Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5187
DCP/NRC0917
Docket No.: STN-52-003

June 16, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: RAI RESPONSES RELATED TO AP600 CERTIFIED DESIGN MATERIAL

Dear Mr. Quay:

Attached are responses to requests for additional information 640.1 through 640.58. The requests for additional information are related to the certified design material and were transmitted in NRC letters dated March 4 and April 18, 1997. The April 18 letter contained one additional RAI, 640.59, which is being addressed separately. That response is being sent today, with a package of other RAIs related to PXS intake screens.

The SSAR revisions included in the responses will be included in SSAR Revision 13 and 14. This transmittal completes the Westinghouse actions for these items except for the inclusion of the SSAR changes in a formal SSAR Revision.

Please contact R. Schreiber at (412) 374-5356 if you have any questions.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: D. T. Jackson, NRC (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

1/1 E004

Question 640.1

It is our understanding that the certified design material (Tier 1 material) should be a subset of the Tier 2 information. Therefore, the content of the Tier 1 material should be consistent with what is in the Tier 2 material. However, a number of inconsistencies were identified between the Tier 1 and Tier 2 materials (see specific comments below). In addition, cross references need also to be provided in the certified design material document.

Response:

We have instituted a programmatic review of the ITAAC against the SSAR to ensure that Tier 1 is consistent with Tier 2 and address each of the specific inconsistencies identified by the staff. In addition, we are preparing a tabular cross-reference for inclusion in SSAR Section 14.3.

SSAR Revision:

See the individual RAI responses dealing with inconsistencies identified by the staff.

Question 640.2

During the staff's review of the ABWR standard design, the staff, GE and industry representatives including NEI expended a significant effort developing ITAAC and Tier 1 format that were mutually agreeable. CE followed a similar approach and completed the ITAAC for the System 80+ standard plant design with minimal differences. The approach that Westinghouse is taking reopens many issues that were mutually resolved with the industry and will result in a significant waste of effort and resources for the staff and Westinghouse.

Response:

The staff and industry representatives developed the format for the evolutionary plant submittals, and we have used it where possible. But there are a number of fundamental differences between the AP600 and the evolutionary plants which force a different approach in certain areas. For example, the passive safety design of the AP600 results in clearer segregation of safety and nonsafety systems. As a result, we have fewer, more concise ITAAC, and somewhat less detail on systems that perform purely nonsafety functions. In addition, because the passively safe design reduces reliance on the plant operators, we have somewhat less emphasis on instrument displays, alarms, and habitability outside the main control room.

Another fundamental difference between the AP600 and the evolutionary plants is the level of design completion in certain areas, including piping analysis, structural design, and the Design Reliability Assurance Program. Because our level of completion is higher in areas affecting Design Certification, the AP600 ITAAC focuses on physical features of the completed design, rather than on the process used for design.

SSAR Revision:

None.

Question 640.3

As required by 10 CFR 52.97(b)(1), the "ITAAC are to be necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, ..., the Commission's rules and regulations." From its review of Revision 2 of the AP600 Certified Design Material (CDM), the staff found that the level of detail of the material provided in the civil/structural and piping areas is far below that necessary for the staff to use as a basis for making any safety determination.

Response:

In response to specific staff comments (such as RAI 640.7 on structural analysis and 640.19 on piping), Westinghouse has added the requested detail to ITAAC Revision 3.

SSAR Revision:

None.

Question 640.4

Re: Section 3.3 - Nuclear Island Buildings

In order to ensure that the ITAAC can be effectively implemented, the design description needs to be expanded to include (a) the scope, (b) description of all safety related structures, (c) design codes, (d) design loads, and (e) figures to show the configuration of the nuclear island structures including the foundation mat.

Response:

The design description section of ITAAC Section 3.3 has been revised to describe the elements of the buildings (expanded beyond the Nuclear Island (NI) Buildings and the general purpose of each building to address in general items (a) and (b) as shown below.

The NI structures include the containment building, and the shield and auxiliary buildings. The containment building, shield and auxiliary buildings are structurally integrated on a common basemat which is embedded below the finished plant grade level. The containment building is a cylindrical welded steel vessel with elliptical upper and lower heads, supported by embedding a lower segment between the containment internal structures concrete and the basemat concrete. The shield building, in conjunction with the internal structures of the containment building, provides shielding for the RCS and the other radioactive systems and components housed in the containment. The auxiliary building houses the safety-related mechanical and electrical equipment located outside the containment building and shield building.

The annex building houses the personnel access, technical support center, non-1E electrical equipment, and the hot machine shop. The radwaste building houses the low-level waste processing and storage.

The design codes and loads are defined in the structural analyses for the NI structures that are considered to be seismic Category I which address items (c) and (d).

Figures 3.3-1 through 3.3.-15 have been included as a reference to Table 3.3-1 to indicate the configuration of the NI buildings and annex and radwaste buildings. Included are two sections and several plan views at various plant elevation levels.

SSAR Revision:

Add the following after the first sentence in subsection 1.2.3 on page 1.2-16 (rev 11) and add a new Table 1.2-1.

The plant elevation levels for each of the principal structures are defined on Table 1.2-1.

Table 1.2-1
Plant Elevation Levels

Applicable Plant Elevation Level Designation	Containment	Auxiliary Building	Shield Building	Turbine Building	Annex Building	Radwaste Building
0	60′-66″	60′-6"	60′-6″	-	-	-
1	71′-6″	66′-6″	66′-6″	-	-	-
1.1	81′-0″	-	-	-	-	-
1.2	83′-0″	-	-	-	-	-
2	84′-6″	82′-6″	82′-6″	89′-0″	-	-
2.1	96′-6″	92′-6″	-	-	-	-
2.3	98′-0″	94′-3″	-	-	-	-
3	107′-2″	100′-0″	100′-0″	100′-0″	100′-0″ & 107′-2″	100′-0″
3.1	-	107′-2″	-	-	-	-
4	118′-6″	117′-6″	117′-6″	117′-6″	117′-6″	-
5	135′-3″	135′-3″	135′-3″	135′-3″	135′-3″	
6	162′-1″	153′-0″ & 160′-0″	153′-0″	161′-0″	154′-0″ & 156′-0″	-
7	209′-0″	180′-0″ & 160′-6″	180′-0″ & 160′-6″	190′-0″	162′-6 & 166′-0″	
8	-	-	200′-0″	-	-	-
9	-	-	220′-0′	-	-	-

Table 1.2-1 (cont.) Plant Elevation Levels						
Applicable Plant Elevation Level Designation	Containment	Auxiliary Building	Shield Building	Turbine Building	Annex Building	Radwaste Building
10	-	-	236′-0″	-	-	-
11	-	-	241′-0″	-	-	-
12	-	-	246′-0″	-	-	-
12.1	-	-	250′-4″	-	-	-
12.2	-	-	270′-0″	-	-	-
12.3	-	-	283′-10″	-	-	-
13	-	-	288′-10″	-	-	-
13.1	-	-	294′-6″	-	-	-
13.2	-	-	306′-6″	-	-	-
14	-	-	308′-6″	-	-	-

Question 640.5

Re: Section 3.3 - Nuclear Island Buildings

Figures to be provided should include the floor plan at each elevation and cross-sections of structures including key dimensions such as dimension of the foundation mat, thickness of floors and major walls, thickness of foundation mat, embedment depth, etc.

Response:

Certified Design Material Figures 3.3-1 through 3.3-15 have been included as a reference to Table 3.3-1 to indicate the configuration of the Nuclear Island buildings and annex and radwaste buildings. Included are two sections and several plan views at various plant elevation levels. Table 3.3-1 was added to define key dimensions such as thickness of floors and major walls, thickness of foundation mat, and embedment depth.

SSAR Revision:

None.

Question 640.6

Re: Section 3.3 - Nuclear Island Buildings

From the review of Item 1 (nuclear island structures) of Table 3.3-4 (ITAAC), the staff is unable to determine what design commitments are, what structures are to be inspected and/or tested, and what acceptance criteria are. Westinghouse should use the ITAAC for either GE ABWR or ABB/CE System 80+ as an example and redevelop the AP600 ITAAC for all seismic Category I structures and structural elements including the nuclear island foundation mat.

Response:

Item 1 (Nuclear Island structures) has been revised to be more consistent with the NRC suggested example of the ABB/CE System 80+ ITAAC. New commitments have been included as 1.b, 1.c, 1.d, 1.e, and 8. The following are the new commitments:

1.b) The top of the Nuclear Island basemat is located below the design plant level per Table 3.3-1.

1.c) The containment and its penetrations are designed and constructed to ASME Code, Section III, Class MC.

1.d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.

1.e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.

8. The reactor cavity sump has a minimum concrete thicknesses shown on Table 3.3-1 between the bottom of the sump and the steel containment.

Details of these are included in Table 3.3-6 of the revised ITAAC.

SSAR Revision:

None.

Question 640.7

Re: Section 3.3 - Nuclear Island Buildings

In order to ensure that the nuclear island structures as constructed can withstand the structural design basis loads, Westinghouse should commit in the ITAAC that a structural analysis will be performed which reconciles the as-built data with the structural design basis loads specified in the design description.

Response:

Westinghouse has included a commitment to reconcile the as-built data with the structural design specified in the design descriptions as evidenced by commitment 1.a). The NRC and Westinghouse, through various structural audits, have defined a set of structural locations (plans or sections) on the Nuclear Island that are considered to be key to the structural capability to withstand the design basis loads. Concrete thickness and required reinforcement section properties have been defined for these critical sections and included in a new Table 3.3-1 included in the ITAAC (Revision 3). These section data will provide the NRC with as-built structural parameters that reflect the structural design. Also referenced by Table 3.3-1 is a set of plan and section drawings to aid the reviewer in understanding the key locations.

SSAR Revision:

None.

Question 640.8

Re: Section 3.3 - Nuclear Island Buildings

ITAAC should be provided to ensure that the containment vessel and containment penetrations are designed and constructed to ASME Code, Section III, and an analysis report does exist to conclude that the as- built containment vessel and penetrations are able to withstand the design basis loads defined in the design description.

Response:

Westinghouse has included a commitment within the ITAAC to verify that vessel and penetrations are designed and constructed to the ASME code and able to withstand design basis loads. See design commitments 2, 3, 4, and 5 in Table 2.2.1-3 of the AP600 Certified Design Material.

Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2.a) The components identified in Table 2.2.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.2.1-1 as ASME Code Section III.
2.b) The piping identified in Table 2.2.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.2.1-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.2.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3.b) Pressure boundary welds in piping identified in Table 2.2.1-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	i) A hydrostatic or pressure test will be performed on the components required by the ASME Code Section III to be tested.	iii) A report exists and concludes that the results of the pressure test of the components identified in Table 2.2.1-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
	ii) Impact testing will be performed on the containment and pressure-retaining penetration materials in accordance with the ASME Code Section III, Subsection NE, to confirm the fracture toughness of the materials.	iv) A report exists and concludes that the containment and pressure-retaining penetration materials conform with fracture toughness requirements of the ASME Code Section III.
4.b) The piping identified in Table 2.2.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic or pressure test will be performed on the piping required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the piping identified in Table 2.2.1-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.1-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.1.1-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) The as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.

SSAR Revision:

None.

Question 640.9

Re: Section 3.3 - Nuclear Island Buildings

A commitment needs to be made in the ITAAC that the containment vessel and penetrations will maintain the leakage rate less than the maximum allowable leakage rate as required by regulations.

Response:

Westinghouse has included a commitment to verify the integrated leakage from the containment and penetrations will be less than that required by regulation, that is, L_a

Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.	i) A containment integrated leak rate test will be performed. ii) Testing will be performed to demonstrate that remotely operated containment isolation valves close rapidly.	i) The leakage rate from containment for the integrated leak rate test is less than L_a. ii) A report exists and concludes that the containment purge isolation valves close within 5 seconds and all other containment isolation valves close within 60 seconds upon receipt of an actuation signal.

SSAR Revision:

None.

Question 640.10

Re: Section 3.3 - Nuclear Island Buildings

Figures should be provided to show the configuration of the fire water tank. In addition, the ITAAC should commit to perform tests to ensure that no leakage of water from either the PCS tank or the fire waste tank, and to identify any deflection of roof structures during and after the first fill of the tank water.

Response:

A fire water tank has been designed to fit within the concrete bounds of the passive containment cooling system (PCS) water tank and to be physically separated from it. A fire water tank overflow connection is provided to the PCS tank, which allows a flow circulation path and limits the amount of fire water in the tank. The location of the tank has been included in Figures 3.3-1, 3.3-2, and 3.3-10. An ITAAC commitment has been included as defined below to address the leakage from the fire water or PCS tanks.

The deflection of the roof structures during and after the first fill of the tank water has not been included in the ITAAC because a practical means of measuring the roof deflection is not available.

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The shield building roof, PCS storage tank, and the fire water storage tank support and retain the PCS and fire water sources.	Visual inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank and fire water storage tank for significant water leakage (>100 gal/hr as measured by water level change).	The as-built inspection report exists and concludes that the water leakage does not exceed 100 gal/hr.

SSAR Revision:

None.

Question 640.11

Re: Section 3.3 - Nuclear Island Buildings

ITAAC should be provided for the spent fuel pool structure and fuel racks.

Response:

The spent fuel pool structure is an integral part of the overall auxiliary building structure; therefore, a separate ITAAC was not prepared to cover this structure. However, the spent fuel pool walls (concrete thickness) and their associated reinforcement have been included as part of the structural ITAAC defined in Section 3.3, paragraph 1.a). A separate ITAAC has been prepared for the spent fuel racks and included in the fuel handling system (FHS) Section 2.1.1.

SSAR Revision:

None.

Question 640.12

Re: Section 3.3 - Nuclear Island Buildings

ITAAC should also be provided for the construction sequence of the seismic Category I structures including the nuclear island foundation mat, embedded exterior walls, shield building roof structures, etc.

Response:

The construction sequence or construction approach becomes relevant only for soft soil sites having unconsolidated deposits with shear wave velocities in the range from 1,000 to 2,000 feet per second. Existing analysis shows that for any other site exceeding these limits, the construction sequence is independent of the soil conditions and will have no adverse impact on the seismic Category I structures. An ITAAC has been included to address only soft soil sites per the following:

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The construction approach for soft soil sites includes two limits:	A visual inspection of the as-built auxiliary building, shield building, and containment structures will be performed during construction to confirm that one of the two limits were met:	The as-built inspection concludes that the construction limits have not been exceeded.
i.) Shield building construction ahead of auxiliary building or	i.) The north walls of the auxiliary building are completed to elevation level 2 prior to placement of concrete in the shield building above elevation level 2 or in-containment structures above elevation level 2.	

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
ii.) Auxiliary building construction ahead of shield building. **This commitment applies for only soft solid sites having unconsolidated deposits with shear wave velocities in the range from 1,000 to 2,000 feet per second.**	ii.) The concrete was not placed in the auxiliary building above elevation level 4 before the shield building was completed to elevation level 2.	

SSAR Revision:

None.

Question 640.13

Re: Open Item 5071

The third sentence of Page 5.0-1, "For cases where a site characteristic .... does not exceed the capacity of the design," does not belong to the certified design material and should be deleted.

Response:

The third sentence, as identified in this RAI, has been deleted from Section 5.0.

SSAR Revision:

None.

Question 640.14

Re: Open Item 5072

The maximum ground water level and maximum flood level at plant elevation of 100 ft (design grade elevation) as stated in this table are inconsistent with those stated in SSAR Section 2.4 (Revision 8) which stated that the ground water level and flood level are up to the plant elevation. Also, SSAR Section 3.4 (Revision 8) stated that the high water interface is at two feet below the grade elevation. Furthermore, SSAR Table 2-1 (Revision 10) stated that the flood level and ground water level are less than the plant elevation of 100 ft. Clarification for these inconsistencies is needed.

Response:

The AP600 is designed for a maximum normal ground water of plant elevation 98 feet or 2 feet less than the design plant grade of plant elevation 100 feet. The SSAR will be revised as indicated, and Table 5.0-1 (Site Parameters) of the Certified Design Material is revised accordingly.

SSAR Revision:

Section 2.4, first sentence, revise to state:

| The AP600 is designed for a normal ground water elevation up to plant elevation 98 feet and a flood level up to plant elevation 100 feet.

Table 2-1 (Sheet 2 of 2), third entry, revise to state:

| **Ground Water Level** Less than plant elevation ~~100~~98′

Question 640.15

For the tornado wind, the maximum pressure drop in addition to the maximum wind speed should be provided in this table.

Response:

Section 5.0 of the Certified Design Material has been modified to include the maximum tornado pressure differential of 2.0 lb/in^2 in Table 5.0-1.

SSAR Revision:

None.

Question 640.16

Re: Open Item 5074

For the soil bearing strength, the minimum static soil bearing capacity instead of the average bearing reaction due to dead load and the maximum static bearing reaction should be provided.

Response:

The AP600 is designed for an average static bearing reaction due to the dead weight of the Nuclear Island of about 8000 lb/ft^2. The SSAR is revised as indicated below, and Table 5.0-1 (Site Parameters) of the Certified Design Material is revised accordingly.

SSAR Revision:

Section 2.5.4.2, first sentence, revise to state:

The average bearing reaction of the AP600 is about 8,000 pounds per square foot. The minimum average allowable static soil bearing capacity shall be 8,000 ponds per square foot over the footprint of the nuclear island at its excavation depth (see Table 2-1).

Table 2-1 (Sheet 1 of 2), first entry under **Soil**, revise to state:

~~Bearing Strength~~	~~Soils must support the AP600 under specified conditions. The average static bearing reaction due to the dead weight of the AP600 nuclear island is about 8000 pounds/square foot; the maximum static bearing reaction at a corner is about 12,000 pounds per square foot.~~
Average allowable static soil bearing capacity	Greater than or equal to 8,000 pounds per square foot over the footprint of the nuclear island at its excavation depth.

Question 640.17

Re: Open Item 5075

Section 5.0 - Site Parameters

For the soil shear wave velocity, the phrase, "or acceptable comparison of floor response spectra to the certified design based on site-specific soil-structure interaction analyses," should not belong to the certified design material and should be deleted from this table.

Response:

Table 5.0-1 of Section 5.0 of the Certified Design Material is modified to delete: "or acceptable comparison ... interaction analyses" from the entry for soil shear wave velocity.

SSAR Revision:

None.

Question 640.18

Re: Open Item 5076

640.18 For the safe shutdown earthquake, (1) the design ground response spectra as shown in SSAR Figures 3.7.1-1 and 3.7.1-2 should be provided in this section, and (2) the phrase, "SSE free field ground acceleration of 0.3g with Regulatory Guide 1.60 response spectra," should be replaced by, "SSE free field ground acceleration of 0.3g with modified Regulatory Guide 1.60 response spectra."

Response:

Section 5.0 of the Certified Design Material is modified to include the vertical and horizontal design response spectra for safe shutdown earthquake and indicates that these are modified Regulatory Guide 1.60 response spectra.

SSAR Revision:

None.

Question 640.19

In Revision 2 to the AP600 Certified Design Material, the Design Description and ITAAC for piping have been eliminated and placed in the respective system-based design description and ITAAC. The staff's review of the proposed changes finds that the relocation of the certified piping design (Tier 1) commitments and ITAAC to the specific system is not acceptable. Through this change, many technical and policy issues that have been resolved in the previous reviews of the evolutionary plant applications have now been reopened. The following summarizes some of the more significant issues that need to be resolved as a result of the change.

Elimination of the piping design description reopens the policy issue related to level of detail needed for design certification as it pertains to piping system design. The level of detail issue is discussed at length in SECY-90-377 and in the staff requirements memorandum dated February 15, 1991. In resolving this issue, the staff proposed in SECY-92-053 the use of Design Acceptance Criteria (DAC) for piping design. Therein, the staff stated that the DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies in making a final safety determination to support design certification. DAC would have to be sufficiently detailed to provide an adequate basis for the staff to make a final safety determination regarding piping design. The staff further stated that it would specify DAC in the design certification rule (DCR) that would enable the staff to make a final safety determination on all piping issues. The DCR would contain a description of the methodologies, design processes, and acceptance criteria that will be used to complete the design details and verify that the requirements for piping design have been properly implemented.

For ABWR and System 80+, the staff ensured that the piping DAC were sufficiently specified in the Tier 1 design description. The details of the Tier 1 commitments were described in the SSAR as Tier 2* commitments. However, the fundamental design commitments for piping design were included in the Tier 1 design description. Some of the fundamental design commitments for piping included (1) designing the piping to the ASME Boiler and Pressure Vessel Code, Section III to ensure pressure boundary integrity, (2) designing the piping to ensure its functional capability, (3) minimizing the effects of erosion-corrosion, (4) ensuring that equipment nozzle loads are met, (5) benchmarking the piping computer code, (6) ensuring that high-energy line breaks and environmental effects are adequately considered, (7) ensuring that proper materials are used to prevent brittle fracture and reduce the possibility of cracking during service, and (8) ensuring that adequate clearances are provided during construction. All of the above fundamental piping design commitments and more were eliminated in Revision 2 of the AP600 CDM. In order for the staff to reach a final safety determination on the adequacy of the AP600 piping design, these commitments need to be included as Tier 1 commitments.

