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Docket No.: STN-52-003

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of April 15, 1994, April 29, 1994, May 2, 1994, May 5, 1994, May 18, 1994, May 23, 1994, May 24, 1994, May 26, 1994, and June 8, 1994. In addition, revisions of 12 responses previously submitted are provided.

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

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

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

Brian A. McIntyre for

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

/nja

Enclosure

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

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NTD-NRC-94-4257
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED AUGUST 8 1994

| RAI No. | Issue |
|---|---|
| 220.090R01 | Type and charactreistics of water seals |
| 230.016R01 | Axisymmetric containment shell model |
| 230.053R01 | Inclusion of other site conditions |
| 230.062R01 | Validity of fixed base seismic analysis |
| 230.078R01 | Time discretization of ground motion time hist. |
| 230.080R01 | Adequacy of using only 3 soil site conditions |
| 230.084R01 | Additional information in Section 3.7.2.1.1 |
| 230.090R01 | Dynamic model usage |
| 231.001R01 | Site-specific soil interaction analyses |
| 231.022R01 | SSI studies for the rock model |
| 260.026 | Pre-op tests for first plant only |
| 410.108 | SWS tests and inspections |
| 410.109 | SWS criteria |
| 410.115 | SWS potential for water hammer |
| 410.139 | TG functional limitations imposed by RCS |
| 410.143 | TG design criteria |
| 410.171 | DID classification of DG suppor systems |
| 410.188 | SFS safety grade concerns |
| 410.255 | Permissible cooling water leakage |
| 410.257 | Quality group classification for MCES |
| 410.258 | SSAR section 10.4.3.2.1 |
| 410.259 | Turbine steam sealing system |
| 410.260 | Aux. steam system necessary controls/indicators |
| 435.076 | Regulatory oversight of full load rejection cap. |
| 440.057 | EPGs for shutdown and mid-loop operations |
| 440.066 | Applicable modes for SSAR safety analyses |

NTD-NRC-94-4257
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED AUGUST 8 1994

| RAI No. | | Issue |
|---|---|---|
| 440.090 | : | PXS component parameters |
| 440.121 | : | Emergency procedures for feed & bleed operation |
| 440.146 | : | WCOBRA/TRAC |
| 440.147 | : | WCOBRA/TRAC |
| 440.149 | : | WCOBRA/TRAC |
| 440.150 | : | WCOBRA/TRAC |
| 440.151 | : | WCOBRA/TRAC |
| 440.152 | : | WCOBRA/TRAC |
| 440.153 | : | WCOBRA/TRAC |
| 440.156 | : | WCOBRA/TRAC |
| 440.176 | : | Methods to identify adverse systems interactions |
| 440.202 | : | Credit for break isolation in PRHR break tree |
| 440.203 | : | Makeup water needed for extended PRHRS operation |
| 440.208 | : | Boron precipitation for large break LOCA |
| 440.221 | : | Radiolytic gas impact on PRHRS operation |
| 480.004R01 | : | HWRF Test Data |
| 952.027R01 | : | RCS behavior versus ADS test facility |
| 952.075 | : | SPES-2 SG recirculation ratios |

Question 220.90

For the nuclear island structures, provide, in the SSAR, the type and characteristics of water seals to be used at the penetrations (mechanical and electrical) and accesses located below the flood level for preventing and mitigating the external flooding effects.

Response: (Revision 1)

SSAR Chapter 2 and Subsection 3.4.1.1.1 establish the maximum flood at less than the finished grade. As stated in the response to RAI 410.38, the AP600 design minimizes the number of penetrations through the walls below grade. There are no electrical penetrations through the exterior walls below grade. Those few process piping penetrations located below the maximum flood level (elevation 100') will be watertight. Any process piping penetrations ~~and electrical conduits~~ through the exterior walls below grade will be embedded in the wall or will be welded to a steel sleeve embedded in the wall. There are no accesses or tunnels penetrating the exterior walls below grade.

SSAR Revision:

Add the following at the end of Subsection 3.4.1.1.1:

There are no electrical penetrations through the exterior walls below grade. Process piping penetrations through the exterior walls below grade are embedded in the wall or are welded to a steel sleeve embedded in the wall. There are no access openings or tunnels penetrating the exterior walls below grade.

Question 230.16

Describe the method used to construct a stick model from the axisymmetric shell model of the containment vessel (Section 3.7.2.3.2).

Response: (Revision 1)

The method used to construct the stick model for the containment vessel is described in the SSAR revision shown below.

