welds and materials that are used to seal the stator and rotor cans and operate in the same environment. Westinghouse has a great deal of operational experience in designing and manufacturing stator and rotor cans. This experience will transfer to creating an enclosure that is sufficiently robust to withstand the cyclic fatigue and corrosion fatigue experienced by a canned motor reactor coolant pump flywheel assembly.

b. The welds are designed and examined to ASME Code Section III requirements. The inspections and testing of the enclosure welds are not relied on to *assure* the 60-year design life. There will not be any requirement for inservice inspection or testing of the flywheel assembly enclosure welds. The enclosure welds are not credited in the flywheel missile analysis. The approach used to show safe operation of the reactor coolant pump does not rely on the integrity of the flywheel. Therefore, the inspection and testing criteria for the flywheel enclosure are not important to the safety evaluation.

SSAR Revision: NONE

Question 210.128

Provide the data that has been developed to demonstrate the retention of adequate fracture toughness of the uranium flywheel for the 60-year life under operating conditions, i.e., constant temperature, temperature cycling, and stress and temperature cycling together.

Response:

As noted in Subsection 5.4.1 of the SSAR and the response to RAI 251.2, the AP600 canned motor reactor coolant pump uses a different approach to demonstrate safe operation than the approach for previous reactor coolant pump designs. This approach does not rely on the integrity of the flywheel. Therefore, the fracture toughness of the flywheel does not need to be considered in the safety evaluation.

The heat treatment on the flywheel material is 1000°C for 24 hours followed by a slow cool at less than 100°C per hour in a vacuum of 10^{-4} torr. The slow cooling will provide a stable metallurgical structure that is relatively stress free. The operating temperature of the flywheel is approximately 150°F. Even in less stable, quenched condition, aging will only occur with temperatures in the region of 300°C (518°F). (See Reference 210.128-1). The stresses on the flywheel are low and will not produce strain induced changes in the flywheel properties. The low thermal and stress transients of the design are not expected to produce degradation of properties.

Reference:

210.128-1 Metals Handbook, Desk Edition ASM 1985

SSAR Revision: NONE

Question 210.129

Uranium is an anisotropic material, i.e., it has different mechanical properties in different crystallograghic directions. The WCAP allows for the uranium flywheel to be forged. With forged components, the plastic deformation of the process promotes preferred crystallographic orientations. What data has been developed to demonstrate the retention of dimensions of the uranium flywheel as a casting and as a forging for the 60-year life under operating conditions, i.e., stresses at constant temperature, temperature cycling, and stress and temperature cycling together?

Response:

The AP600 canned motor reactor coolant pump will not use forged material for the flywheel. The uranium casting material does not exhibit anisotropy. Metalograpy shows an equiaxed grain structure to exist.

SSAR Revision: NONE

Question 210.130

Provide the inspection methods, procedures, and rejection criteria for the finished Inconel 600 enclosure and its welds.

Response:

The nondestructive examination for the enclosure is visual 5X, and PT with no bleeds.

SSAR Revision: NONE

Question 210.131

Specify the means of monitoring the leak tightness of the Inconel 600 enclosure during the 60-year life and revise the SSAR accordingly.

a. Explain how the soundness (freedom from significant casting defects) of the parts produced from cast depleted uranium is ensured.

b. Describe the inspection requirements for the finished machined uranium flywheel.

c. Provide procedures, special techniques, and acceptance standards used.

Response:

As stated in SSAR paragraph 5.4.2.3.6.3, a leak in the flywheel enclosure will be detected by an out of balance vibration in the RCP, which is monitored by the RCP vibration monitors. The pump vibration monitors are designed and will be calibrated to detect small variations in the flywheel mass.

a. Soundness and (freedom from significant defects) is ensured by a two-step process. The first step is to employ strict casting procedures. The melting and casting processes are performed in a vacuum. The second step involves strict inspection techniques. The soundness of the casting will be verified by UT and PT and then the final machined surfaces of the flywheel will be PT tested for flaws.

b. The finished machined uranium flywheel will be liquid penetrant tested (PT) for flaws.

c. The liquid penetrant examination of a sample flywheel was performed to ASME Code, Section V criteria. The planned dye penetrant method to be used is the solvent removable red dye on 100 percent of the finished machined surfaces. Acceptance will be as specified in the applicable code or standard. Specific procedures and acceptance criteria will be developed by the pump design organization as part of the final design efforts. These procedures will not be developed in time to support Design Certification.

   The ultrasonic testing of the uranium alloy casting is performed in both the axial and radial directions. Ultrasonic testing of a sample flywheel casting used a 1/8 inch diameter flat bottom hole and good resolution was obtained. Based on this experience, the acoustic properties of a uranium casting support inspection to the requirements of SA-609. The configuration may allow shear wave examination although there is not an available history of performing shear wave testing on this material. Specific procedures and acceptance criteria will not be developed in time to support Design Certification.

SSAR Revision: NONE

Question 210.132

Provide any special procedures required to protect personnel from toxicity of the uranium while performing routine maintenance and inspection in an operating plant.

Response:

There are no routine inspection and maintenance activities that would require opening of the reactor coolant pump flywheel assembly. No special AP600 plant procedures are required during routine inspection and maintenance to provide protection from uranium toxicity.

SSAR Revision: NONE

Question 210.133

a. Explain whether or not the ultrasonic testing procedures of the uranium flywheel will be subject to the requirements of Appendix VIII of Section XI, 1989 Addenda.

b. Demonstrate that the ultrasonic testing procedure(s) for the uranium flywheel is adequate and appropriate.

Response:

a. Ultrasonic examination will be performed in accordance with ASME Code Section III, Paragraph NB-2574. There are not requirements to do insevice inspection of the flywheel assembly.

b. Since it is not feasible to perform radiographic examinations on the flywheel casting due to the thickness and density of the uranium, ultrasonic testing is the choice for volumetric inspection that satisfies the ASME Code requirements. Based on results from UT testing that has been performed on the sample flywheel, the acoustic properties of uranium cast material are acceptable. This material is capable of being inspected to the requirements of the ASME Code, Section III,Paragraph NB-2574, specifically using criteria from procedure ASTM-A-609.

SSAR Revision: NONE

Question 210.134

Evaluate whether or not the location of the RCP housings and associated components interfere with eddy-current examination of all of the steam generator tubes.

Response:

The reactor coolant pump is attached to and supported by the steam generator nozzle. It does not interfere with access through the steam generator manways, eddy current inspection, or other inspection or maintenance activities of steam generator tubes.

SSAR Revision: NONE

Question 210.135

Describe the means of ensuring the leak tightness of the Inconel 600 enclosure for the 60-year design life.

Response:

The leak tightness of the flywheel enclosure is provided by the manufacturing and inspection procedures as outlined in the response to RAI 210.130. The response to RAI 210.131 describes how a leak in the flywheel enclosure would be detected.

SSAR Revision: NONE

Question 210.136

Identify the largest pipe (system, size, and maximum pressure) that is not qualified for leak-before-break that is used to determine maximum overspeed of the RCP.

Response:

The largest diameter pipe connected to the reactor coolant loop that will not be qualified for leak-before-break is 3 inches. Previous piping and reactor coolant pump analyses have demonstrated that a 3 inch pipe break will not release sufficient reactor coolant system volume to create a reactor coolant pump 125% overspeed condition. Therefore, the maximum overspeed is conservatively 125 percent of nominal speed.

SSAR Revision: NONE

AP600

Question 210.137

In the case of a loss in leak tightness of the Inconel 600 flywheel enclosure, will there be procedures in place that will specify the shipping, storing, handling, and overhaul to be required for the repairs of the RCP and its associated heat exchanger contaminated with uranium corrosion products? Is Westinghouse providing guidelines for the development of each procedures? These guidelines should be referenced in the SSAR.

Response:

The procedures required to store, handle, ship, or overhaul the canned motor pump with postulated depleted uranium contamination are not significantly different than those for any type of radioactive contamination. Precautions for addressing postulated depleted uranium contamination will be included in the reactor coolant pump inspection manual. Referencing these guideline in the SSAR is not required.

SSAR Revision: NONE

Question 210.138

In the event of a leak developing in the Inconel 600 enclosure:

a. What controls or operational procedures will exist to prevent spreading the uranium corrosion products once failure of the flywheel's enclosure has been detected?

b. What would be the consequences of the spread of uranium corrosion product to all components in contact with reactor coolant?

c. What would be the exposure hazard to workers in the worst-case scenario?

d. What are the specified corrective measures in the event of a loss of leak tightness of the Inconel 600 flywheel enclosure?

e. What are the requirements/procedures for repair in the event of loss of leak tightness of the Inconel 600 enclosure?

Response:

a. To prevent spreading of depleted uranium contamination from a postulated leak of the flywheel enclosure the plant would be shut down and the affected pump stopped.

b. The consequences of the spread of postulated depleted uranium flywheel corrosion product to components in contact with the reactor coolant would be no more adverse than the consequences of corrosion product transport or the release of fission products from fuel failures.

c. There would be minimal exposure hazard due to the worst case postulated leak of a flywheel enclosure to properly protected workers.

d. Recommended corrective measures for addressing a leak in the flywheel enclosure will be included in the reactor coolant pump instruction manual. These could include shutdown and replacement or refurbishment of the pump.

e. The requirements for the repair and refurbishment of a pump found with a leaking flywheel enclosure will be determined during an inspection of the reactor coolant pump.

SSAR Revision: NONE

Question 210.139

Specify the effectiveness of the heat exchanger water system and its motor purge water to restrict the introduction of uranium corrosion products into the reactor coolant.

Response:

The flywheel is encased in a welded enclosure. It is not expected that there will be any communication of depleted uranium corrosion products across the enclosure or the enclosure welds.

The function of the cooling jacket on the outside of the motor housing and the thermal barrier between the pump casing and rest of the pump internals is to maintain the pump motor at an acceptable temperature. The cooling jacket and thermal barrier have no effect on the movement of contaminants into or out of the pump. The AP600 reactor coolant pump will not have a motor purge.

SSAR Revision: NONE

Question 210.140

In the event of flywheel fracture with the boundary remaining intact:

a. What is the threat of these broken parts getting into the reactor vessel and causing other damage or preventing other components from performing their functions?

b. Are there any radiation hazard conditions in such a scenario?

Response:

a. In the event of a flywheel failure, the parts will remain contained within the flywheel region of the reactor coolant pump. The pump has been designed to limit the communication between the fluid in reactor coolant system and the fluid internal to the pump. This is accomplished by the circulatory flow of the reactor coolant pump cooling system and the labyrinth seal between the reactor coolant pump motor region and hydraulic region.

b. There is no special radiation hazard in the event of a fracture of the depleted uranium flywheel.

SSAR Revision: NONE

Question 210.141

Are weld repairs to a cast or forged uranium flywheel allowed? Will weld build-up repair be allowed for machining errors, dents, or gouges on the flywheel?

Response:

There are no weld repairs allowed or qualified for repair of the uranium flywheel.

SSAR Revision: NONE

Question 220.50

The factor of safety against sliding and overturning the nuclear island due to tornado and wind should be provided. In Table 3.8.5-1, provide the rationale for the buoyancy force criterion for the submerged structure (Section 3.8.5 of the SSAR).

Response: (Revision 1)

SSAR Subsections 3.8.5.5.3 and 3.8.5.5.4 are revised to provide the factors of safety against sliding and overturning of the nuclear island due to tornado and design wind loads.

In Table 3.8.5-1, the buoyant force on submerged structure, "B", is applied to consider the effect of high ground water table. Table 3.8.5-1 is revised to clarify that "B" is considered in conjunction with SSE, wind and tornado events in the AP600 design.

~~The factors of safety (F.S.) against sliding and overturning of the nuclear island due to tornado and design wind loads are as follows:~~

- ~~F.S. due to Tornado Load:~~

  ~~Sliding, N-S direction = 6.8 , E-W direction = 6.0~~

  ~~Overturning, N-S direction = 19.6 , E-W direction = 8.0~~

- ~~F. S. due to Design Wind Load:~~

  ~~Sliding, N-S direction = 10.2 , E-W direction = 9.3~~

  ~~Overturning, N-S direction = 47.3 , E-W direction = 22.8~~

~~The buoyant force on the submerged structures used in the flotation evaluation is that due to the maximum high ground water level specified in Table 2.0-1 of the SSAR. The design condition for high ground water table is a severe environmental condition. A minimum factor of safety equal to 1.5 is applied in the evaluation of buoyancy force on the submerged structure.~~

SSAR Revision:

Replace existing SSAR Sections 3.8.5.5.2, 3.8.5.5.3 and 3.8.5.5.4, and SSAR Table 3.8.5-1 with the following revision as shown below. Please note that the revision to SSAR Section 3.8.5.5.4 includes changes identified per Revision 1 of RAI 220.78.