To minimize staff resources required to review the AP600 CDM report as presently formatted, and Westinghouse resources required to respond to many potential Requests for Additional Information on each applicable system in Section 2.0, the staff believes that the AP600 CDM report should be revised to add a "Piping Design" subsection in Section 3.0 "Non-System Based Design Description and ITAAC." This new subsection should be similar to the "Piping Design" subsections in the two evolutionary plant CDMs.

Response:

As discussed in our response to RAI 640.2, the treatment of piping analysis in the evolutionary plant submittals is not appropriate to AP600, due to our higher level of design completion. We have incorporated the eight fundamental design commitments listed above into Revision 3 of the ITAAC, as discussed at our meeting with the Civil and Structural Branch on April 17 in Monroeville.

SSAR Revision:

None.

Question 640.20

In Revision 2 of the AP600 CDM, the section which was included in the evolutionary plant CDMs under, "General Provisions, Verification for Basic Configuration for Systems," has been eliminated and placed under system-based ITAAC. The staff's review of this change finds the relocation of this information into system-based ITAAC is acceptable. However, in relocating this information to ITAAC, the intent of the verification appears to have been unacceptably changed as noted below.

The verification of the seismic qualification of mechanical and electrical equipment was intended to be an inspection of the type tests, analyses, or combination of type test and analyses to ensure that the as-built equipment including associated anchorages is qualified to withstand design basis dynamic loads without a loss of its safety function. In other words, the ITAAC should be an inspection; not the type tests and analyses themselves. The type tests and analyses are performed by the equipment vendor at the test location not on site using certain anchorages. The ITAAC should be an inspection of the as-installed equipment to ensure that the installed configuration including anchorages is similar to the configuration tested or analyzed by the vendor.

The same comment noted above for equipment seismic qualification also applies to the relocation of the verification of basic configuration for MOVs. The tests or type tests of MOVs is not the ITAAC. Rather, the ITAAC should be an inspection of the tests or type tests for the MOV to ensure that the as-installed MOV has been qualified for the intended function. The inspection should verify that a test report exists that demonstrates that the as-installed MOV is qualified to perform its safety function under design basis differential pressure, system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and/or maximum stroke times.

To ensure adequate welding, the ITAAC may be the NDE inspection required by the ASME Boiler and Pressure Vessel Code, Section III because this is an inspection of the as-installed ASME Code components. In this case, the acceptance criteria should be the ASME Code, Section III acceptance criteria for pressure boundary welds not a report.

The AP600 CDM should be revised to address all of the above staff comments in each applicable system in Section 2.0.

Response:

The requested changes are incorporated in Revision 3 of the AP600 Tier 1 submittal.

SSAR Revision:

None.

Question 640.21

Page 14.3-1, sixth para- it is stated that "The Certified Design Material design descriptions delineate the principal design bases and principal design characteristics that are referenced in the design certification rule." The design description (DD) in the November 7, 1996 submittal is a duplicate of the Design Commitment given in the ITAAC table. Westinghouse is not following their own committed approach given in the SSAR by the present form of the Design Description.

Response:

Westinghouse is following the approach committed to for the design descriptions given in the Certified Design Material. As stated in paragraph 6 on page 14.3-1, the design descriptions in the ITAAC consist of two parts: first, the principal design basis and principal design characteristics that are referenced in the design certification rule and second, the inspections, tests, analyses, and acceptance criteria (ITAAC) required by 10 CFR 52.47(a)(1)(vi) to be part of the design certification application. The principal design basis or characteristic is given as the first paragraph of the design description. The design commitments accompany the design description, as stated.

SSAR Revision:

None.

Question 640.22

Page 14.3-4, sixth para- The seven factors we used for determining what information is significant to safety in the ABWR and System 80+ Design Description (DD) review are replaced by four factors. These four factors do not meet the intent of the seven factors we approved for ABWR and System 80+.

(a) "Whether the feature of function is necessary to satisfy the NRC's regulations in Parts 20, 50, 73 and 100." This should be added.

(b) "Whether the feature or function represents an important assumption or insight from the probabilistic risk assessment." This should be added.

(c) "Whether the feature or function is important in preventing or mitigating severe accidents." This should be added.

(d) "Whether the feature or function in question has had a significant impact on the safety or operation of the plant." This should be added.

(e) "Whether the feature or function in question is typically the subject of a provision in the Technical Specifications." This should be added.

Response:

The following paragraph is added to SSAR subsection 14.3.2.1 as the next to the last paragraph in the second bullet under the heading of "Selection Criteria." This paragraph describes additional criteria used for ITAAC selection.

SSAR Revision:

"In addition, the following questions were considered for each structure, system, or component not already selected for ITAAC using the above selection criteria:

- Are any features or functions necessary to satisfy the NRC's regulations in Parts 20, 50, 52, 73, and 100?
- Are there any features or functions that represent an important assumption for probabilistic risk assessment?
- Are any features or functions important in preventing or mitigating severe accidents?
- Are there any features or functions that have a significant impact on the safety and operation of the plant?
- Are any features or functions the subject of a provision in the Technical Specifications?

If the answer to any of the above questions is yes, then a design description and ITAAC are prepared using the appropriate functions stated in the SSAR and the parameters from the system design calculations."

Question 640.23

Page 14.3-5, third para-There is one exception to the rule. This pertains to nuclear fuel, and rod cluster control assemblies. These components should be described in the certified design descriptions due to their importance to safety and the desire to control their overall design throughout the lifetime of a plant that references AP600 standard plant design.

Response:

Revision 3 of the Tier 1 submittal will commit to seismic design of the fuel and rod cluster assemblies. This commitment will be contained in Section 2.1.3, Reactor System. The design description of these assemblies will be provided in the Tier 2* submittal.

SSAR Revision:

None.

Question 640.24

Page 14.3-6, third para- There should be a discussion of the detailed review and verification of the input parameters and assumptions used for the various analyses such as flooding analyses, overpressure protection, containment analyses, core cooling analyses etc. (refer to the similar write-up given for ABWR and System 80+).

Response:

The following paragraph is added to SSAR subsection 14.3.2.1 as the second paragraph under the heading "Selection Criteria":

SSAR Revision:

"A review of those sections of the AP600 SSAR that document plant safety evaluations was conducted. Specifically, reviews were conducted of the following chapters of the AP600 SSAR: the flooding analysis in Chapter 5, the analysis of overpressure protection in Chapter 5, containment analysis in Chapter 6, the core cooling analysis in Chapters 6 and 15, the analysis of fire protection in Chapter 9, the safety analysis of transients in Chapter 15, the analysis of anticipated transients without scram (ATWS) in Chapters 7 and 15, the radiological analysis in Chapter 15, the resolution of unresolved or generic safety issues and Three Mile Island issues in Chapter 1, and the PRA and severe accident information in Chapter 19. These reviews were important in identifying safety-related system design information warranting consideration in the design descriptions and the accompanying design commitments."

Question 640.25

Page 14.3-6, last para- Design Description entries for safety-related systems are significantly different than the entries given in the SRP. Westinghouse should justify the deviations from the SRP.

Response:

The following bullets, shown in italics, are added to the paragraph delineating the content of the design description entries for safety-related systems under the heading "Selecti... Methodology" in SSAR subsection 14.3.2.1:

SSAR Revision:

"For safety-related systems, application of this criteria results in design description entries that include the following information, as applicable:

- System name and scope
- System purpose
- Summary of the system's safety-related components (usually shown by a figure)
- *Equipment seismic and ASME classifications*
- *Piping ASME classification and Leak Before Break criteria*
- Type of electrical power provided for the system
- System's important instruments, controls, and alarms to the extent located in the main control room or remote shutdown workstation
- *Equipment to be qualified for harsh environments*
- Motor-operated valves within the system that have an active safety-related function
- Other features or functions important to safety"

With the addition of these entries to the AP600 design descriptions, the intent of the draft SRP has been met.

Also, Revision 3 of the AP600 Certified Design Material includes table of content entries for all AP600 systems. Therefore, the last sentence of the last paragraph in SSAR subsection 14.3.2.1 is deleted.

Question 640.26

Since the RTNSS issue is not resolved, it is difficult to finalize the ITAAC for the following systems:

(a) Normal RHR
(b) Diesel Generators
(c) CVCS
(d) Start-up Feedwater System

Response:

The important functions of these four systems are addressed in the ITAAC at the appropriate level of detail for a nonsafety function. The issues remaining to be resolved on regulatory treatment of nonsafety systems (RTNSS) will not cause any of these systems to be deleted from the ITAAC or to be treated at a different level of detail. While it is important to achieve closure on both issues, we do not believe that review of the ITAAC on these four systems need wait for closure on RTNSS.

SSAR Revision:

None.

Question 640.27

Westinghouse should provide information in tabular form, in Section 14.3 of the SSAR that cross references the important design information and parameters of the following analyses to their treatment in Tier 1.

(a) flooding analyses
(b) overpressure protection
(c) containment analyses
(d) core cooling analyses
(e) fire protection
(f) transient and accident analyses
(g) ATWS
(h) Steam Generator Tube Rupture
(i) radiological analyses
(j) USIs/GSIs
(k) TMI-2 Action Items

Response:

The requested tables will be included in Section 14.3 of SSAR. These tables will cross reference the important design information and parameters for the analyses listed in the RAI with the exception of unresolved safety issues/generic safety issues (USIs/GSIs) and Three Mile Island-2 (TMI-2) Action Items.

Design features important in the discussion of generic safety issues, unresolved issues, and TMI-2 issues are included in the discussion of the system design and design features within the SSAR. The function of SSAR Section 1.9 is to summarize the conformance with the regulatory guidance on these issues and reference other SSAR sections that provide additional detail. Because the important parameters and assumptions are identified in other SSAR sections and are encompassed in the tables for the other safety analyses, there is no additional information in the areas of generic safety issues or unresolved issues, and TMI-2 issues that require inclusion in a separate table.

SSAR Revision:

Revised Section 14.3 is being provided under separate cover.