SSAR Revision:

Revise Subsection 3.7.2.3.2 as follows:

The steel containment vessel is a freestanding, cylindrical, steel shell structure with ellipsoidal upper and lower steel domes. The three-dimensional, lumped-mass stick model of the steel containment vessel is developed based on the axisymmetric shell model. Figure 3.7.2-5 presents the steel containment vessel stick model. In the stick model, the properties are calculated as follows:

- Members representing the cylindrical portion are based on the properties of the actual circular cross section of the containment vessel.
- Members representing the bottom head are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in vertical and horizontal directions.
- Shear, bending and torsional properties for members representing the top head are based on the average of the properties at the successive nodes, using the actual circular cross section. These are the properties that affect the horizontal modes. Axial properties, which affect the vertical modes, are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in the vertical direction.

This method used to construct a stick model from the axisymmetric shell model of the containment vessel is verified by comparison of the natural frequencies determined from the stick model and the shell of revolution model as shown in Table 3.7.2-14. The shell of revolution vertical model ($n = 0$ harmonic) has a series of local shell modes of the top head between 23 and 30 Hertz. These modes are predominantly in a direction normal to the shell surface and cannot be represented by a stick model. These local modes have small contribution to the total response to a vertical earthquake as they are at a high frequency where seismic excitation is small.

Table 3.7.2-14

**Comparison of frequencies for containment vessel seismic model**

| MODE NO. | VERTICAL MODEL | | HORIZONTAL MODEL | |
|---|---|---|---|---|
| | Shell of Revolution Model | Stick Model | Shell of Revolution Model | Stick Model |
| 1 | 17.71 Hertz | 18.33 Hertz | 7.39 Hertz | 7.56 Hertz |
| 2 | 23.59 Hertz | 30.06 Hertz | 20.88 Hertz | 22.0 Hertz |

Question 230.53

According to the SSAR, only three soil conditions (shear wave velocity equal to 1000 ft/sec, 2500 ft/sec and 8000 ft/sec) were used in the seismic design of AP600 standard plant. Provide justification for not including the site conditions with other shear wave velocities, such as 1500 ft/sec and 3500 ft/sec and different depths from grade to bedrock.

Response: (Revision 1)

The AP600 design soil profiles used in the design of seismic Category I structures, components, and seismic subsystems are derived from a set of generic soil profiles encompassing a wide range of soil parametric variations that bound most existing nuclear power plant sites as shown in Appendix 2A.1 of the SSAR. The design soil profiles are expected to bound site conditions with other shear wave velocities, such as 1500 ft/sec and 3500 ft/sec and different depths from grade to bedrock.

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

SSAR Revision: NONE

Question 230.62

Justify the validity of performing a fixed-base seismic analysis for the site conditions with shear wave velocity equal to or greater than 8000 ft/sec.

Response: (Revision 1)

See response to RAI 230.35 (e).

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

SSAR Revision: NONE

Question 230.78

Expand, in the SSAR, the descriptions of the three components (H1, H2 and H3) of the ground motion time histories used in the analyses to include indications of the time discretization (Nyquist frequency) being used and the appropriateness of this time step for the frequency ranges of interest.

Response: (Revision 1)

SSAR Section 3.7.1.2 is revised to provide the additional information.

~~The ground motion time histories (H1, H2, and V) are generated with time step size of 0.010 second for applications in soil structure interaction analyses. For applications in the fixed-base mode superposition time-history analyses, the time step size is reduced to 0.005 second by linear interpolation. The maximum "cut-off" frequency for both the soil structure interaction analyses and the fixed-base analyses is 34 Hz, which is well within the Nyquist frequency limit.~~

SSAR Revision:

Revise the first paragraph of Subsection 3.7.1.2 as shown below:

A "single" set of three mutually orthogonal, statistically independent, synthetic acceleration time histories is used as the input in the dynamic analysis of seismic Category I structures. The synthetic time histories were generated by modifying a set of actual recorded "TAFT" earthquake time histories. The design time histories include a total time duration equal to 20 seconds and a corresponding stationary phase, strong motion duration greater than 6 seconds. The acceleration, velocity, and displacement time-history plots for the three orthogonal earthquake components, "H1", "H2" and "V", are presented in Figures 3.7.1-3, 3.7.1-4, and 3.7.1-5. The ground motion time histories (H1, H2, and V) are generated with time step size of 0.010 second for applications in soil structure interaction analyses. For applications in the fixed-base mode superposition time-history analyses, the time step size is reduced to 0.005 second by linear interpolation. The cutoff frequency used in the horizontal and vertical seismic analysis of the nuclear island for the hard rock site is 34 hertz. The cutoff frequencies used in the soil structure interaction analyses are 33 hertz for the soft rock site, and 15 hertz and 21 hertz for the soft-to-medium stiff soil site in the horizontal and vertical directions, respectively. The maximum "cut-off" frequency for both the soil structure interaction analyses and the fixed-base analyses is well within the Nyquist frequency limit.