3.8.5.5.2 Flotation

The factor of safety against flotation of the nuclear island is calculated as follows:

$$F.S. = \frac{W}{F}$$

where:

$F.S.$	= factor of safety against flotation from design basis flood
W	= total weight of structures and foundation
F	= buoyant force due to the design basis flood

The factor of safety against flotation for the nuclear island is 3.2. As shown in Table 3.8.5-1, the minimum required factor of safety against flotation is 1.1.

3.8.5.5.3 Sliding

The factor of safety against sliding of the nuclear island during a tornado or a design wind is calculated as follows:

$$F.S. = \frac{F_s + F_p}{F_H}$$

where:

$F.S.$	=	factor of safety against sliding from tornado or design wind
F_s	=	shearing or sliding resistance at bottom of basemat
F_p	=	maximum soil passive pressure resistance, neglecting surcharge effect
F_H	=	maximum lateral force due to active soil pressure, including surcharge, and tornado or design wind load

The factor of safety against sliding of the nuclear island during a tornado is 6.9 in the north-south direction and 6.1 in the east-west direction. The factor of safety against sliding of the nuclear island during a design wind is 12.5 in the north-south direction and 9.5 in the east-west direction.

When only 50% of the maximum soil passive pressure resistance is considered, the factor of safety against sliding of the nuclear island during a tornado equals 5.4 in the north-south direction and 4.4 in the east-west direction. The factor of safety against sliding of the nuclear island during a design wind equals 9.8 in the north-south

direction and 6.9 in the east-west direction.

As shown in Table 3.8.5-1, the minimum required factor of safety against sliding during a tornado and a d gn wind are 1.1 and 1.5, respectively.

The factor of safety against sliding of the nuclear island during a safe shutdown earthquake (SSE) is calculated as follows:

$$F.S. = \frac{F_s + F_P}{F_D + F_H}$$

where:

$F.S.$	=	factor of safety against sliding from a SSE
F_s	=	shearing or sliding resistance at bottom of basemat
F_P	=	maximum soil passive pressure resistance, neglecting surcharge effect
F_D	=	maximum dynamic lateral force, including dynamic active earth pressures
F_H	=	maximum lateral force due to all loads except seismic loads

As shown in Table 3.8.5-1, the minimum required factor of safety against sliding during a SSE is 1.1.

For the hard rock and soft rock sites, the factor of safety against sliding of the nuclear island during a SSE is 1.5 in the north-south direction and 1.6 in the east-west direction. When only 50% of the maximum soil passive pressure resistance is considered, the factor of safety equals 1.2 in the north-south direction and 1.3 in the east-west direction.

For the soft-to-medium stiff soil site, the factor of safety against sliding of the nuclear island during a SSE is 1.3 in the north-south direction, and 1.5 in the east-west direction. For a factor of safety equal to 1.1, the minimum required factor of safety as shown in Table 3.8.5-1, the required soil passive pressure resistance equals 67% and 51% of the maximum, in the north-south direction and the east-west direction, respectively.

3.8.5.5.4 Overturning

The factor of safety against overturning of the nuclear island during a tornado or a design wind is calculated as follows:

$$F.S. = \frac{M_R}{M_O}$$

where:

$F.S.$	=	factor of safety against overturning from tornado or design wind
M_R	=	resisting moment
M_O	=	overturning moment of tornado or design wind

The factor of safety against overturning of the nuclear island during a tornado is 17.7 in the north-south direction and 8.6 in the east-west direction. The factor of safety against overturning of the nuclear island during a design wind is 61.0 in the north-south direction and 24.5 in the east-west direction. As shown in Table 3.8.5-1, the minimum required factor of safety against overturning during a tornado and a design wind are 1.1 and 1.5, respectively.

The factor of safety against overturning of the nuclear island during a safe shutdown earthquake (SSE) is evaluated using the static moment balance approach assuming overturning about the edge of the nuclear island at the bottom of the basemat. The factor of safety is defined as follows:

$$F.S. = \frac{M_R}{M_O}$$

where:

$F.S.$	=	factor of safety against overturning from a SSE
M_R	=	nuclear island's resisting moment against overturning
M_O	=	maximum SSE induced overturning moment acting on the nuclear island, applied as a static moment

The resisting moment is equal to the nuclear island dead weight minus maximum SSE vertical force and buoyant force from ground water table multiplied by the distance from the edge of the nuclear island to its center of gravity. The factor of safety against overturning of the nuclear island during a SSE is 3.5 in the north-south direction and 2.0 in the east-west direction. As shown in Table 3.8.5-1, the minimum required factor of safety against overturning during an SSE is 1.1.

Table 3.8.5-1

Minimum Required Factor of Safety for Overturning and Sliding of Structures

Load Combination	Overturning	Sliding	Flotation
D + H +B + W	1.5	1.5	-
D + H +B + E_s	1.1	1.1	-
D + H +B + W_t	1.1	1.1	-
D + F	-	-	1.1

where:

D = dead load excluding the fluid loads
H = lateral earth pressure
B = buoyant force on submerged structure due to high ground water table
W = wind load
E_s = safe shutdown earthquake load
W_t = tornado load
F = buoyant force due to the design basis flood

Question 220.87

For evaluating the dynamic stability of the NI structures against overturning, provide formulas for calculating the energy component due to embedment effect "Wp" and energy component due to buoyancy "Wb" in Section 3.8.5.5.4 of the SSAR.

Response: (Revision 1)

The methodology used in the dynamic stability evaluation of the nuclear island structures against overturning was changed from the energy balance technique to the static moment balance method. The factor of safety against overturning of the nuclear island during a safe shutdown earthquake is defined as follows:

$$F.S. = \frac{M_R}{M_O}$$

where:

$F.S.$	=	factor of safety against overturning
M_R	=	nuclear island's resisting moment against overturning
M_O	=	maximum SSE induced overturning moment acting on the nuclear island, applied as a static moment

This calculation uses the static moment balance method assuming overturning about the edge of the nuclear island at the bottom of the basemat. The resisting moment is equal to the nuclear island dead weight minus maximum SSE vertical force and buoyant force from ground water table multiplied by the distance from the edge of the nuclear island to its center of gravity. The factor of safety against overturning of the nuclear island is 3.5 in the north-south direction and 2.0 in the east-west direction. As shown in Table 3.8.5-1, the minimum required factor of safety against overturning during an SSE is 1.1.

~~In the dynamic stability evaluation of the NI structures against overturning, the energy component due to embedment effect "Wp" was conservatively neglected and assumed equal to zero. The energy component due to buoyancy "Wb" is determined as follow:~~

$$\cancel{W_b = B \times h}$$

~~where: B = buoyant force~~

~~h = height to which the center of mass must be lifted to reach the overturning position~~

SSAR Revision:

SSAR Section 3.8.5.5.4 is revised as shown in the SSAR revision presented in the response to RAI 220.50, Revision 1.

Question 220.93

Under severe accident loading, thermal expansion of the containment shell is restrained at the transition region. The restraint of thermal expansion produces a compressive hoop stress in the containment vessel in the vicinity of the discontinuity. Thus, the effect of severe accident temperature loading needs to be evaluated to assure that the expected compressive hoop stresses resulting from the load at the transition region (along the entire periphery of the shell) do not lead to buckling of the containment shell causing a loss of containment function. Provide the results of the buckling analysis of the containment shell under the severe accident temperature loading in the SSAR.

Response: (Revision 1)

~~The effect of temperature loading at the base of the containment vessel is being evaluated. The results of this evaluation will be provided by December 1994.~~

The containment vessel design includes a Service Level A combination in which the vessel above elevation 100′ is conservatively specified at the design temperature of 280°F and the portion of the embedded vessel (and concrete) is specified at a temperature of 70°F. Stress analyses and buckling analyses have been performed for this condition.

Containment shell buckling was evaluated using a BOSOR-5 model of the portion of the shell above elevation 100′ extending up to the horizontal stiffener at elevation 132′ 3″. Material yield and stiffness properties were based on properties at the design temperature of 280°F. Temperature differences were raised by small increments until buckling was predicted. Buckling occurred 20 inches above elevation 100′ for a circumferential wave number, N = 190, at a factor of 6.0 times the design differential temperature condition. The half buckling wave length is less than $0.5\sqrt{(rt)}$. This is not a significant buckling issue; buckling did not occur for wave numbers below N = 60, which is the critical range for the cylinder and top head under external and internal pressure.

The containment pressure and temperatures during a severe accident are described in the response to RAI 480.78. There are no cases to be considered within ASME Service Level C limits. The case for severe accident concurrent with loss of the PCS water cooling is a very low probability and is outside the design basis. The maximum containment shell temperature for this case is 295°F as described in the response to RAI 480.78. This is close to the 280°F evaluated against Service Level A limits for the design basis accidents. This severe accident case would not be critical since the design basis accident is evaluated against Service Level A limits and would be more critical than the severe accident case.

The thermal, stress and buckling analyses described above demonstrate that the compressive hoop stresses resulting from the load at the transition region do not lead to buckling of the containment shell causing a loss of containment function.

SSAR Revision: NONE

Question 231.26

The properties of the soft-to-medium soil column given in Table 2A-6 of the SSAR show the shear wave velocity varying linearly from 1000 fps to 2400 fps. Typical variations at sandy soil sites are expected to be curvilinear, with most of the increase in soil stiffness occurring near the upper one-third part of the soil layer due to the nonlinear effects of depth of burial on stiffness. Because such variations may lead to significant differences in soil pressures over the depth of embedment of the NI, as well as changes in free-field ground motions at the foundation mat, provide a comparison of free-field motions at the foundation level obtained from SHAKE deconvolution analysis to indicate the sensitivity of response to this assumption.

Response: (Revision 1)

The case of soft-to-medium soil with shear wave velocity of 1000 ft/sec at grade level increasing to 2400 ft/sec at 240 ft depth with parabolic distribution was analyzed. The maximum low-strain shear wave velocity and the strain-compatible velocity obtained from the SHAKE analysis, using the 1990 Idriss soil strain degradation curve described in the response to RAI 230.79, are shown in Figure 231.26-1. The response spectra of the free-field motion at 40 ft depth (foundation mat elevation) corresponding to parabolic and linear distribution of shear wave velocity case are compared in Figure 231.26-2. As shown in this figure, the response motion using parabolic velocity distribution shows small variations above and below that of the linear velocity distribution depending on the frequency. In order to study the effect of parabolic velocity distribution on the SSI responses, the SSI model in the EW direction was re-analyzed. In this case, the water table was considered at the grade level and the base rock was modeled at 120 ft depth. The strain-compatible properties obtained from the SHAKE analysis were used in the SSI analysis.

The SSI responses of this case are compared with the SSI responses of the same case with linear velocity distribution and enveloping design responses obtained from 3D analysis in Figures 231.26-3 through 231.26-13. These comparisons show that the responses with the parabolic distribution are generally similar to those with the linear distribution and enveloped by the design response spectra obtained for the 3D analyses. There is additional rocking of the nuclear island at a frequency of 3 Hertz which also results in higher peaks at this frequency for the containment internal structures (CIS) and the steel containment vessel (SCV). This increase at the 3 Hertz frequency is attributable to a combination of the following:

- The seismic input at the foundation level is higher for the parabolic distribution as shown in Figure 231.26-2.
- The stiffer soil properties for the parabolic distribution increase the rocking frequency for the nuclear island. This increased stiffness shifts the response closer to that of the soft rock. A comparison of the responses from the 3D design analyses are provided in SSAR Figures 3.7.2-25, 3.7.2-26 and 3.7.2-27.
- The H2 seismic input has significant frequency content at 3 Hertz. This is shown in SSAR Figure 3.7.1-7.

The design soil profile used for the 3D SASSI design analyses for the soft-to-medium soil case will be revised to the parabolic profile using the 1990 Idriss soil strain degradation curve. As shown in the figures, this will amplify the responses at about 3 Hertz and slightly reduce the responses at 2 Hertz. The revised analyses will be documented in the SSAR by July, 1995.