Question 640.28

Listing of instruments with their tag numbers in a table is not sufficient. Minimum set of instruments should be shown in the figure to show the functional arrangement. The overall locations of the instruments are essential for the function. The instruments' exact locations need not be shown in the diagram. Typically in the P&ID, the instruments are shown without showing the exact place where they are installed.

Response:

The overall location of the instruments, where important, is identified in the "Equipment Name" column of each table (for example, RCS *Hot Leg 1* Narrow Range Temperature Sensor). We believe this approach is equivalent to showing instruments on the figures and has less potential for misinterpretation.

SSAR Revision:

None.

Question 640.29

We understand that numerical criteria given in the ITAAC are different from the numbers given in the SSAR. The numbers in Tier 2 and Tier 1 should be consistent. If they are different, there should be an analysis or outline of the analysis in Tier 2 justifying the deviation.

Response:

We have instituted a programmatic review of the ITAAC against the SSAR to ensure that Tier 1 is consistent with Tier 2, and to justify any different numbers.

SSAR Revision:

See the individual RAI responses dealing with inconsistencies identified by the staff.

Question 640.30

It is important to state in the beginning of the DD whether the system is a safety grade System or a Defense-in Depth System or a non-safety related System. This statement will dictate the content of the DD using the graded approach. We understand that some systems will be a combination of safety and non-safety. But still it is possible to portray a system. We used this approach in the review of Evolutionary plants and found it useful.

Response:

Because the Certified Design Material is a legal document, precise wording is absolutely essential. Because safety classifications are assigned on a functional rather than a system basis, it would be imprecise and dangerous to describe entire systems as safety or nonsafety.

SSAR Revision:

None.

Question 640.31

In the draft AP600 TIER 1 material submitted on June 28, 1996 a definition of "Defense-in-depth Systems" was given. Why this definition is not included in the November 7, 1996 submittal?

Response:

It is important to distinguish between safety and nonsafety functions because Tier 1 treats them at different levels of detail. But it is unnecessary and potentially misleading to distinguish between different types of nonsafety functions because they are all treated at a similar level of detail. Therefore, we have deleted the references to defense-in-depth (DID) functions and simply labeled them nonsafety functions.

SSAR Revision:

None.

Question 640.32

Add "Division (for mechanical systems or component)."

Response:

The AP600 Certified Design Material does not use the term division to refer to mechanical systems or components. Therefore, it would be misleading to include it in the definitions.

SSAR Revision:

None.

Question 640.33

Add "Maximum Reactor Core Thermal Power."

Response:

This has been added in Revision 3.

SSAR Revision:

None.

Question 640.34

Submit the Design Description for Nuclear Fuel System and Control Rod Drive System (System 80 + DCD may be referred for an acceptable submittal). Even though "ITAAC" will not be required for these systems, a basic configuration inspection will be required. Tier 2* documentation for these systems should be submitted for staff review.

Response:

Tier 2* documentation for these systems will be submitted for staff review.

SSAR Revision:

None.

Question 640.35

Digital Metal Impact Monitoring System described in SSAR Section 4.4.6.4 which is used for monitoring loose parts in the reactor should be in the ITAAC.

Response:

An ITAAC for the Digital Metal Impact Monitoring System has been developed and was provided to the NRC in Revision 3 of the AP600 ITAAC.

SSAR Revision:

None

Question 640.36

Since ITP will be significantly used for verification of ITAAC, resolution of our comments on ITP are essential for completion of the ITAAC review.

Response:

While it is important to achieve closure on both ITAAC and ITP, review of the ITAAC need not wait for closure on ITP. We encourage the staff to perform these reviews in parallel.

SSAR Revision:

None.

Question 640.37

The certified design material (CDM) and the inspection, tests, analyses and acceptance criteria (ITAAC) for the AP600 I&C systems should provide information on the design process and implementation, with appropriate tests, inspection and acceptance criteria, based on supporting information in SSAR Chapter 7 and Section 14.3. The material should include information on the design controls, development, and qualification processes for I&C hardware, software, and other design features.

Response:

The Certified Design Material and ITAAC will be revised to provide additional detail of the design process. This additional detail will include a description of the life cycle stages for which the process is applicable; a description of the elements of the process as relating to software management, configuration management, and verification and validation; a description of the elements of the process relating to commercial dedication of commercial off-the-shelf hardware and software; and a description of the inspection and acceptance criteria used to show that the process was used. SSAR subsection 7.1.2.15 will be expanded to list the life cycle stages for which the design process, described in WCAP-13383, is applicable. The SSAR will also be revised to describe that the control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant.

SSAR Revision:

The following changes will be made in subsection 7.1.2.15 of the SSAR:

WCAP-13383 ~~also~~ provides ~~for the use of commercial off the shelf hardware and software through a commercial grade dedication process.~~ a planned design process for hardware and software development during the following life cycle stages:

- Design requirements phase
- System definition phase
- Hardware and software development phase
- System test phase
- Installation phase

WCAP-13383 also provides for the use of commercial off-the-shelf hardware and software through a commercial dedication process. Control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant as described in subsection 13.5.1.

Question 640.38

The CDM should address the hardware and software development process to be used in the design, testing, and installation of I&C equipment and should also include the description of the design process to be followed for hardware and software development, design commitments, the inspections, tests, and analysis to be performed to verify that the design is consistent with the commitments, and acceptance criteria against which the design will be judged.

The commitment in the ITAAC should reflect the elements, activities, and documentation required of the various phases of the life cycle as shown in Figure 1 of SRP Section 14.3.5.

Response:

The Certified Design Material and ITAAC will be revised to provide additional detail of the design process. This additional detail will include a description of the life cycle stages for which the process is applicable; a description of the elements of the process as relating to software management, configuration management, and verification and validation; a description of the elements of the process relating to commercial dedication of commercial off the shelf hardware and software; and a description of the inspection and acceptance criteria which is used to show that the process was used. SSAR subsection 7.1.2.15 will be expanded to list the life cycle stages for which the design process, described in WCAP-13383, is applicable. The SSAR will also be revised to describe that the control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant.

SSAR Revision:

The following changes will be made in subsection 7.1.2.15 of the SSAR:

WCAP-13383 ~~also~~ provides ~~for the use of commercial off the shelf hardware and software through a commercial grade dedication process.~~ a planned design process for hardware and software development during the following life cycle stages:

- Design requirements phase
- System definition phase
- Hardware and software development phase
- System test phase
- Installation phase

WCAP-13383 also provides for the use of commercial off-the-shelf hardware and software through a commercial dedication process. Control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant as described in subsection 13.5.1.

Question 640.39

Provide criteria in the CDM and SSAR to guide the design process throughout the digital I&C systems life cycle stages. The ITAAC should provide the acceptance criteria for verifying the design through the stages while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design. The ITAAC for software and hardware for the I&C systems should verify the design stages within the overall design process as specified in the WCAP-13383, Revision 1:

a) Design requirement phase
b) Definition phase
c) Development phase
d) Test phase (integration, verification, and validation)

In addition to the four phases listed above, the staff believes that two more phases should be added:

e) Installation phase
f) Operation and maintenance phase.

The ITAAC for software development should include, but not be limited to the following elements:

* software quality assurance plan (SQA)
* software management plan (SMP)
* software configuration management plan (CMP)
* software development plan (SDP)
* verification & validation plan (V&VP)
* software safety plan (SSP)
* software operation and maintenance plan (SOMP)

Response:

The Certified Design Material and ITAAC are revised to provide additional detail to guide the design process throughout the specified life cycle stages. This additional detail includes requirements for the review of specific design documentation and reviews during the following life cycle stages:

a) Design requirements phase
b) System definition phase
c) Hardware and software development phase
d) System test phase
e) Installation phase

Control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant.

In addition, the ITAAC will be revised to provide a description of the software development as described in the following elements:

- Software management as typically documented in the software quality assurance plan, software management plan, software development plan, software safety plan, and software operation and maintenance plan
- Software configuration management plan
- Software verification and validation plan

SSAR Revision:

The following changes will be made in subsection 7.1.2.15 of the SSAR:

WCAP-13383 ~~also~~ provides ~~for the use of commercial off the shelf hardware and software through a commercial grade dedication process.~~ a planned design process for hardware and software development during the following life cycle stages:

- Design requirements phase
- System definition phase
- Hardware and software development phase
- System test phase
- Installation phase

WCAP-13383 also provides for the use of commercial off-the-shelf hardware and software through a commercial dedication process. Control of the hardware and software during the operational and maintenance phase is the responsibility of the Combined License applicant as described in subsection 13.5.1.

Question 640.40

The CDM should address the development and qualification process for I&C equipment. The discussion should include:

a) design processes and acceptance criteria to be used for safety-related systems using programmable microprocessor-based control equipment,

b) a program to assess and mitigate the effects of electromagnetic interference on I&C equipment,

c) a program to establish setpoints for safety-related instrument channels,

d) a program to qualify safety-related I&C equipment for in-service environmental conditions, including mild environmental conditions with the potential for local hot spots due to abnormal conditions.

e) a program to verify the conformance of the safety-related I&C systems in accordance with guidance provided in IEEE standards 279 and 603.

f) a program to verify the independence between redundant divisions. In addition to separation requirements, the isolation aspects should be also addressed.

Response:

The Certified Design Material (CDM) and ITAAC are revised to address the NRC concerns as follows:

a) The revised CDM and ITAAC describe the design process and acceptance criteria which will be used for the safety-related equipment as discussed in the response to NRC RAI numbers 640.37, 640.38, and 640.39.

b) The revised CDM and ITAAC include requirements for qualification of the safety-related I&C equipment for electrical surge withstand capability (SWC), electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD).

c) The revised CDM and ITAAC include requirements for the development of setpoints for safety-related instrument channels using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.

d) The revised CDM and ITAAC include requirements for qualification of the safety-related I&C equipment for in-service environmental conditions, including room ambient temperature, humidity, pressure, and mechanical vibration.

e) Conformance of the safety-related I&C systems to the guidance provided in IEEE standards 279 and 603 is part of the design process and related acceptance criteria described in a). The need to meet these standards is established as part of the design requirements phase. The verification and validation activities provide the reviews necessary to establish that the installed hardware meets all requirements established during the design requirements phase.

f) The revised CDM and ITAAC include requirements to verify the independence of redundant safety-related divisions including requirements for isolation devices.

SSAR Revision:

None.

Question 640.41

The CDM should include an Instrumentation and Control Systems Architecture Block Diagram similar to Figure 7.1-1 in the SSAR.