Question 230.80

Subsection 3.7.2.1 of the SSAR (pg. 3.7.2-3) indicates that separate seismic analyses are performed for the nuclear island (NI) for each of the soil profiles defined in Section 3.7.1.4 and the three sets of in-structure seismic responses are enveloped to obtain the seismic design envelope (design member forces, nodal accelerations, nodal displacements, and floor response spectra) used in the design and analysis of seismic Category I structures, components, and seismic subsystems. The staff is concerned that the seismic design of the structures, systems and components of the AP600 standard plant may not be sufficient because it considers only three generic site conditions characterized with soil shear wave velocities that are far apart from each other. An example of the staff's concern is shown in the floor response spectra (FRS) plots of Figure 3.7.2-25. As shown in these plots, the horizontal (EW component in particular) FRS envelope in the control room area may not cover the FRS from two possible intermediate site conditions, one with a shear wave velocity between 1000 ft/sec and 2400 ft/sec (approximately 1500 ft/sec) and the other with a shear wave velocity between 2400 ft/sec and 8000 ft/sec (approximately 3500 ft/sec). Justify the adequacy of using only three generic site conditions for the AP600 standard plant design.

Response:

As discussed in Appendix 2A of the SSAR, the AP600 design soil profiles were derived by considering a wide range of soil parametric variations that bound most of the existing nuclear power plant sites. Therefore, the three design soil profiles are adequate for the AP600 standard plant design.

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

SSAR Revision: NONE

Question 230.84

In Section 3.7.2.1.1 of the SSAR, provide information and explanation to demonstrate:

a. that for site conditions with soil shear wave velocity equal or greater than 8000 ft/sec, the SSI effects between NI structures and soil foundation are negligible and the use of fixed base models is adequate for calculating seismic responses of the NI structures, and

b. that for structures with multimodes, the amplification procedures, described for cases where the responses of soil founded structures exceed the responses of rock (shear wave velocity equal or greater than 8000 ft/sec) founded structures, will provide reasonable results for the design of structures, systems and components.

Response: (Revision 1)

See response to RAI 230.35 (e).

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

SSAR Revision: NONE

Question 230.90

Because many different dynamic models (3D stick model, 3D finite element model, 3D stick model coupled with 3D finite element soil foundation model, etc.) and analysis methods (response spectrum analysis method using SAP computer code, time history analysis method using SAP computer code, time history analysis method using SASSI computer code, etc.) were used for the seismic analyses of NI structures, provide a detailed description in the SSAR to show which model combined with which analysis method was used for generating what kind of dynamic responses for the design.

Response: (Revision 1)

SSAR Subsection 3.7.2 is revised as shown below to provide the requested information.

SSAR Revision:

Add the following paragraph at the end of Subsection 3.7.2:

Table 3.7.2-14 summarizes the types of models and analysis methods that are used in the seismic analyses of the nuclear island. It also summarizes the type of results that are obtained and where they are used in the design.

Add the attached Table 3.7.2-14 to Section 3.7.2.

Table 3.7.2-14

**Models - Analysis Methods - Types of Dynamic Responses Generated**

| Model | Analysis Method | Program | Type of Dynamic Response/Purpose |
|---|---|---|---|
| 2D lumped mass stick models coupled with 2D model of the foundation | Complex frequency response analysis | SASSI | To identify governing site properties and design soil profiles. |
| 2D lumped mass stick, fixed base models | Mode superposition time history analysis | BSAP | |
| 3D lumped mass stick, fixed base models | Mode superposition time history analysis | BSAP | Performed for hard rock profile.<br>To develop time histories for generating floor response spectra.<br>To obtain the following:<br>Maximum absolute nodal accelerations (ZPA).<br>Maximum displacements relative to basemat.<br>Maximum member forces and moments for all structures, except the containment internal structures. |
| | Response spectrum analysis | BSAP | To obtain the seismic force and moment response of the containment internal structures (Subsection 3.7.2.2) including the high frequency modal effect.<br><br>Used also to determine governing responses for design of structures. |
| 3D lumped mass stick models coupled with 3D model of the foundation | Complex frequency response analysis | SASSI | Performed for the soft rock and soft-to-medium soil profiles.<br>To develop time histories for generating floor response spectra.<br>To obtain the following:<br>Maximum absolute nodal accelerations (ZPA).<br>Maximum displacements relative to basemat.<br>Maximum member forces and moments.<br><br>To determine governing responses for design of structures. |