SSAR Revision: NONE

Figure 231.26-1
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

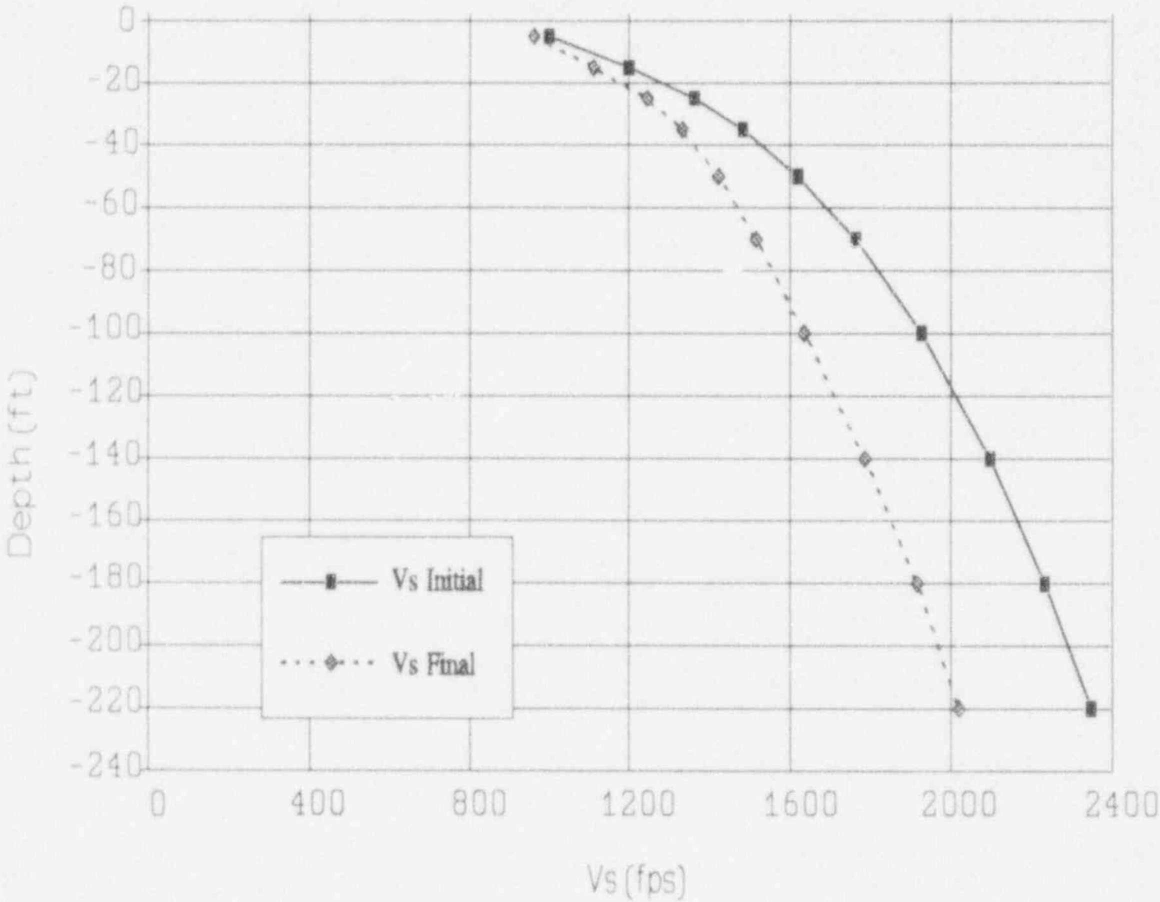


Figure 231.26-2
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

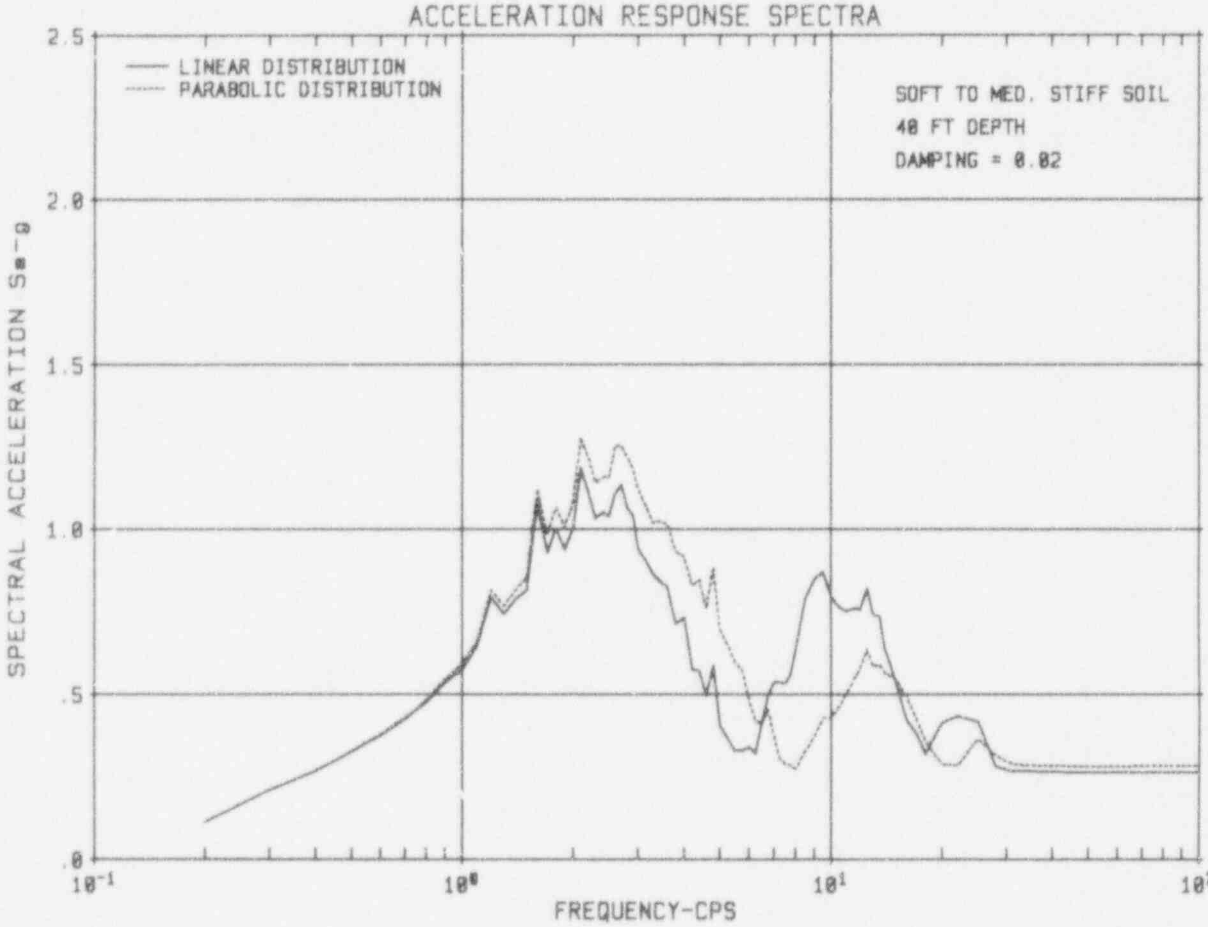


Figure 231.26-3
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

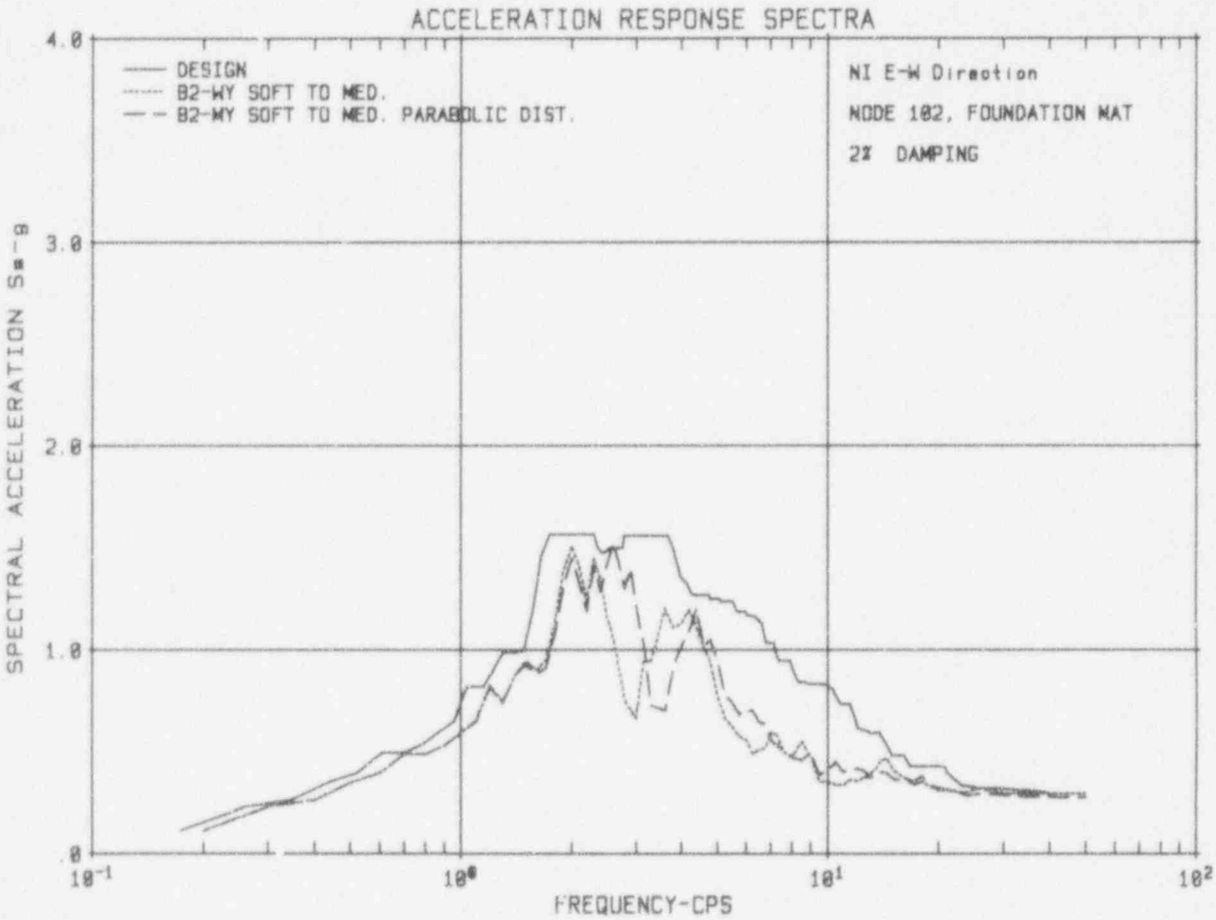


Figure 231.26-4
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

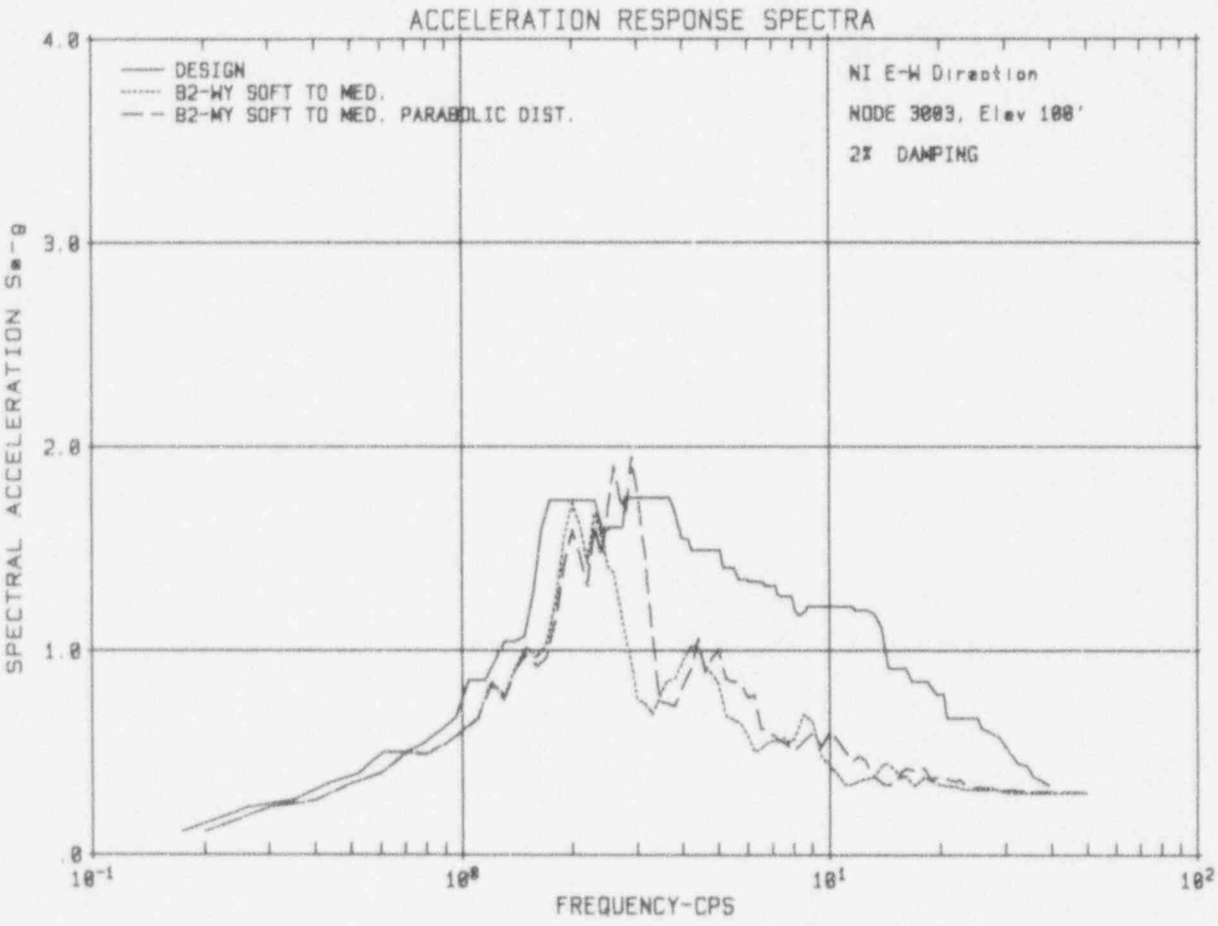


Figure 231.26-5
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

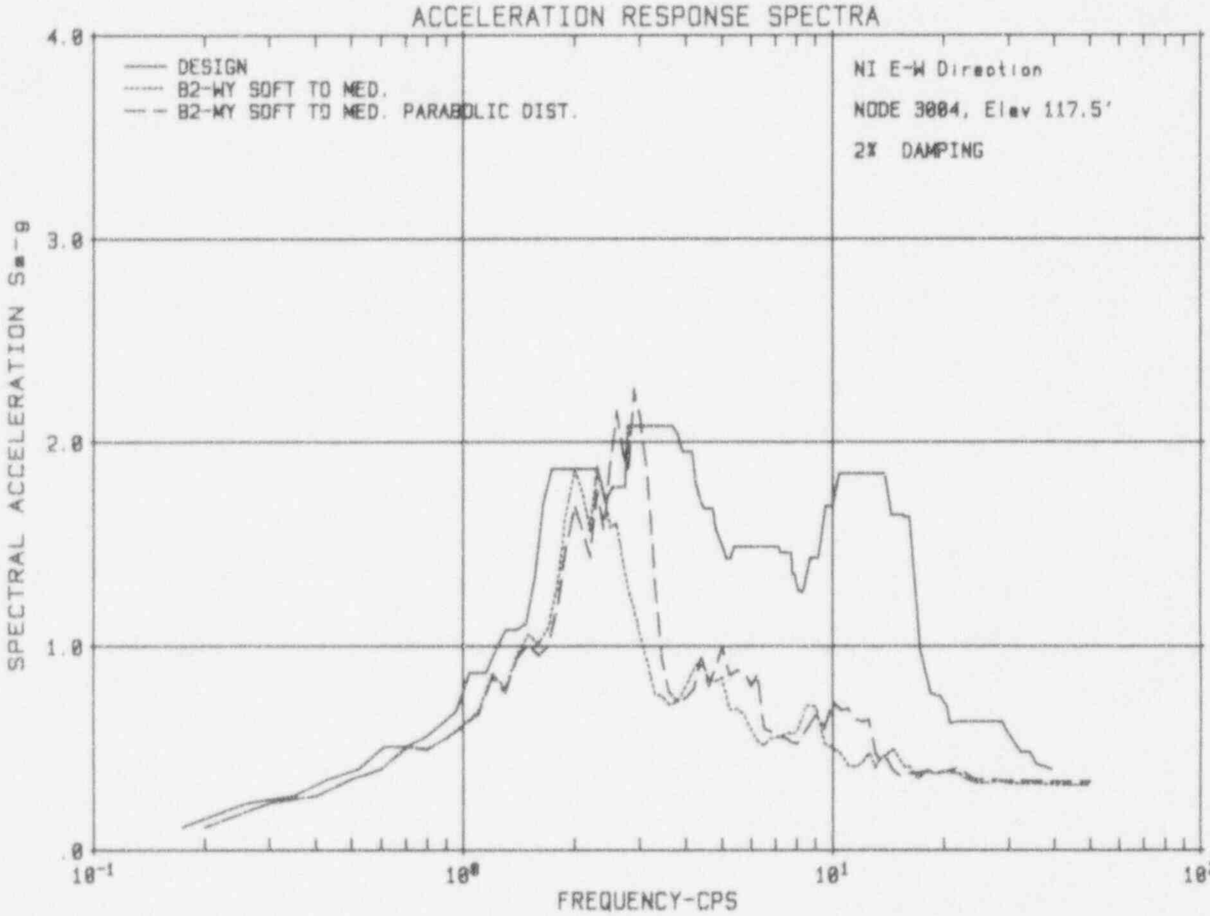


Figure 231.26-6
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

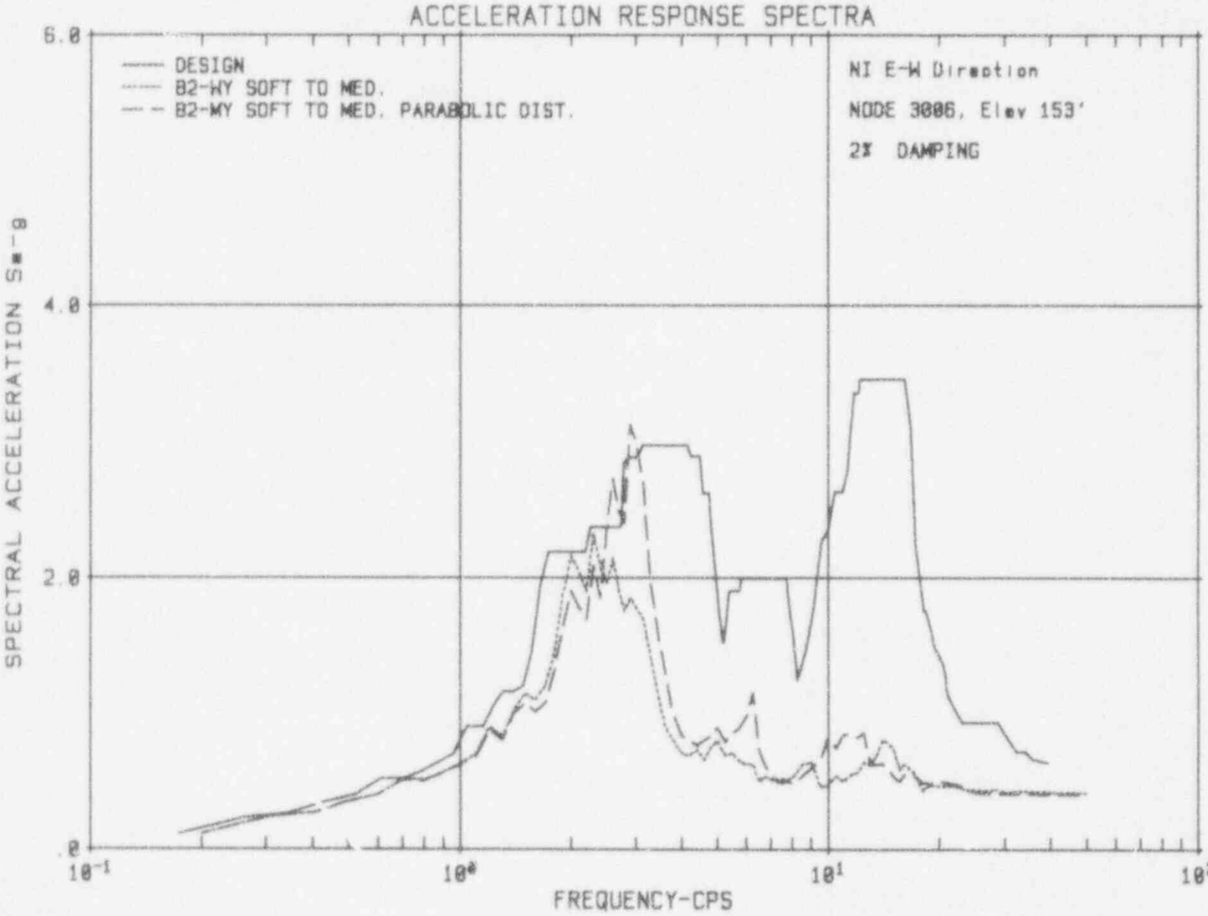


Figure 231.26-7
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

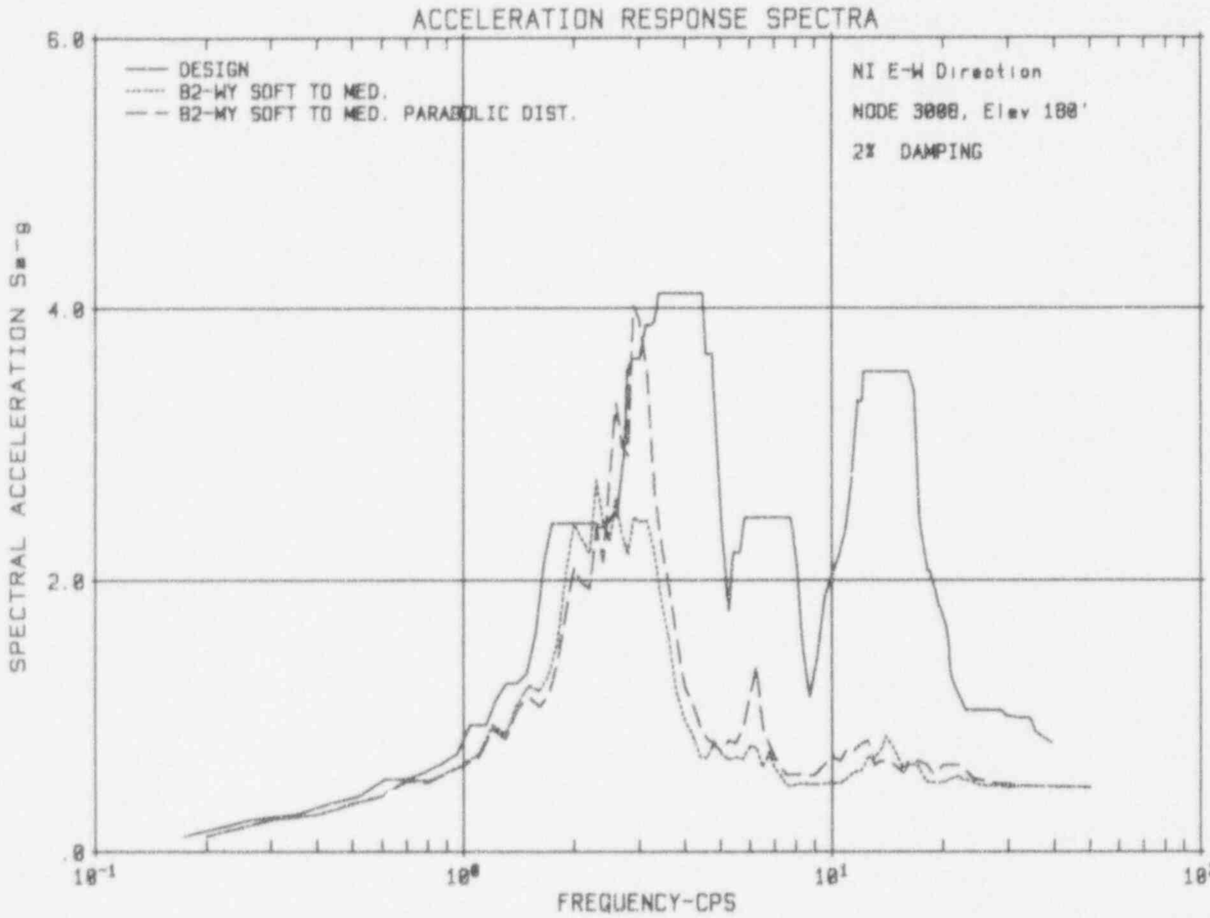


Figure 231.26-8
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

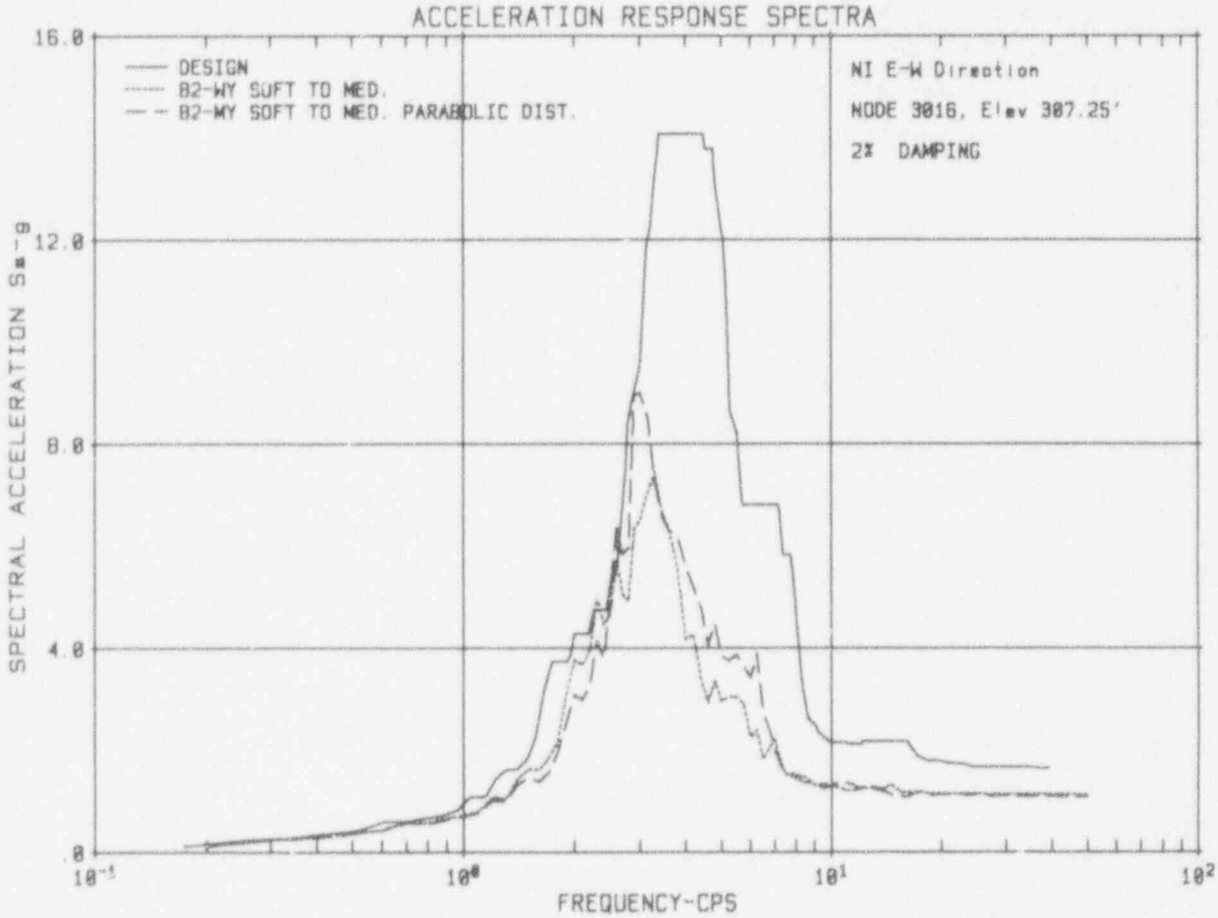


Figure 231.26-9
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

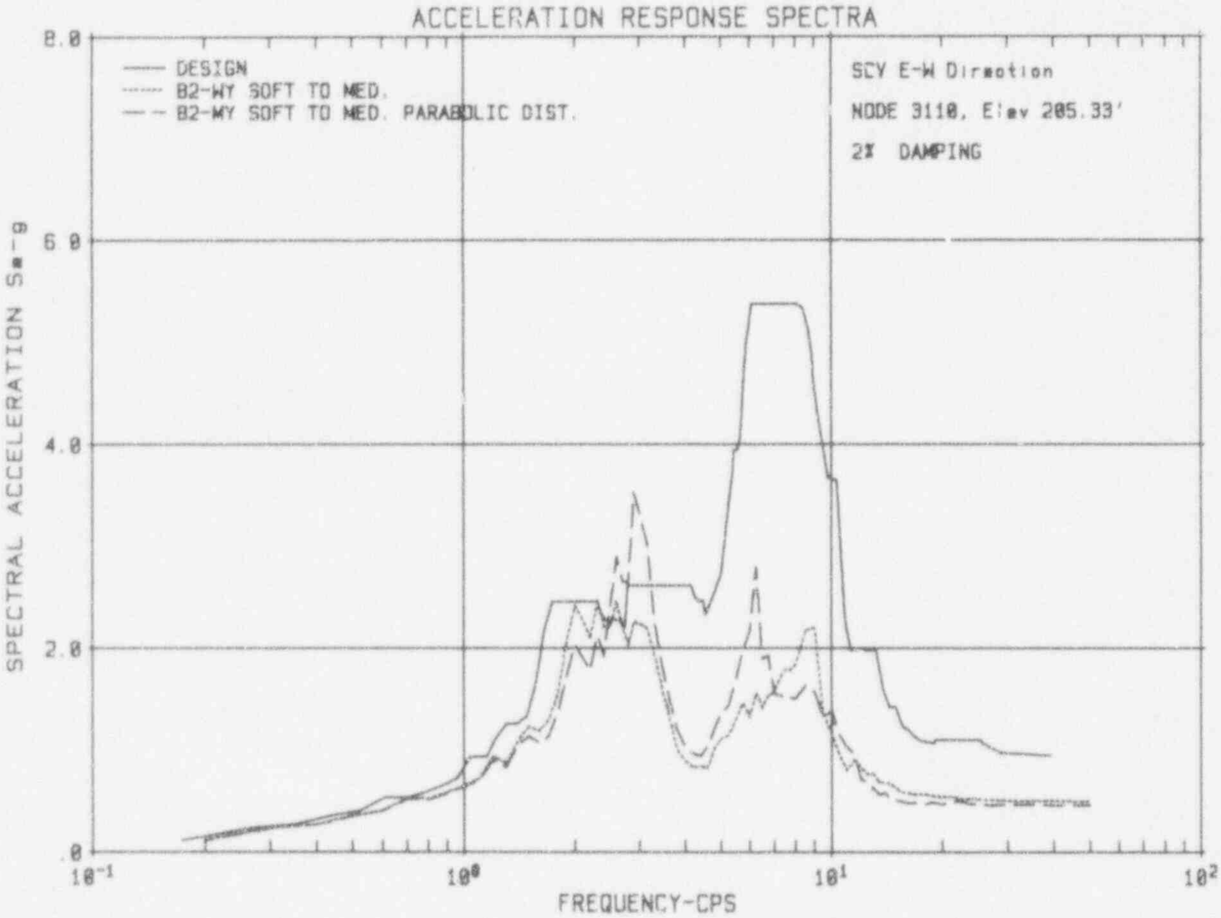


Figure 231.26-10
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

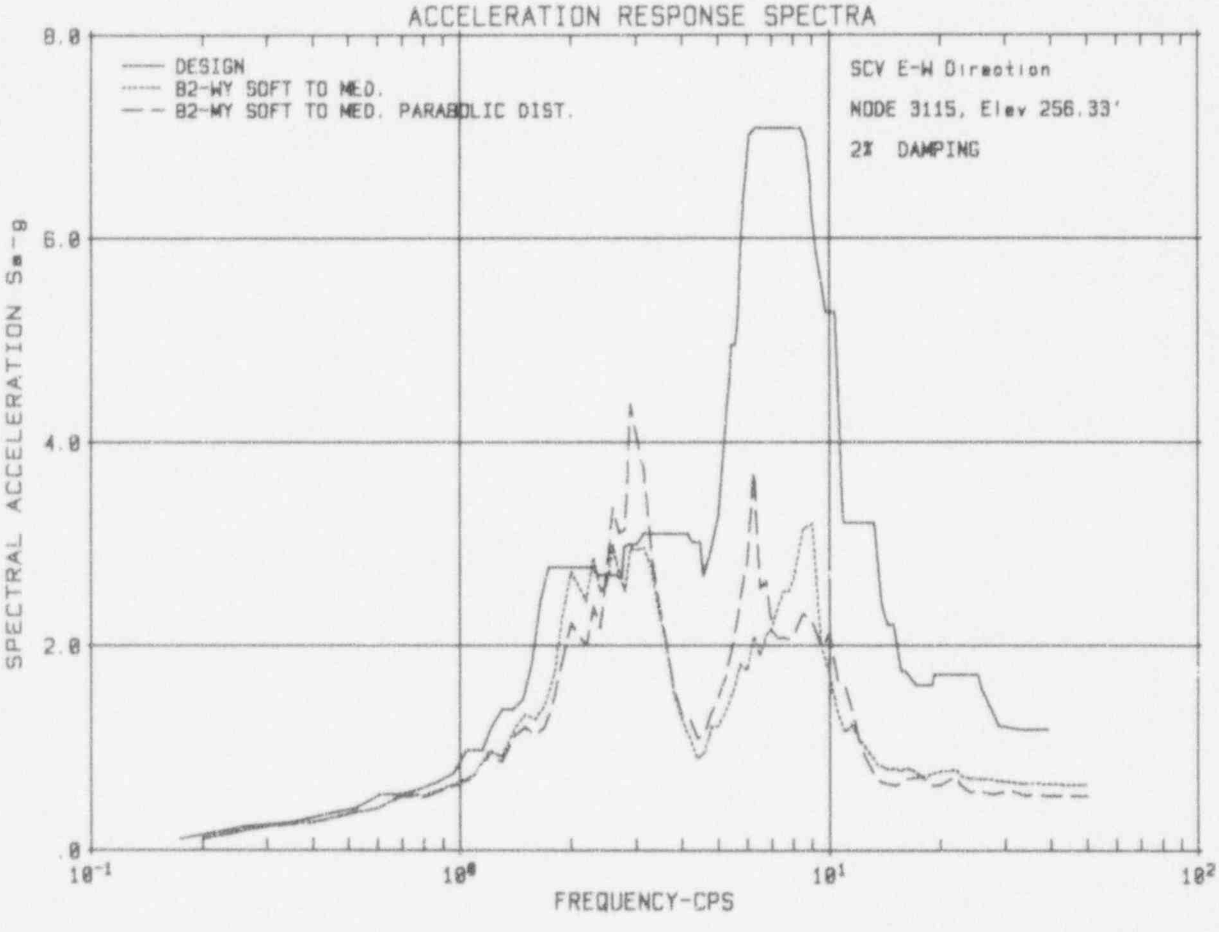


Figure 231.26-11
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