Response:

The Certified Design Material is revised to include a figure depicting the functional arrangement of the protection and safety monitoring system. A figure that depicts a detailed hardware configuration will not be provided in the Certified Design Material. The Certified Design Material defines the design process which will be used; however, the final hardware implementation will depend on the technology available for use during the hardware development phase of the design process life cycle.

SSAR Revision:

None.

Question 640.42

In addition to the PMS and DAS, the I&C CDM and ITAAC should include the following I&C systems:

PLS - Plant Control System
DDS - Data Display and Processing System
OCS - Operations and Control Centers System
IIS - Incore Instrumentation System
SMS - Special Monitoring System

Response:

The Certified Design Material and ITAAC have been revised to include the PLS, DDS, OCS, IIS, and SMS.

SSAR Revision:

None.

Question 640.43

Re: Open Item 5101

The CDM and ITAAC should include the communication system that verifies the communication between the main control room and the local control stations, and the remote shutdown station and the local control stations.

Response:

The communication system has been included in Revision 3 to the Certified Design Material and ITAAC.

SSAR Revision:

None.

Question 640.44

In the CDM for PMS, the description of the logic and control should have more detail when addressing automatic decision-making and trip logic functions, and manual initiation functions associated with the safety actions of the safety-related systems.

Response:

The Certified Design Material and ITAAC for the protection and safety monitoring system (PMS) have been revised to provide more detail of the automatic decision-making and trip logic functions associated with bypassing of reactor trip and engineered safety feature actuation channels. The Certified Design Material and ITAAC will describe that the two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. If a second channel is bypassed, the PMS two-out-of-four initiation logic reverts to a one-out-of-two coincidence logic. The PMS automatically produces a reactor trip or engineered safety feature initiation upon an attempt to bypass more than two channels of a function that uses two-out-of-four initiation logic. All bypassed channels are alarmed in the main control room.

SSAR Revision:

None.

Question 640.45

The CDM and ITAAC for the DAS should follow the commercial grade item dedication program as defined in the WCAP-13383 Revision 1. The DAS CDM should address defense-in-depth considerations for protection against common mode failures in the PMS.

Response:

The Certified Design Material and ITAAC for the diverse actuation system (DAS) have been revised to provide details of the design process that will be used for the DAS. This DAS design process will provide for the use of commercial off-the-shelf hardware and software. Changes to the Certified Design Material and ITAAC for the DAS, to address defense-in-depth considerations for protection against common mode failures in the PMS, are not required. The Certified Design Material and ITAAC for the DAS defines specific features to provide such common mode failure protection. The presence of these features is inspected as part of the DAS ITAAC process.

SSAR Revision:

None.

Question 640.46

Subsection 3.4, "Initial Test Program:" While the staff agrees that "ITAAC aimed at verification of the initial test program are not necessary," the initial test program design description needs to summarize, in comprehensive detail, the fundamental initial test program objectives, phases, and organizational elements as described in SSAR Section 14.2 (subsections 14.2.1, "Summary of Test Program Objectives," through 14.2.3, "Test Specifications and Test Procedures").

Response:

Section 3.4, Initial Test Program, of the AP600 Certified Design Material, has been modified to include a discussion of the fundamental initial test program objectives and phases as described in SSAR Section 14.2 (subsections 14.2.1, "Summary of Test Program Objectives," through 14.2.3, "Test Specifications and Test Procedures"). Because the detailed description of organization elements is a combined license applicant requirement, a description of the organizational details would be inappropriate to include in Section 3.4 of the AP600 Certified Design Material.

SSAR Revision:

None.

Question 640.47

Re: Design Reliability Assurance Program (D-RAP)

The Certified Design Material information for the AP600 should contain a high level commitment to a D-RAP for use in the detailed design and equipment specification of risk-significant SSCs prior to fuel load. The D-RAP design description should describe the scope, purpose, objectives and essential elements of the D-RAP, including, (a) a commitment for a process to evaluate, prioritized and list SSCs based on their risk-significance, (b) a commitment that the process used to determine dominant failure modes will consider industry experience, analytical models, and applicable requirements, and (c) a commitment that for risk-significant SSCs, the key assumptions and risk insights will consider operations, maintenance, and monitoring activities.

Response:

The Certified Design Material for the AP600 contains a high level commitment to a design reliability assurance program (D-RAP) for use in the design and equipment specification of risk-significant structures, systems, and components (SSCs) prior to fuel load. This commitment is confirmed by the existence of the list of risk-significant SSCs provided in the Certified Design Material. The SSC list is given priorities based upon risk-significance in terms of risk achievement worth, risk reduction worth, Fussel Vesely worth, and a number of expert panel considerations. The SSCs are evaluated so that inclusion and exclusion of SSCs is not based solely on PRA measures, but also on expert panel insights, industry experience, and applicable requirements. Dominant failure modes of the risk-significant SSCs are captured from industry experience, analytical models, and applicable requirements, which together provide a large source of information. The Certified Design Material Acceptance Criteria (Table 3.7-3) states that the estimated reliability of the SSCs must be at least equal to the assumed reliability.

The Certified Design Material commitment that for risk-significant SSCs, the key assumptions and risk insights will consider operations, maintenance, and monitoring activities is documented by the inclusion of the actual SSC equipment list (Table 3.7-1) of the D-RAP Certified Design Material.

SSAR Revision:

None.

Question 640.48

Re: Section 3.3 - Nuclear Island Buildings

There is no ITAAC on the configuration and thickness of shield walls. Such an ITAAC is needed to validate the SSAR calculations of 1 or plant radiation dose rates. It was expected that such an ITAAC would exist, possibly in the building ITAAC (section 3.3). One way of accomplishing the ITAAC would be to have a set of drawings that show the walls and their thickness.

Response:

Two ITAACs have been created to address the configuration and shielding thicknesses for both the Nuclear Island and the annex and radwaste buildings. These ITAACs are shown below.

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Selected walls of the NI buildings as defined on Table 3.3-1 provide shielding during normal operations. The shield wall thicknesses of the NI buildings are defined on Table 3.3-1 except for designed openings or penetrations.	Inspection of the as-built NI building wall thicknesses, identified on Table 3.3-1 will be performed.	The as-built inspection report exists and concludes that the shield walls of the NI buildings as defined on Table 3.3-1 are consistent with the minimum shield wall thicknesses defined on Table 3.3-1.
3. Selected walls of the annex building and the radwaste building as defined on Table 3.3-1 provide shielding during normal operations. The shield wall thicknesses of the annex building and the radwaste building are defined on Table 3.3-1 except for designed openings or penetrations.	Inspection of the as-built annex building and the radwaste building wall thicknesses, identified on Table 3.3-1 will be performed.	The as-built inspection report exists and concludes that the shield walls of the annex building and the radwaste building as defined on Table 3.3-1 are consistent with the minimum shield wall thicknesses defined on Table 3.3-1.

SSAR Revision:

None.

Question 640.49

There was no ITAAC on ventilation flow rates. Such an ITAAC is needed to validate the SSAR calculations of inplant concentrations of airborne radioactivity.

Response:

ITAAC 2.7.5 for the Radiologically Controlled Area Ventilation System and ITAAC 2.7.6 for the Containment Air Filtration System have been added to address this concern.

SSAR Revision:

None.

Question 640.50

Re: Open Item 5108

Section 3.3.4.c) states that the separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas as identified in Table 3.3.2. ITAAC Table 3.3-4 Section 4.c) also refers to fire areas for the cable separation. Inspection of the as built will be done in general plant areas for 12 in. vertical separation and 6 in. horizontal separation for open cable trays. The following areas need to be clarified:

a) Why are you referring to fire areas? It is not for fire protection review.

b) SSAR Section 8.3.2.4.2, Rev. 8 states that within general plant areas (limited hazard areas), the minimum vertical separation is 12 inches and the horizontal separation is 6 inches for open cable trays with low voltage power cables for sizes < 2/0 AWG.

Response:

Certified Design Material and ITAAC Section 3.3.4 have been revised as follows:

- ITAAC 3.3.4.c.ii will be deleted.
- The section will be replaced with Section 3.3.4.d, which will address electrical separation (versus fire separation).

The ITAAC for electrical separation is consistent with SSAR subsection 8.3.2.4.2

SSAR Revision:

None.

Question 640.51

Re: Open Item 5109

Chapter 1 of the SSAR and the CDM do not show what the boundary is for the AP600 design scope (even Figure 1.2-2 is unclear). In order to meet the requirements of 10 CFR 52.47(b)(1), Westinghouse needs to identify the structures and systems that are wholly or partially outside (or inside) the scope of the design to be certified and specify the boundary of the certified design scope (see SRP 14.3, page 30 for guidance). For example, it appears that the following structures are in the AP600 scope:

Nuclear Island (Containment, Shield, and Aux. buildings)
Turbine building
Annex building
Diesel generator building
Radwaste building
Are the outer walls of these buildings considered the boundary??

Response:

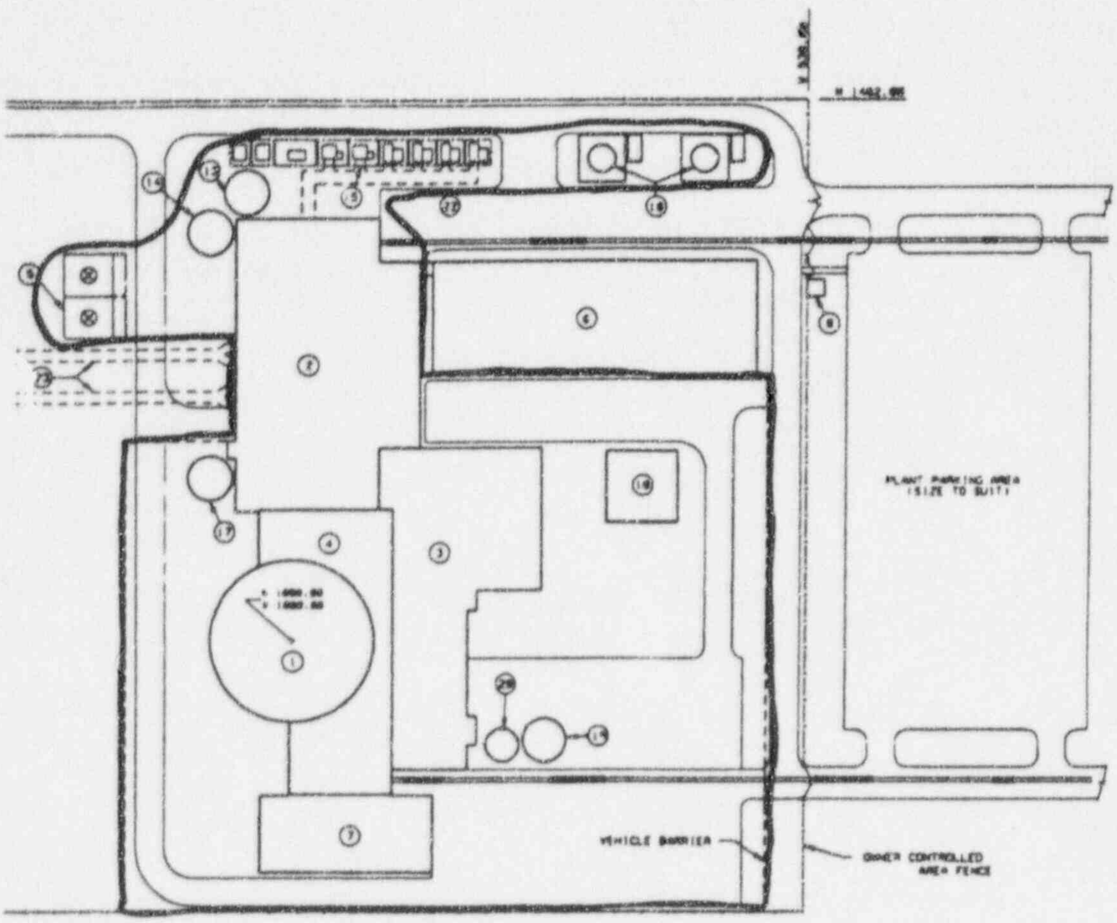
As stated in Section 1.8 of the AP600 SSAR, the AP600 plant design included in the application for Design Certification incorporates the entire Nuclear Island, the annex building, the diesel/generator building, the turbine building, the radwaste building, and their associated equipment, associated yard structures, and security structures. A SSAR change is included below to clarify this boundary definition.

SSAR Revision:

1.8 Interfaces for Standard Design

Second Paragraph

The AP600 is a plant design incorporating the entire nuclear island, the annex buildings and associated equipment, the diesel/generator building and associated equipment, the turbine generator building, the turbine/generator equipment and the radwaste facilities. The physical boundary of the portion of the AP600 design included in this application for Design Certification is shown on the site plan, Figure 1.2-2. It includes arrangement and placement of structures within the indicated boundary including the vehicle barriers necessary for security, but not the boundary fence. As a result, no interfaces need to be identified between or among these portions of the plant. They are addressed in their appropriate section of this SSAR. There are, however, a number of safety-related informational, administrative or operational interfaces between the AP600 design and other portions of a completely licensed facility which must be addressed by parties that reference the AP600 design. These interfaces are identified in Table 1.8-1 in the order they are presented in this SSAR.



LEGEND

14. FIRE WATER STORAGE TANK
15. TRANSFORMER AREA
16. SWITCHYARD
17. CONDENSATE STORAGE TANK
18. DIESEL GENERATOR FUEL OIL STORAGE TANKS
19. DEMINERALIZED WATER STORAGE TANK
20. BORIC ACID STORAGE TANK
21. HYDROGEN STORAGE TANK AREA
22. TURBINE BUILDING LAYDOWN AREA
23. CIRCULATING WATER PIPE
24. WASTE WATER RETENTION BASIN

Question 640.52

Re: Open Item 5110

Interface requirements - Section 1.8 of the SSAR and 4.0 of the CDM are unacceptable. In order to meet 10 CFR 52.47(a)(vii) & (ix), Westinghouse needs to specifically identify the structures and systems that are wholly or partially outside the design scope and specify the interface requirements for those systems. Also, Westinghouse needs to describe the method to be used to verify the interface requirements in order to meet 52.47(a)(viii). Refer to 4.0 of ABWR ITAAC to see how GE did this and SRP 14.3-30.

Response:

The Certified Design Material includes identification of all structures and systems that are wholly or partially in the AP600 Design Certification scope, regardless of safety significance.

SSAR Revision:

See response to RAI 640.51 for revised Section 1.8.

Question 640.53

Re: Open Item 5111

Westinghouse needs to identify all structures and systems that are wholly or partially in the AP600 design scope in Tier 1 ITAAC, regardless of safety significance. Each system needs at least one page in the ITAAC book (see example from ABWR) and more detail can be provided in Tier 2 as necessary.

Response:

The extent of the AP600 design included in the application for Design Certification is clarified in our response to RAI 640.51. That response includes a revision to the SSAR for clarification. Section 4 of the Certified Design Material has been changed to state that there are no requirements to be met by those portions of the plant for which the application does not seek certification under 10 CFR Part 52. This statement is consistent with the interface requirements included in the CE System 80+ certified Design Material.

SSAR Revision:

None.

Question 640.54

Basic configuration - It appears that some verification capability was lost in the CDM because the term "Basic Configuration" was replaced with "Functional Arrangement" (such as design descriptions that do not become verified commitments and verification against design drawings - "Bridge Concept").

Response:

We have revised the definition of "Functional Arrangement" in Revision 3 to reference the design descriptions.

SSAR Revision:

None.

Question 640.55

The ITAAC in Section 2.2.1 do not include stroke times for the containment isolation valves.

Response:

ITAAC Section 2.2.1, Revision 3, includes a commitment to test the stroke times of the containment isolation valves as follows:

Table 2.2.1-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.	i) A containment integrated leak rate test will be performed. ii) Testing will be performed to demonstrate that remotely operated containment isolation valves close rapidly.	i) The leakage rate from containment for the integrated leak rate test is less than L_a. ii) A report exists and concludes that the containment purge isolation valves close within 5 seconds and all other containment isolation valves close within 60 seconds upon receipt of an actuation signal.

SSAR Revision:

None.

Question 640.56

The following comments relate to Section 2.3.9, "Containment Hydrogen Control System," of the CDM which involve the hydrogen recombination subsystem and the hydrogen ignition subsystem. The hydrogen recombination subsystem provides hydrogen control during and the following a design basis LOCA while the hydrogen ignition subsystem provides hydrogen control during and following a degraded core or core melt scenario.

For the hydrogen recombination subsystem, the applicant has specified (a) Design Commitments, (b) Inspection, Tests, Analyses, and (c) Acceptance Criteria in Table 2.3.9-2 of the CDM document. The HRS is provided to meet the requirements of GDC 41, "Containment Control Systems in Light-Water-Cooled Power Reactors." Westinghouse references RG 1.7, "control of Combustible Gas Concentrations in Containment following A Loss-Of-Coolant Accident," in the SSAR as the methodology used for implementing these regulations. The ITAAC fail to address these requirements and Westinghouse's commitment to the methodology in RG 1.7.

a) Section 6.2.4.1.1, "Containment Mixing", of the SSAR is to provide an analysis which shows that excessive stratification of combustible gases will not occur within the containment or within a containment subcompartment. Verification of the analysis that supports the design commitment to provide a system and features to mix the combustible gases within containment has been omitted from the ITAAC.

Response:

The containment mixing commitment is incorporated in the ITAACs in a distributed form. The mixing function is driven by the passive containment cooling function. The application of water to the external surface maintains the containment shell at a cool temperature. The condensation of the steam on the interior of the containment shell creates a downward flowing layer at the wall to prevent stagnation in the dome. As the air flows downward along the wall, the wall layer also entrains surrounding mixture creating significant mixing forces. Additional details of containment mixing are discussed in subsection 6.2.4 of the SSAR. The inclusion of ITAACs for the Passive Containment Cooling System (PCS) components and operation and the configuration of the Containment System and the Nuclear Island buildings confirms the presence of systems to mix combustible gases within containment.

b) Design Commitment No. 3.a of Table 2.3.9-2 fails to verify conformance with design criteria, such as NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.F.1, Attachment 6, containment Hydrogen Monitor, and RG 1.97, for the containment hydrogen monitor.

Response:

Design commitments are established in Revision 3 of ITAAC 2.3.9 to provide for containment hydrogen monitoring consistent with the requirements of NUREG-0737 and Regulatory Guide 1.97 relative to equipment qualification, redundancy, power source design, quality assurance, and display. Other aspects of the requirements – such as channel availability, range, equipment identification, interfaces and servicing, testing, and calibration – are more appropriately addressed in the SSAR and Technical Specifications.

Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The seismic Category I equipment identified in Table 2.3.9-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.9-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.9-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
2.a) The equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on equipment located in a harsh environment.	A report exists and concludes that the equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
2.b) The Class 1E components identified in Table 2.3.9-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.9-1 when the assigned Class 1E division is provided the test signal.
2.c) Separation is provided between VLS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Certified Design Material, Section 3.3, Nuclear Island Buildings.	See Certified Design Material, Section 3.3, Nuclear Island Buildings.

Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
4.a) The VLS provides hydrogen monitors for indication of the containment hydrogen concentration.	Inspection for the existence of three Class 1E hydrogen monitors inside containment will be performed.	Three hydrogen monitors powered by a Class 1E power source are provided inside containment.
6. Safety-related displays identified in Table 2.3.9-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.9-1 can be retrieved in the MCR.

c) Design Commitment No. 3.b of Table 2.3.9-2 verifies the existence of a report that establishes the depletion rate for a single full-size PAR. There is no link between this acceptance criteria and the installed PARs.

Response:

The recombination rate specified in Design Commitment 4.b. of Revision 3 of the ITAACs specifies that "A report exists and concludes that the PAR depletion rate for each installed PAR is greater than or equal to 1 scfm of hydrogen at a prevailing concentration of 3 volume-percent...". The basis for this value is that under conservative design basis assumptions, the hydrogen production rate (Figure 6.2.4-3 of the SSAR) by the time containment concentration reaches a concentration of 3 percent (Figure 6.2.4-2) is less than 1 scfm.

d) Either the criteria used to locate the PARs inside containment or a description of their specific location inside containment should be provided in the ITAAC. Because the criteria used to locate the recombiners was subjective and based on engineering judgement, the staff recommends the use of a detailed description or figure to verify appropriate location of the PARs.

Response:

The specific and exact location of the passive autocatalytic recombiners (PARs) are not critical to proper operation of the hydrogen control function. Generally, it is desirable that the units be above the loop compartment (elevation 148') and several feet in from the side wall in order to be within a well mixed, nondown-draft region above the operating deck. The present design commitment verifies elevation location but does not confirm distance from the containment wall. This commitment will added to the ITAAC as indicated below.