Table 3.7.2-14

## Models - Analysis Methods - Types of Dynamic Responses Generated

| Model | Analysis Method | Program | Type of Dynamic Response/Purpose |
|---|---|---|---|
| 2D lumped mass stick models coupled with 2D model of the foundation | Complex frequency response analysis | SASSI | To identify governing site properties and design soil profiles. |
| 2D lumped mass stick, fixed base models | Mode superposition time history analysis | BSAP | |
| 3D finite element, fixed base models, coupled Auxiliarly/Shield buildings and Containment internal structures | Response spectrum analysis | BSAP | Performed for the hard rock profile. To obtain the in-plane forces[1] for the design of floors and walls. |
| 3D shell of revolution model of steel containment vessel | Equivalent static analysis using nodal accelerations from 3D stick model | | Obtain SSE Stress for the containment vessel |

[1] The in-plane forces for the hard rock profiles are increased by a scaling factor when, based on a comparison of force responses of the 3D lumped-mass stick model, either the soft rock or soft-to-medium stiff soil cases give higher element forces than the hard rock case. The scaling factor, at a given plant elevation, is equal to the ratio of the largest 3D stick model element forces over the 3D stick model element force for the hard rock case.

Question 231.1

Clarify the following statement in Section 2.5 (p. 2.0-3): "For the site where the soil characteristics differ significantly...site-specific soil structure interaction analyses may be performed to demonstrate acceptability..." Referring to Section 2A.6 of the SSAR in which the base rock depth of design soil profiles was specified at 37 m (120 ft), will site-specific seismic analyses be required if the site base rock depth is, for example, 46 m (150 ft), which is deeper than the 37 m (120 ft) condition analyzed? See staff comment on Section 3.7.1.4 of the SSAR (Q230.10).

Response: (Revision 1)

Section 2.5 of the SSAR presents the site interface criteria and requirements of potential AP600 plant sites. The site qualification flow chart is shown in Figure 2.5-1, attached to this response.

The AP600 design soil profiles used in the design of seismic Category I structures, components, and seismic subsystems are derived from a set of generic soil profiles encompassing a wide range of soil parametric variations that bound most existing nuclear power plant sites shown in Appendix 2A.1.

The soil structure interaction (SSI) analyses of the generic soil profiles described in Appendix 2A encompass a wide range of site parameters, including a wide range of depth to base rock. Based on the SSI results, 120 feet depth to base rock is selected because it induces a larger nuclear island seismic response than other depths considered. Therefore, it would not be necessary to perform a site-specific analysis if the site base rock depth is 150 feet. Similar SSI analyses were performed and evaluated for other site parameters. The AP600 design soil profiles were derived from the combination of all soil parameters that produce the bounding SSI response on the nuclear island.

For this reason, proposed plant sites with properties within the range of parameters considered for the generic SSI analyses are qualified for siting the AP600 plant. For proposed plant sites with properties outside the range of parameters considered, the necessity for site-specific analysis by the combined license applicant will be established according to the site interface criteria presented in SSAR Section 2.5.