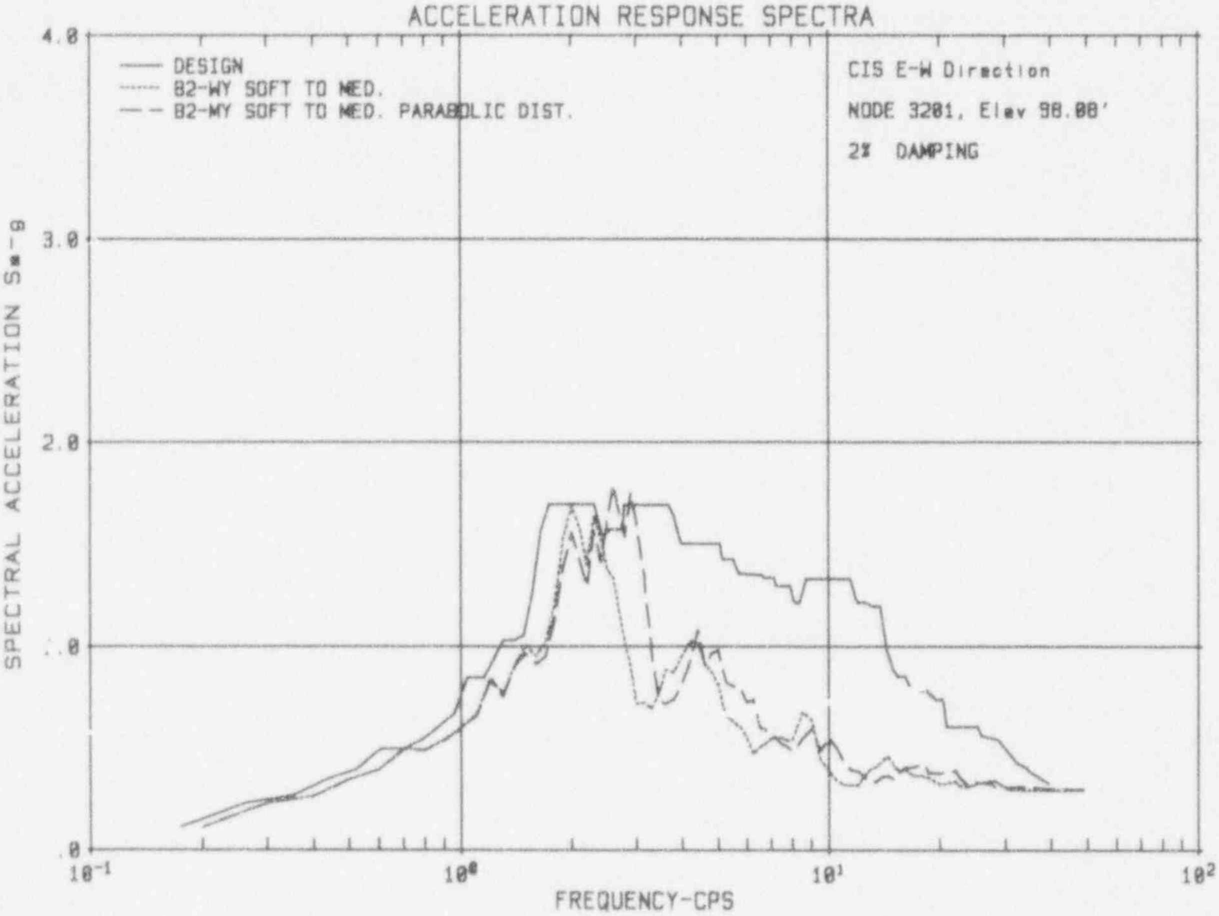


Figure 231.26-12
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation

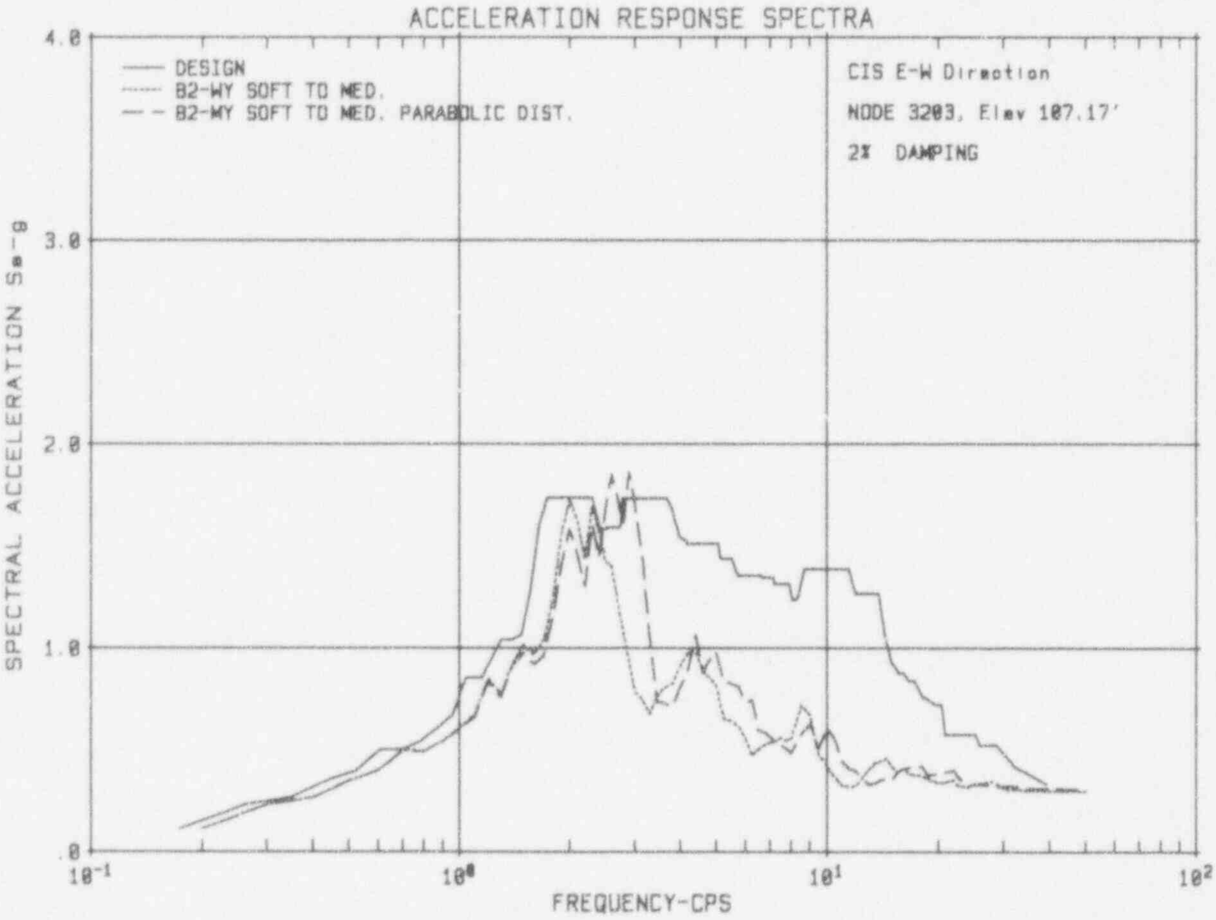
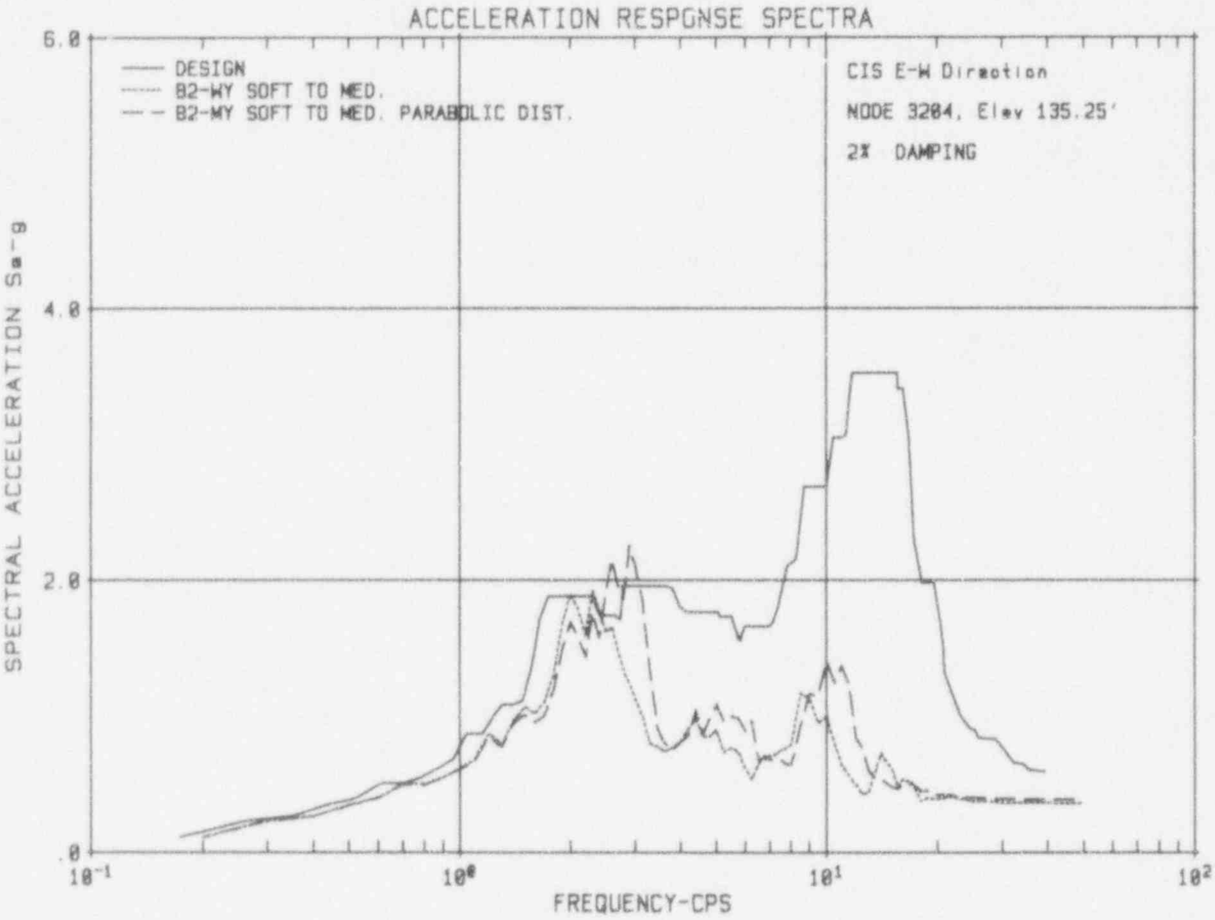


Figure 231.26-13
2D SASSI Analysis, E-W Direction
Parabolic Soil Profile Evaluation



Question 281.22

In the description of elemental iodine removal by deposition on containment walls (Section 15.6.5.3.2.1 of the SSAR), a limiting decontamination factor of 200 was postulated. What is the technical basis for specifying this value?

Response:

As stated in the response to RAI 470.1, the value of 200 for the limiting decontamination factor (DF) for elemental iodine removal was selected to be consistent with Section 6.5.2 of the SRP. While this DF limit of 200 is specified by the SRP as applying to the retention of elemental iodine in the sump solution, its use for defining a limit to the deposition removal of elemental iodine is appropriate since it is expected that a portion of the iodine that has deposited onto surfaces would be washed off into the containment sump as the result of water condensing on the surfaces. It is conservatively assumed that all of the deposited iodine is carried to the sump.