Table 2.3.9-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.b) The VLS provides PAR devices for control of the containment hydrogen concentration during and following a design basis accident.	i) Inspection for the existence of two PAR devices inside containment will be performed.	i) Two PAR devices are provided inside containment within the upper compartment between elevations 150 and 175 ft and with PAR centerline greater than 10 ft from the containment shell.
	ii) Type tests, analyses, or a combination of type tests and analyses will be performed on the PARs.	ii) A report exists and concludes that the PAR depletion rate for each installed PAR is greater than or equal to 1 scfm of hydrogen at a prevailing concentration of 3 volume-percent for a test conducted at atmospheric pressure +2 psi and an ambient temperature of 120.

e) The ITAAC should verify the existence of a report that concludes that the installed PARs are qualified for a harsh environment and can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function. The report should also address the potential of the fission products that make up the post-accident radiation environment to be catalytic poisons.

Response:

Design commitment 2.a is made is to confirm that the PARs are qualified for operation during and following a harsh environment. Section 3.11 and subsection 6.2.4 of the SSAR specify environment for the equipment qualification. For the PARs, the environment includes potential catalyst poisons which could be present for the scenarios, including core damage events up through in-vessel releases as identified in these SSAR sections.

Table 2.3.9-3 Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
2.a) The equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on equipment located in a harsh environment.	A report exists and concludes that the equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

f) For the hydrogen ignition subsystem (HIS), the applicant has specified (a) Design commitments, (b) Inspection, Tests, Analyses, and (c) Acceptance Criteria in Table 2.3.9-2 of the CDM Document. The HIS is provided to safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal water reaction as required by 10 CFR 50.34(f)(2)(ix). The system also ensures that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent (by volume). This is accomplished by initiating a deflagration at the lower level of hydrogen flammability.

The ITAAC fail to verify several important design features of the HIS as described in Section 6.2.4.2.3 of the SSAR. The igniters have been divided into two power groups. Power to each group will be normally provided by offsite power, however should offsite power be unavailable, then each of the power groups is powered by one of the onsite non-essential diesels and finally should the diesels fail to provide power then approximately 4 hours of igniter operation is supported by the non-Class 1E batteries for each group. Assignment of igniters to each group is based on providing coverage for each compartment or area by at least one igniter from each group. The igniter assembly is designed to maintain the surface temperatures within a range of 1600 to 1700 degrees F. These design features are essential in establishing the HIS's ability to initiate a deflagration at the lower level of hydrogen flammability and should be verified by the ITAAC.

Response:

Table 2.3.9-2 of the ITAACs identifies the location (that is, compartment coverage of each igniter and the power group to which each igniter is assigned. Design commitment 3 confirms power group assignment, while design commitment 5 verifies both the igniter location and surface temperature of the igniters. Testing of the nonessential standby diesels is performed in ITAAC Section 2.6.4 "Onsite Standby Power System". The capacity of the of the non 1E batteries is verified in ITAAC Section 2.6.2 "Non-Class 1E dc and Uninterruptible Power Supply System."

Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3. The components identified in Table 2.3.9-2 are powered from their respective non-Class 1E power group.	Testing will be performed by providing a simulated test signal in each non-Class 1E power group.	A simulated test signal exists at the equipment identified in Table 2.3.9-2 when the assigned non-Class 1E power group is provided the test signal.
5. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	i) Inspection for the number of igniters will be performed. ii) Operability testing will be performed on the igniters.	i) At least 60 hydrogen igniters are provided inside containment at the locations specified in Table 2.3.9-2. ii) The surface temperature of the igniter exceeds 1700°F.

g) Either the criteria used to locate the igniters inside containment or a description of their specific location inside containment should be provided in the ITAAC. Because the criteria used to locate the recombiners was subjective and based on engineering judgement, the staff recommends the use of a detailed description or figure to verify appropriate location of the recombiners.

Response:

The location of the igniters is addressed in part f) above and the location of the PARs is addressed in d) above.

SSAR Revision:

None.

Question 640.57

The following comments relate to Section 2.2.2, "Passive Containment Cooling System," of the CDM.

a) A report should be prepared as part of Design commitment 6.a) to provide documentation that the integrated flow from the three PCS flow phases, in combination with the inventory verification under Design Commitment 6.e), assures that the PCCWST can provide cooling water for the required 72 hour period. The minimum flow requirements of Design Commitment 6.a) are not sufficient. Documentation of the measured flow rates can also be used to determine degraded flow capability over the life of the plant.

Response:

Design commitment 8.a) provides a commitment to verify integrated flow for the three phases of flow rate to the containment. Design commitment 8.a) also verifies the system capability to provide continuous cooling water flow for the first 72 hours following actuation. The passive containment cooling system (PCS) testing under the plant initial test program will measure and record the actual flow rates which are used subsequently to evaluate system degradation.

Table 2.2.2-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8.a) The PCS provides the delivery of water to the outside of the containment vessel.	i) Testing will be performed to measure the PCCWST delivery rate from each of the two parallel flow paths.	i) When tested separately, each of the two flow paths delivers greater than or equal to: – 442 gpm at a PCCWST water level of 23.75 ft ± 0.25 ft above the lowest standpipe – 122 gpm at a PCCWST water level of 20.65 ft ± 0.25 ft above the lowest standpipe – 71.5 gpm at a PCCWST water level of 13.55 ft ± 0.25 ft above the lowest standpipe.
	ii) Testing and or analysis will be performed to demonstrate the PCWST inventory provides 72 hours of cooling.	ii) When tested and/or analyzed with both flow paths delivering and an initial water level at 300.75 ± 0.25, the water inventory provides greater than or equal to 72 hours of flow with a flow rate greater or equal to 62 gpm.
	iii) Inspection will be performed to determine the PCCWST standpipes elevations.	iii) The elevations of the standpipes above the bottom standpipe are: – 6.7 ft ± 0.25 ft – 14.2 ft ± 0.25 ft – 21.7 ft ± 0.25 ft

b) The acceptance criteria for Design Commitment 6.b) needs to include a measurement of the surface area coverage from the PCS water at the upper spring line for each of the three phases of the PCS flow. The minimum coverage fractions need to be verified and consistent with the water distribution test, for example at least 90 percent coverage for the initial phase. In addition it needs to be confirmed that the side wall water coverage is consistent with the water distribution test, both in minimum area coverage and uniformity around the circumference. Documentation of the measured coverage fractions and uniformity of the flow can also be used to determine degraded surface conditions over the life of the plant.

Response:

Design commitment 8.b) has been modified to eliminate a specific reference to the location of measurement and opts to verify containment shell wetting is consistent with the value predicted by the wetting coverage methodology such that the wetting will be greater than or equal to the assumptions of the containment analysis. The measurement is, however, taken only when the flow rate from the PCS system is at a minimum because this point will be the most challenging for wetting coverage.

Table 2.2.2-3 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.b) The PCS provides wetting of the outside surface of the containment vessel.	i) Testing will be performed to measure the wetted surface of the containment vessel from either of the two parallel flow paths to the containment vessel.	i) A report exists concluding that with a PCCWST water level of 6.2 ft ± 0.25 ft above the bottom of the tank, water delivery to the containment shell provides a coverage that is equal to or greater than the amount predicted by the wetting coverage methodology used in the safety analysis. The wetted coverage will be verified with each of the two parallel paths tested separately.
	ii) Inspection of the containment exterior coating will be conducted.	ii) A report exists and concludes that the containment exterior surface is coated with an inorganic zinc coating.

c) Recent design changes to the PCS to address post 72-hour actions in response to the staff requirements memorandum of January 15, 1997, on SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," have not been incorporated into the ITAAC. The most recent description of the design changes is provided in Westinghouse letter NSD-NRC-97-5024, "AP600 Design Changes to Address Post 72-hour Actions," B. A. McIntyre to T. R. Quay, dated March 14, 1997. New design features include increased inventory in the PCCWST, the addition of an on-grade PCS auxiliary water storage tank, and two recirculation pumps which provide the required makeup flow to the PCCWST from the auxiliary tank for the post 72-hour period (for up to seven days). In addition, the PCCWST now also provides makeup to the spent fuel pool (SFP) and the interface between the PCS and SFP systems have not been included in the ITAAC.

The Design Description, Figure 2.2.2-1 and Table 2.2.2-1 need to be updated to include the new post 72-hour design features and the SFP interface.

d) The acceptance criterion for Design Commitment 6.e) needs to be updated to the new PCCWST inventory, and as appropriate CDM 2.3.7, "Spent Fuel Pool Cooling System," needs to be updated to include the PCCWST interface requirements.

e) Design Commitment 6.f), concerning long-term makeup to the PCCWST needs to be modified, and if necessary additional sub-sections added, to address the new post 72-hour design features. For example, demonstration that each recirculation pump can deliver the required flow rate to the PCCWST, that the on-grade PCS auxiliary water storage tank is seismically qualified and can withstand wind and tornado loadings, instrumentation is available to measure water level in the tank, the pump can be supplied from the on-site diesel generator, and verification of the minimum volume of the auxiliary storage tank.

Response:

Revision 3 of Section 2.2.2 has been prepared to update the design description, design commitments and inspection, test, and analysis to address the additional functions of the PCS identified in NSD-NRC-97-5024, "AP600 Design Changes to Address Post 72-hour Actions." The design commitments relative to the power supply for the PCS recirculation pumps are provided in Sections 2.6.1 and 2.6.3 of the ITAAC. An additional design commitment has been added to Revision 3 of the ITAAC to confirm the capability of the components to withstand a seismic event and wind loadings.

5.a) The seismic Category I equipment identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.

b) The equipment and piping provided to address continued passive containment cooling function in the post-72-hour period will withstand a seismic event and hurricane wind loads.

Table 2.2.2-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
5.a) The seismic Category I equipment identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.2-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.2.2-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) The report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
5.b) The equipment and piping provided to address continued passive containment cooling function in the post 72 hour period will withstand a seismic event and hurricane wind loads.	Inspection will be performed for the existence of a report verifying that the as-installed passive containment cooling equipment and piping from the PCCAWST through the recirculation pumps and to the PCCWST including anchorage is designed to withstand the loads resulting from a seismic event and wind loads associated with a hurricane.	The report exists and concludes that the as-installed passive containment cooling equipment and piping from the PCCAWST through the recirculation pumps and to the PCCWST including anchorage is designed to withstand the loads resulting from a seismic event and wind loads associated with a hurricane.

f) The design basis performance of the PCS requires that the containment be coated with an inorganic paint on the exterior surface to enhance surface wetability and PCS water area coverage, and on the interior surface to promote development of the condensation film. Adequate PCS water area coverage on the exterior surface is also based on a system of weirs (referenced to here as water collection troughs, Tag No. PCS-MT-04) which collect and uniformly redistribute the PCS water from the water distribution bucket to the upper spring line of the containment shell. The ITAAC does not address either the inorganic paint or the uniform distribution of the PCS water over the exterior shell surface.