Based on a post-LOCA sump solution volume of 78,900 cubic feet, a containment free volume of 1.73E6 cubic feet, a sump solution pH of 7.0 and iodine partition coefficients from Figure 33 of Reference 281.22-1, there is sufficient retention capacity in the sump solution to support the DF of 200 for elemental iodine.

Reference:

281.22-1 NUREG/CR-2900, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," Oct. 1978, A. K. Postma, R. R. Sherry, P. S. Tam

SSAR Revision: NONE

Question 281.23

In the description of particulate iodine removal by sedimentation and deposition (Section 15.6.5.3.2.2 of the SSAR), three different values for removal coefficient were postulated for three different time periods: 0.35 hr^{-1} for 0 - 5 hrs, 1.3 hr^{-1} for 5 - 5.5 hrs, and 0.5 hr^{-1} for >5.5 hrs and the limiting decontamination factor was assumed to be 1000. What is the technical basis for postulating these values? Also, what were the specific mathematical models or computer codes that were used in calculating them?

Response:

Reference 281.23-1 provides the basis for the determination of post-LOCA particulate removal coefficients. This document was transmitted by EPRI to the NRC on April 30, 1993. The document reports the following removal coefficients for the AP600:

0 - 10.3 hours	0.49 hr^{-1}
10.3 - 11.0 hours	0.72 hr^{-1}
> 11.0 hours	0.52 hr^{-1}

Westinghouse has conservatively simplified these values to 0.5 hr^{-1} for all time periods. This value supersedes the removal coefficients that were used in the LOCA dose analysis reported in the SSAR (0.35 hr^{-1} for 0 - 5 hrs, 1.3 hr^{-1} for 5 - 5.5 hrs, and 0.5 hr^{-1} for >5.5 hrs). Implementation of the 0.5 hr^{-1} removal coefficient for all time periods results in a slight reduction in the calculated thyroid doses both at the Site Boundary (0 - 2 hour dose) and at the low population zone (LPZ) boundary (0 - 30 day dose); thus, the doses currently reported in the SSAR remain bounding. The SSAR will be updated in Revision 2 to reflect the change in particulate removal coefficients.

In Revision 1 to the response to RAI 470.9, an analysis has been provided of the LOCA doses based on the draft NRC source term (June 1992 draft of NUREG-1465). This analysis utilized the revised particulate removal coefficient of 0.5 hr^{-1} for all time periods and is thus consistent with the set of values presented in Reference 281.23-1.