The Design Description needs to be updated to include the inorganic paint as part of the PCS. Requirements for surface preparation, for example requirements of the Steel Structures Painting Council, and application of the paint to its required thickness, based on paint manufacture's requirements and consistent with the Westinghouse PCS test program, need to be included.

Response:

The containment surface wetability assumptions within the containment analysis are dependent on the application of an inorganic zinc coating on the containment shell. Design commitment 8.b) of Revision 3 of the ITAAC confirms the surface is coated with an inorganic zinc coating in addition to the confirmation of performance consistent with the containment analysis.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.b) The PCS provides wetting of the outside surface of the containment vessel.	i) Testing will be performed to measure the wetted surface of the containment vessel from either of the two parallel flow paths to the containment vessel.	i) A report exists concluding that with a PCCWST water level of 6.2 ft ± 0.25 ft above the bottom of the tank, water delivery to the containment shell provides a coverage that is equal to or greater than the amount predicted by the wetting coverage methodology used in the safety analysis. The wetted coverage will be verified with each of the two parallel paths tested separately.
	ii) Inspection of the containment exterior coating will be conducted.	ii) A report exists and concludes that the containment exterior surface is coated with an inorganic zinc coating.

The presence of the weirs assemblies or (water collection troughs) is confirmed in design commitment number 1, referencing the system schematic as a basis for the configuration.

Table 2.2.2-3
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the applicable portions of the PCS is as shown in Figure 2.2.2-1.	Inspection of the as-built system will be performed.	The as-built PCS conforms with the functional arrangement shown in Figure 2.2.2-1.

g) The design basis performance of the PCS is also based on adequate heat removal from the containment atmosphere by internal, structural heat sinks. The design basis analyses of the performance is based on a maximum steel jacket-to-concrete air gap thickness which accounts for shrinkage of the concrete over the life of the plant. An increased air gap thickness will reduce the effective heat transfer and result in an increase in the containment pressure response following a design basis accident. To assure that this maximum air gap thickness is not exceeded over the life of the plant, the concrete composition (for example aggregate size and moisture content), steel T-pin length and initial pour need to be controlled and verified. The ITAAC does not address the minimum air gap thickness.

Suitable requirements concerning the concrete composition, steel T-pin design and initial pour need to be included in the Design Description. Figure 2.2.2-1 and Table 2.2.2-1 should also be modified as appropriate.

Response:

Minor construction details have not been incorporated into the design commitments of the ITAAC. The air gap thickness does not have a major impact on the performance of the PCS and is, therefore, not considered to be a sufficiently high level detail to warrant consideration within the ITAAC.

h) Schematic Figure 2.2.2-1 does not indicate the presence of the combination flow restricting and flow measuring orifice on each PCS water delivery line, as shown in Figure 6.2.2-1, "Passive Containment Cooling System Piping and Instrumentation Diagram," of the SSAR, Revision 11. Installation of the proper orifice on each line is essential to the PCS performance. Figure 2.2.2-1 and Table 2.2.2-1 should be updated to include the orifices.

Response:

The presence of the flow measuring orifices, but more importantly the effect (flow control) of the orifices, will be verified during testing for design commitment item 8.a). Because proper system operation is verified without the actual inspection for the presence of the orifices, there is no need for inclusion of this "inspection/test/analysis."

SSAR Revision:

None.

Question 640.58

The following comments concern Severe Accident Mitigation Features.

a) Draft SRP Section 14.3.11 has the reviewer ensure that appropriate treatment of severe accident design features and containment design features are included in Tier 1. The supporting information regarding the detailed design and analyses should remain in Tier 2. For many of the design features, it may be impractical to test their functionality because of the absence of simulated severe accident conditions. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, may be considered sufficient in Tier 1 treatment. Design features essential to maintaining containment integrity and assuring a low conditional containment failure probability (CCFP) in severe accidents should be selected for treatment in Tier 1. For AP600, these systems would include the reactor cavity flooding system, the hydrogen igniter system, and the ability to manually depressurize the RCS following core damage, since these features are critical to maintaining a low CCFP as shown in Chapter 50 of the PRA. Westinghouse should provide cross references in the appropriate sections of Tier 2 to show how the design features and SSCs found important from PRA, external event analyses, shutdown risk study, and severe accident analyses are verified by the ITAAC. Westinghouse has not adequately addressed severe accident design features in the CDM or provided cross references to show how the important insights or assumptions from the PRA are verified by the ITAAC.

b) Design criteria for severe accident mitigative features are contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The Commission approved these design criteria in an SRM dated July 21, 1993. At a minimum, the key systems and features provided in the AP600 design to address the criteria described in the hydrogen control, core debris coolability, high-pressure core melt ejection, containment performance, dedicated containment vent penetration, and equipment survivability sections of SECY-93-087 should be provided in the ITAAC.

c) One of the more important assumptions in the PRA is the high probability of maintaining reactor vessel integrity during a core melt scenario. Heat is removed from the molten core debris through boiling on the outside of the flooded reactor vessel. This phenomena is often referred to as in-vessel retention in the PRA. In order to credit the in-vessel retention approach, two design objectives must be met. First, the cavity flooding system must cover the lower reactor vessel prior to relocation of the core. Important design criteria for the cavity flooding system are addressed in Section 2.2.3, "Passive Core Cooling System," of the CDM. Second, the reactor insulation system must allow the ingress of water and not interfere with the boiling process. Some of the important criteria to meet this design objective are: flow paths and clearances, ball and cage check valve design, steam vent damper design, ability to sustain differential pressure loads given in Section 39 of the PRA, provisions to prevent plugging by debris. These important criteria associated with the reactor insulation design should be incorporated into the ITAAC.

d) Another important assumption is the containment's ability to meet Service Level C. Important design criteria associated with the containment shell, equipment hatch, electrical penetrations and mechanical bellows should be verified in the ITAAC.

Response:

a) **Cavity Flooding** - The passive core cooling system (PXS) containment recirculation lines are used to drain the IRWST water into the reactor cavity to submerge the reactor vessel. The containment recirculation lines and valves are safety-related and Class 1E and are verified in ITAAC 2.2.3. The manual action to flood is performed through the protection and safety monitory system (PMS) or diverse actuation system (DAS). The ability to manually actuate the containment recirculation is called out in the PMS ITAAC 2.5.2 in Table 2.5.2-4 and in the DAS ITAAC 2.5.1 in Table 2.5.1-2.

   Hydrogen Igniters - The hydrogen igniters, monitoring and actuation system (VLS) is verified in ITAAC 2.3.9.

   Manual RCS Depressurization - Manual actuation of the automatic depressurization system (ADS) is performed through the PMS or DAS. The ability to actuate Stages 1, 2, 3, and 4 ADS valves is called out in PMS ITAAC 2.5.2 in Table 2.5.2-4 and in DAS ITAAC 2.5.1 in Table 2.5.1-2.

b) **Hydrogen Control** - Hydrogen control is verified in ITAAC 2.3.9.

   Core Debris Coolability - The first level of defense against core-concrete interaction (CCI) in the AP600 is provided by in-vessel retention of core debris. Defense-in-depth is provided according to SECY-93-087, by:

   - Cavity flooding to quench debris.
   - Cavity floor area to enhance debris spreading.
   - A layer of concrete to protect the containment shell.
   - Containment capacity to remain below service level C for 24 hours of CCI noncondensible gas generation.

   The in-vessel retention is enabled by RCS depressurization (ITAAC 2.1.2) and cavity flooding (ITAAC 2.1.3). The containment structural capability is verified to meet the ASME code by ITAAC 2.2.1.

   High Pressure Melt Ejection - According to SECY-93-087, high pressure melt injection (HPME) and direct containment heating (DCH) are mitigated by:

   - Providing a reliable RCS depressurization system.

   - Providing cavity design features to decrease the amount of ejected core debris that reaches the upper compartment.

   The AP600 ADS system is verified by ITAAC 2.1.2.

Containment Performance - According to SECY-93-087, the containment performance is assured by the following:

- Containment stresses do not exceed se: ice level C for 24 hours after the onset of the more likely severe accident scenarios.

- After 24 hours, the containment continues to provide a barrier against the uncontrolled release of fission products.

Based on the level 2 PRA results, the systems that allow the AP600 to meet the containment performance goals are those that enable in-vessel retention (IVR) and containment hydrogen control. The passive containment cooling system (PCS) provides margin so that the containment stresses do not exceed service level c for the more likely severe accident scenarios. These systems also allow the containment to remain below service level c stresses beyond 24 hours to continue to provide a barrier against the release of fission products:

- ADS (ITAAC 2.1.2)
- Cavity flooding via containment recirculation lines (ITAAC 2.1.3)
- Reflective insulation
- Hydrogen control (ITAAC 2.3.9)
- PCS (ITAAC 2.2.2)

Dedicated Containment Vent - SECY-93-087 states that the need for a vent is evaluated on a design-specific basis. No dedicated containment vent has been identified for the AP600 because it provides highly reliable, redundant passive containment cooling capability.

Equipment Survivability - Equipment used to achieve a controlled, stable plant condition following a severe accident is evaluated to demonstrate reasonable assurance of operability at the time it is called upon to perform in the severe accident sequence. These analyses are performed using the design basis equipment qualification for which the equipment is currently "ITAAC'ed." No special considerations have been identified as necessary for reasonable assurance of severe accident survivability. This will be documented in a forthcoming revision to the PRA.

c) Reactor vessel insulation functional requirements important for in-vessel retention will be added to the SSAR.
d) The important containment component capacities to meet the ASME code are verified in ITAAC 2.2.1.

SSAR Revision:

A new subsection 5.3.5 will be added to SSAR, Revision 14, as follows.

In the unlikely event of a beyond design basis accident, the reactor vessel insulation does not interfere with heat removal from core debris via boiling on the outside surface of the reactor vessel. The functional requirements important to in-vessel retention are as follows:

a) A water inlet is provided at the bottom of the insulation. The water inlet is sized such that the pressure drop through the inlet is negligible during the circulation of water associated with the IVR phenomena.

b) The insulation provides a steam vent at the top of the biological shieldwall. The steam vent area is greater than or equal to the minimum flow area in the structures forming the circulation loop (not including the insulation itself). The minimum flow area is 7.5 ft^2.