The selection of a limiting decontamination factor (DF) of 1000 for particulates removal was made to assure a conservative determination of activity releases during the later stages of the accident. The particulate removal is expected to continue until virtually all particulates are removed from the containment atmosphere and, with the high humidity environment, there would be little chance of resuspension of particulates. The use of a DF limit of 1000 for particulates is a more conservative approach than past practice of taking unlimited removal credit for particulates (e.g., paragraph III.4.d of Section 6.5.2 of the SRP).

Question 281.23

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Reference:

281.23-1 "Passive ALWR Containment Natural Aerosol Removal," April 29, 1993, prepared by David E. Leaver (Polestar Applied Technology, Inc.), Jun Li (TENERA, L.P.), and Rudolph Sher (Rudolph Sher Associates).

SSAR Revision: NONE

Question 440.253

Provide a drawing of the PRHR illustrating the configuration and dimensions of the baffle surrounding each tube bundle, including the dimensions of the flow openings. The figure attached to letter NTD-NRC-94-4108, dated April 29, 1994, does not show the configuration of the plate above the tube bundles or of the outward-facing plate, and does not provide dimensions for the flow openings. (If this drawing is to scale, the scale is not legible on our copy of the drawing, and a full-sized print is needed.) The figure also appears to show that the highest opening for flow in the baffles is below the upper horizontal section of the PRHR tubes; is that the case? If not, clarify the design of the side baffles.

Response:

The figure attached to letter NTD-NRC-94-4108 showed an early design of the passive heat removal system heat exchanger. The current design has an open type frame as opposed to the sheet metal housing with the cutouts as shown in the previous figure. A figure showing the design of the passive heat removal system heat exchanger is included with the response for RAI 210.039.

SSAR Revision:

See the SSAR Revision provided with the response to RAI 210.39.

Question 471.23

Verify that the airborne radiation monitors described in Section 11.5.2.3.2 of Chapter 11 of the SSAR will be sensitive enough to detect 10 DAC-hrs in any area of the plant that can be accessed by plant personnel.

Response:

Ten (10) airborne process radiation monitors are described in Section 11.5.2.3.2. These radiation monitors are part of the permanently installed AP600 radiation monitoring system and provide general area monitoring. These radiation monitors are supplemented by local portable continuous air monitors (CAMs). CAM use is directed by the Health Physics staff during maintenance operations with a high potential for airborne radioactivity levels.

Of the ten radiation monitors mentioned above, nine (9) monitor selected areas of the plant that can be accessed by plant personnel. These selected areas are as follows:

1) Fuel Handling Area
2) Auxiliary Building
3) Annex Building
4) Main Control Room and Technical Support Center
5) Containment
6) Health Physics and Hot Machine Shop
7) Radwaste Building

The tenth airborne process monitor (Gaseous Radwaste Discharge Radiation Monitor) is strictly a process monitor and measures the concentration of radioactive materials in the release from the gaseous radwaste system to the plant vent. Therefore, this monitor will not be discussed in terms of its ability to detect 10 DAC-hours.

Areas 1, 2, 3, 6, and 7 are monitored by measuring the concentration of radioactive materials in the exhaust air from each area.

Area 4 is monitored by measuring the concentration of radioactive materials in the supply air.

Area 5 is monitored by three separate airborne process monitors:

1) The Containment Air Filtration Exhaust Radiation Monitors measure the concentration of radioactive materials in the containment purge exhaust air.

2) The Containment High Range Radiation Monitors measure the radiation from the radioactive gases in the containment atmosphere.

3) The Containment Atmosphere Radiation Monitor measures the concentration of radioactive airborne particulate, iodine, and gaseous contamination inside the containment.

The Containment High Range Monitors are strictly used to monitor post-accident conditions inside containment. Therefore, these monitors will not be discussed in terms of their ability to detect 10 DAC-hours.

The remaining eight (8) monitors are sensitive enough to detect 10 DAC-hours as discussed below.

SSAR Table 11.5-1 provides a listing of each detector, the principal isotope(s) it monitors, and the detector's nominal range. The lower value of the detector's nominal range corresponds to the detector's minimum detectable level. These minimum detectable levels are achieved with a 95% confidence level at standard operating conditions. The following table summarizes Table 11.5-1 and includes the DAC occupational values from Table 1, Column 3, of Appendix B (Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage) of 10 CFR 20.

Airborne Process Radiation Monitor	Isotope(s)	Detector Minimum Detectable Level	DAC Occupational Values - 10 CFR 20, Appendix B, Table 1
Fuel Handling Area Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Auxiliary Building Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Annex Building Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
MCR Supply Air Duct Particulate	Sr-90 Cs-137	1.0E-13 μCi/cc 1.0E-13 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc
MCR Supply Air Duct Iodine	I-131	1.0E-12 μCi/cc	2.0E-8 μCi/cc
MCR Supply Air Duct Gas	Kr-85 Xe-133	1.0E-7 μCi/cc 1.0E-7 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Containment Air Filtration Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Health Physics and Hot Machine Shop Exhaust	Sr-90 Cs-137	1.0E-13 μCi/cc 1.0E-13 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc
Radwaste Building Exhaust	Sr-90 Cs-137	1.0E-13 μCi/cc 1.0E-13 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc
Containment Atmosphere Particulate	Sr-90 Cs-137	1.0E-13 μCi/cc 1.0E-13 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc

Airborne Process Radiation Monitor	Isotope(s)	Detector Minimum Detectable Level	DAC Occupational Values - 10 CFR 20, Appendix B, Table 1
Containment Atmosphere Iodine	I-131	1.0E-12 μCi/cc	2.0E-8 μCi/cc
Containment Atmosphere Gas	Kr-85 Xe-133	1.0E-7 μCi/cc 1.0E-7 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc

The above table shows that for each principal isotope, the minimum detectable level for each monitors detector(s) is two (2) to almost five (5) orders of magnitude below the corresponding 10 CFR 20 DAC occupational value.

These radiation monitors utilize two basic types of detectors, as described in Section 11.5.2.3.2. The particulate (Sr-90/Cs-137) and iodine detectors use shielded fixed filters, located in the sample stream, that are viewed by beta and gamma sensitive scintillators, respectively. The radiogas detectors use beta sensitive scintillators with their sensitive volumes directly exposed to the process or sample stream.

The response time for each fixed filter detector depends upon background radiation levels, airborne radioactivity levels, sample flow rate, and system configuration. When the detectors have achieved statistically accurate operating conditions, the detector response times are as follows:

1) Step change in radioactivity levels above the ALERT setpoint - < 4 seconds, not including sample transport time.

2) Gradually increasing radioactivity levels above the ALERT setpoint - < 2 seconds, not including sample transport time.

The step change requires a longer response time to assure that the change is not a spurious radioactivity spike. The time to achieve statistical accuracy (95% confidence level) can vary from ten minutes to one hour, depending upon radioactivity concentrations. The only time the detectors will not be operating under statistically accurate conditions will be the time following a filter change or a system shutdown for maintenance. Sample transport times are minimized by locating the detectors as close as practicable to the process sample point.

The response time for the in-line detectors is less than ten seconds. These detectors are provided with dynamic background radiation compensation.

Combining the minimum detectable levels shown in the table above with the detector response times discussed above, it has been shown that each monitor is sensitive enough to detect 10 DAC-hours.

SSAR Revision: NONE

Question 471.25

Section 11.5.2.1 of Chapter 11 of the SSAR states that the radiation monitoring system safety-related channels are powered from the Class 1E dc and UPS system. Further, nonsafety-related channels are powered from non-Class 1E dc and UPS system. Describe and justify which radiation monitors are safety-related.

Response:

Section 11.5.1.1 of Chapter 11 of the SSAR describes the radiation monitoring channels that are safety-related. These are the Main Control Room Supply Air Duct Radiation Monitors and the Containment High Range Radiation Monitors. The respective subsections of Section 11.5.2.3.2 of Chapter 11 of the SSAR state that the local radiation processors for each radiation monitoring channel receive Class 1E power.

The Main Control Room Supply Air Duct Radiation Monitors are safety-related to support the safety related requirements of an engineered safety features actuation signal and to support the safety-related isolation of the MCR envelope as described in SSAR Sections 7.3 and 9.4.

The Containment High Range Radiation Monitors are safety-related to support the safety related requirements of an engineered safety features actuation signal and the safety-related qualification requirements as a post accident monitoring variable as described in SSAR Sections 7.3 and 7.5.

The remaining radiation monitoring channels are nonsafety-related. As stated in Section 11.5.1.2, these monitoring channels are designed to support the requirements of 10 CFR 20. These monitoring channels do not perform any of the safety-related functions defined in Section 3.2.2.1.

SSAR Revision: NONE

Question 471.26

Section 11.5.6 of Chapter 11 of the SSAR states that area radiation monitors will have a local readout and alarm module located in each area so that it is visible to the operating personnel before entering the monitored area. Describe what is meant by local alarm module (i.e., is this a local audible and/or visual alarm, is a flashing light used in high noise areas). Justify your answer.

Response:

Local alarm modules represent both audible and visual devices that alert local personnel to the following:

1) radiation levels exceeding the ALERT setpoint
2) radiation levels exceeding the HIGH setpoint
3) radiation levels rate-of-change exceeding a predetermined setpoint.

The audible alarm is physically located within the local radiation processor. The visual alarms can be located at the local radiation processor or at an area where the alarm can be seen prior to entry into the monitored area, and at a visible location within the monitored area.

These alarms are in accordance with Section 5.5 of ANSI/ANS-HPSSC-6.8.1-1981.

SSAR Revision:

Sections 11.5.6.2 and 11.5.6.3 will be revised as follows to clarify "alarm modules":

11.5.6.2 Post Accident Sample Room Monitor

The post accident sample station is the location where samples are collected and/or analyzed after the postulated accident. The post accident sample room area radiation monitor (detector RMS-JE-RE008) is located so that its readout is representative of the radiation to which the operating personnel are exposed. **A local readout, an audible alarm, and visual alarms are provided in the post accident sample room to alert operating personnel of increasing exposure rates. A local readout, an audible alarm, and visual alarms are provided outside of the post accident sample room and are visible to operating personnel prior to entry.** ~~The radiation data are displayed and stored at the associated local radiation processor and transmitted to the main control room.~~

The monitor detector is a gamma sensitive Geiger-Mueller tube. The monitor is an extended-range monitor. The monitor range and principal isotopes are listed in Table 11.5-2.

11.5.6.3 Normal-Range Area Monitors

Normal-range area radiation monitors are located in accordance with the location criteria given in Subsection 11.5.6.1. A local readout, an audible alarm, and visual alarms are provide in each monitored area to alert operating personnel of increasing exposure rates. Visual alarms are provided outside of each monitored area so that they are visible to operating personnel prior to entry. ~~The radiation data is displayed and stored in the monitor's associated local radiation processor and transmitted to the main control room.~~

The monitor detectors are gamma sensitive Geiger-Mueller tubes. The monitors, their ranges, and principal isotopes are listed in Table 11.5-2.

Question 480.15(R1)

WGOTHIC Validation Using (Integral and Large Scale) Test Data

Provide justification for the use of WGOTHIC to predict AP600 prototype behavior based upon integral and large scale test data, discussion and analysis of the WGOTHIC models, and a rigorous scaling analysis that shows that the range of test parameters (when properly scaled) are representative of conditions that would be expected in the AP600 prototype.

In addition, provide information on the modeling approach and model validation of:

(a) subcooling in WGOTHIC,

(b) condensation models for use inside the containment vessel, and

(c) steam/air/hydrogen stratification.

Provide additional information and justification for the use of WGOTHIC in a lumped-parameter mode to predict hydrogen stratification. The staff believes that any lumped-parameter code will tend to underestimate the concentration of hydrogen in the containment dome and overestimate the amount of mixing. These effects are nonconservative. Therefore, justify the exclusive use of a lumped-parameter approach (see Q480.32).

Alternatively, Westinghouse may provide finite-difference-based WGOTHIC calculations to predict hydrogen stratification and a justification for why the finite-difference based calculations accurately predict hydrogen distributions (Chapter 14).

Response: (Revision 1)

~~WGOTHIC validation is not limited to "integral and large scale test data" sources.~~ WGOTHIC validation is based upon several separate effects tests (**AP600 University of Wisconsin Condensation Tests, AP600 Heated Flat Plate Tests**), as well as the integral and large scale tests. and other valid data sources necessary for validation (**dry free convection tests using selected literature sources, such as Hugot, Siegel and Norris, Eckert and Diaguila, etc**).

WGOTHIC contains a set of ~~additional~~ **boundary layer heat and mass transfer** options which Westinghouse added to the GOTHIC code. ~~The Westinghouse modifications are described in WCAP-13246, and~~ The GOTHIC equations and models are discussed in Reference 480.15-1. ~~Numerical Applications, Inc. document "GOTHIC Containment Analysis Package Technical Manual", Version 3.4, July 1991. is included in Appendix A of WCAP-13246, pages A-1 to A-195.~~ **Enhancements to the WGOTHIC heat and mass transfer models presented in WCAP-13246, Revision 0 (Reference 480.15-2) have been made. The code enhancements meet the objective of developing an accurate tool for assessing the passive containment cooling system (PCS) performance of the AP600. The revised heat and mass transfer correlations and their validation with separate effects tests are given in References 480.15-3 and 480.15-4, respectively. A liquid film energy transport model to account for the subcooled film has**

also been incorporated into the code (Reference 480.15-5). As demonstrated in the "AP600 Passive Containment Cooling System Design Basis Analysis Model and Margin Assessment" Report (Reference 480.15-6), a net increase in margin is shown due to model enhancements made since SSAR Rev. 0.

~~An updated scaling analysis and recent revisions to W-GOTHIC will be presented in Revision 1 of WCAP-13246. The schedule for Revision 1 of WCAP-13246 is presented in Reference 1.~~ The AP600 PCS scaling analysis has been submitted to the NRC in two iterations (References 480.15-7 and 480.15-8). The scaling analysis demonstrates the adequacy of the test program to validate the significant phenomena affecting containment heat removal over the range of conditions expected for design basis analysis.

(a) Enthalpy transport by a subcooled liquid film is ~~only expected to be~~ significant only over a small fraction of the dome surface area in a full-scale plant. Neglecting this in the AP600 analysis produces a small, ~~acceptable~~ conservative under prediction of containment heat removal. The large scale test has somewhat more heat removal by subcooling (~20%) compared to the AP600 (~5%). The addition of the liquid film energy transport model to WGOTHIC allows analytical modeling of this effect in both the test and full scale plant. The test cases reported in WCAP-13246 were modeled by forcing the model with the measured heat flux on the dome, so that the code calculation of noncondensable distributions and overall heat transfer could be validated. ~~A subcooling model is currently under development for W-GOTHIC to more conveniently validate the code with data from wetted tests.~~ Since the liquid film energy transport model has been incorporated into the code, the method used in WCAP-13246, Revision 0, will no longer be applied to the large scale test analysis. The net effect of the revised heat and mass transfer models and the energy transport model on the large scale test WGOTHIC predictions of vessel pressure and temperature was reported in Reference 480.15-6. The improved vessel temperature predictions on the dome, where the liquid film energy transport is most effective, are illustrated in that report.

~~The subcooling model will may be validated using results from the flat plate test, the small scale test, and/or the local dome large scale test data.~~ Additional validation of the liquid film energy transport model will be provided. Validation results will be presented in Revision 1 of WCAP-13246 to be submitted in April 1995.

(b) Validation of condensation models is covered in ~~the response to RAI 480.14.~~ Reference 3. It is shown that condensation over the full range of noncondensables expected in AP600 and covered in separate effects tests is well predicted with only a small bias. Additional validation using large scale test local data will be provided in WCAP-13246 Revision 1 to be submitted in April 1995.

(c) Validation of WGOTHIC lumped parameter mode is given in WCAP-13246 Revision 0. The lumped parameter formulation is shown to over mix the containment, and the net effect on total heat removal through the PCS is shown to be conservative. A significant part of the confirmatory PCS analysis plan is to demonstrate the effects of noncondensable distribution on the ability to predict heat removal capability. It is expected that the results will continue to show that an over mixed prediction gives conservative results for long term heat removal. A discussion of this is provided in Reference 4. The additional validation of predictions for steam/air~~/hydrogen~~ stratification will use results from the large scale Phase 2 and Phase 3 tests. ~~which are currently underway.~~ Results will be presented in Rev 1 of WCAP-13246 to be submitted in April 1995.

A discussion of the hydrogen distribution analysis is given in the Westinghouse response to RAI 480.32(R1) (to be submitted). ~~Hydrogen modeling is required for accidents beyond the design basis. Westinghouse is investigating an alternative modeling approach which subdivides the containment above the operating deck, and retains the lumped-parameter formulation for compartments which do not require finer resolution, and for the downcomer and riser in the external air flow path. Results of hydrogen distribution validation will be presented in Revision 1 of WCAP-13246 according to the schedule presented in Reference 1.~~

References

~~1. Letter, N. J. Liparulo to R. W. Borchardt (NRC), "AP600 Design and Design Certification Test Program Overview", Table 3, Revision 3, August 13, 1993.~~

480.15-1. Letter DCP/NRC0183 N. J. Liparulo to R. W. Borchardt (NRC), "GOTHIC Containment Analysis Package, Version 3.4e", August 10, 1994.

480.15-2. Letter ET-NRC-92-3726 N.J. Liparulo to T. Murley (NRC), "WCAP-13246, Westinghouse-GOTHIC: A Computer Code for Analyses of Thermal Hydraulic Transients for Nuclear Plant Containments and Auxiliary Buildings", July 31, 1992.

480.15-3. Letter DCP/NRC0200 N. J. Liparulo to R. W. Borchardt (NRC), "Experimental Basis for the Convective Heat Transfer Correlations Selected for Modeling Heat Transfer From the AP600 Containment Vessel", August 31, 1994.

480.15-4. Letter DCP/NRC0235 N. J. Liparulo to R. W. Borchardt (NRC), "Experimental Basis for the Mass Transfer Correlations Selected for Modeling Condensation and Evaporation on the AP600 Containment Vessel", October 21, 1994.

480.15-5. Presentation to the U.S. NRC ,"AP600 PCCS Tests and Analysis, Part 2", February 23-24, 1994.

480.15-6. Letter DCP/NRC0111 N. J. Liparulo to R. W. Borchardt (NRC), "AP600 Passive Containment Cooling System Design Basis Analysis Model and Margin Assessment", June 30, 1994.

480.15-7. Letter DCP/NRC0171 N. J. Liparulo to R. W. Borchardt (NRC), "AP600 Integrated Structure for Technical Issue Resolution (ISTIR) for Passive Containment Cooling System", July 28, 1994.

480.15-8. Letter DCP/NRC0227 N. J. Liparulo to R. W. Borchardt (NRC), "Scaling Analysis for AP600 Passive Cooling System", October 27, 1994.

SSAR Revision: None
PRA Revision: None

Question 480.32

Hydrogen Control - Prediction of Hydrogen Distribution

Provide a discussion of the methods used to predict the hydrogen distribution within the containment. In particular, the SSAR appears to rely on the WGOTHIC code used in a lumped-parameter mode. Lumped-parameter codes are notorious for overpredicting mixing, and underpredicting the stratification of hydrogen in the containment dome. This is non-conservative. Why is WGOTHIC (used in the lumped-parameter mode) any different? Have any other experiments or calculations been performed to confirm Westinghouse's ability to predict hydrogen distributions in the AP600 containment (WCAP 13388)? (see Q480.15)

Response:

~~The modelling used in the hydrogen mixing analysis was replicated in the analysis of the Large Scale Tests. The results of the test comparisons demonstrated that the noding scheme, employing lumped parameter nodes, was capable of predicting the containment vessel wall temperature distributions observed in the tests. This agreement, as concluded in WCAP 13246, demonstrated that the code using the subject noding scheme was capable of predicting the stratification of steam and noncondensables inside the vessel. This supported the use of the lumped parameter noding scheme for the hydrogen mixing analyses.~~

~~Lumped parameter noding can lead to the over prediction of mixing and thus the under prediction of the stratification of the noncondensables. In the AP600 calculations, this is ameliorated by the detailed noding along the inner wall of the containment vessel. This location was modelled with 28 relatively small nodes, two feet deep, and placed in four, symmetrical, seven node stacks around the inside of containment above the operating deck. This places relatively small nodes along the primary region of condensation inside the AP600 containment.~~

~~Had the hydrogen concentrations inside these nodes, or any other nodes within the AP600 model deviated significantly from the average hydrogen concentration, then these nodes could have been further subdivided. This was not the case, except for the volume representing the in containment refueling water storage tank (IRWST) which was subdivided. The subdivision of the IRWST was required to avoid uniform heating of the tank and over estimating steam condensation on the water surface.~~

~~Noncondensable mixing tests have been performed with the Large Scale Test facility. The test data will be analyzed using the WGOTHIC code (see RAI 480.15). The AP600 hydrogen mixing analysis will be revised if the analysis of these tests requires any modifications to the methodology.~~

Hydrogen combustion and distribution analyses which address 10 CFR 50.34(f) are not presented in the SSAR, but are presented in the PRA Report. Hydrogen burning analyses using the MAAP4 code are presented in Chapter 14 and Appendix N. Hydrogen mixing analyses using the WGOTHIC code are presented in Chapter 15 and Appendix O. These analyses use lumped-parameter nodalization and predict a high degree of mixing in the AP600 containment.

In Appendix R of revision 1 of the PRA, Hydrogen mixing and burning is also discussed in detail in the deflagration and detonation decomposition event tree (DET) analyses which support of the quantification of the containment event tree. The decomposition event tree analyses present mixing and burning results for in-vessel hydrogen generation consequential to 46 to 118 percent oxidation of the active zirconium cladding in the core. The Appendix R mixing analyses were performed with the version of the MAAP4 code containing sub-nodal physics which model stratification in the containment. The MAAP4 sub-nodal physics model has been benchmarked with good results against the large-scale HDR tests E11.2 and T31.5 (ref. 480.32(R1)-1). The AP600 MAAP4 containment model employs eight nodes inside the pressure boundary, and stratification of hydrogen was observed in the analyses.

Based on the Appendix R analyses, the overall results in Chapters 14 and 15 are still considered to be valid. As predicted in the Chapter 14 analysis, the peak pressures predicted in Appendix R for deflagrations of hydrogen inventories up to 100 percent oxidation of the active cladding do not exceed service level C stress intensity limits. As in the Chapter 15 analyses, the peak hydrogen concentrations are predicted in the IRWST gas space, and hydrogen does not accumulate unexpectedly in containment compartments.

Reference

480.32(R1)-1 EPRI Research Project 3131-02, MAAP4 - Modular Accident Analysis Program for LWR Power Plants, Volume 3, May 1994.

SSAR Revision: NONE
PRA Revision: NONE

Question 952.101

Provide a commitment to submit calculations of PCCS interior velocities using the WGOTHIC code.

Response:

Westinghouse will include velocity field information predicted by WGOTHIC in WCAP-13246, Revision 1, to be submitted in April 1995. Comparisons to large scale test velocity measurements and the influence of velocity predictions on the ability to calculate passive containment cooling system heat removal will be discussed.

SSAR Revision: NONE

Question 952.103

Provide a commitment to submit an analysis demonstrating the validity of the (cold wall) water distribution tests to predict percent film coverage when the wall is hot.

Response:

The analysis has been submitted in the water distribution report and its supplement, References 952.103-1 and 952.103-2 respectively.

References:

952.103-1 Letter DCP/NRC0172 N.J. Liparulo to R.W. Borchardt (NRC), Westinghouse AP600 Letter Report, "Method for Determining Film Flow Coverage for the AP600 Passive Containment Cooling System," July, 1994.

952.103-2 Letter DCP/NRC0198 N.J. Liparulo to R.W. Borchardt (NRC), "Supplemental Information on AP600 PCS Film Flow Coverage Methodology", August 1994.

SSAR Revision: NONE

Question 952.104

Provide a commitment to submit information on how the water distribution model findings support the design-basis accident analyses.

Response:

The analysis has been submitted in the water distribution report and its supplement, References 952.104-1 and 952.104-2 respectively.

References:

952.104-1 Letter DCP/NRC0172 N.J. Liparulo to R.W. Borchardt (NRC), Westinghouse AP600 Letter Report, "Method for Determining Film Flow Coverage for the AP600 Passive Containment Cooling System," July, 1994.

952.104-2 Letter DCP/NRC0198 N.J. Liparulo to R.W. Borchardt (NRC), "Supplemental Information on AP600 PCS Film Flow Coverage Methodology", August 1994.

SSAR Revision: NONE