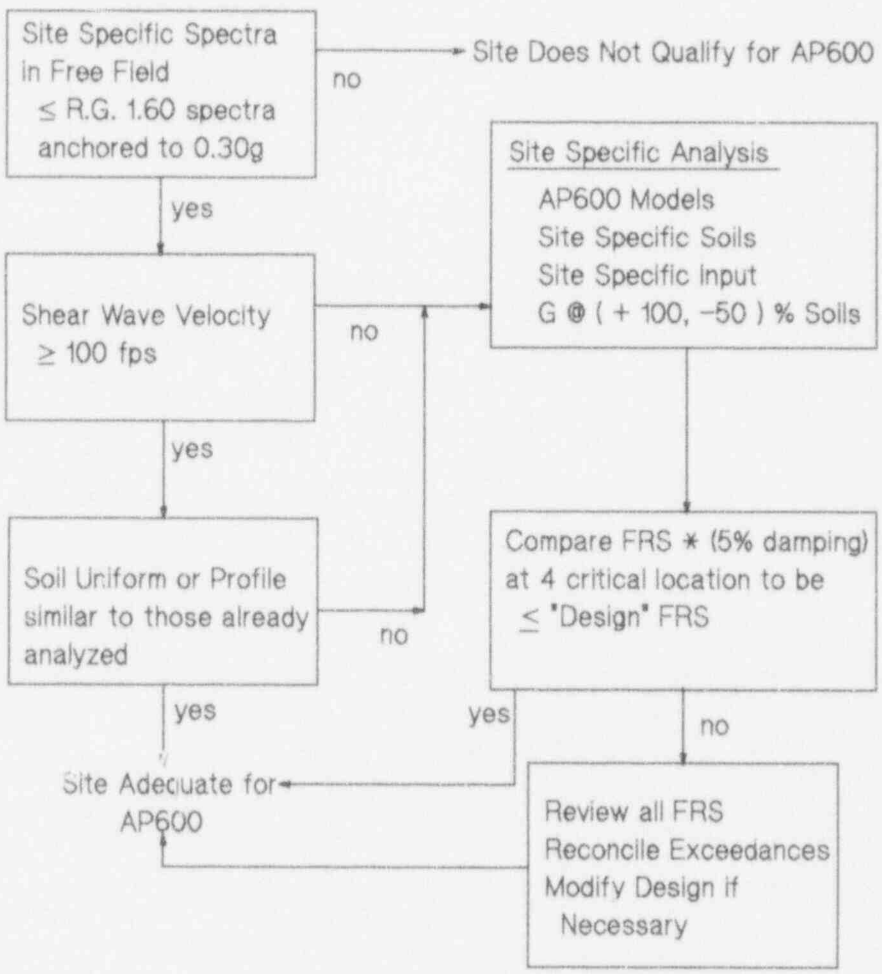
SSAR Revision:

SSAR Section 2.5 will be revised as follows: Note - this revision was incorporated into SSAR Revision 1

For sites where the soil characteristics are outside the range considered in Appendix 2A.2, site-specific soil structure interaction analyses may be performed by the Combined License applicant to demonstrate acceptability by comparison of floor response spectra at the following locations. Comparison of the floor response spectra at these locations is sufficient demonstration that the site seismic conditions are within the AP600 design basis.

Replace figure 2.5-1 with the attached revision.

## Site Qualification for AP600




* Raw FRS - Comparison acceptable with up to five
Exceedances at no greater than 10%

Figure 2.5-1 Site Qualification Flow Chart

Question 231.22

In the January 22, 1993 response to Q231.5 regarding the assumption of an upper bound value of 8000 feet per second (fps) for the shear wave velocity of the hard rock site, Westinghouse states that, for the hard rock site profile with shear wave velocity of 8000 fps (greater than 3500 fps), the nuclear island is analyzed as a fixed base structure. However, the decision to use fixed base analyses should not be based on a specified rock shear wave velocity, but on the relationship between the SSI frequency and the structural frequency of the NI.

Perform necessary SSI studies for the rock model (with rock shear wave velocity ranging from 8000 to 11000 fps discussed in Q231.5 to justify the use of fixed base analysis for the rock site with shear wave velocity of 8000 fps.

Response: (Revision 1)

See response to RAI 230.35 (e).

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

SSAR Revision: NONE

Question 260.26

Section 14.2.1, "Summary of Test Program and Objectives," Section 14.2.4, "Utilization of Reactor Operating and Testing Experience in the Development of Test Program," Section 14.2.6.3, "Power Ascension," and Section 14.2.8 of the SSAR, "Individual Test Descriptions" describe 9 preoperational tests and 8 startup tests that are to be performed only for the first AP600 plant to confirm selected design and analysis assumptions and predictions. The justification provided in Section 14.2.1 states "Because of the standardized AP600 design, it is not necessary to repeat these tests during the initial test programs for successive AP600 plants. There is no need to reconfirm the design and analysis assumptions for the successive AP600 plants."

Revise Section 14.2 to either provide a specific listing of the exceptions to corresponding Regulatory Guides (RGs) with appropriate technical justification for conducting each of these tests only on the first plant, or to commit to performing these tests on all AP600 plants. The following is a list of the tests and the corresponding RGs of concern:

- Preoperational test abstract 14.2.8.1.77, "Reactor Internals and Reactor Coolant System Vibration Test"; RG 1.20, "Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational And Initial Startup Testing," RP C.3.4. Non-Prototype, Category IV(sic).
- Preoperational test abstract 14.2.8.1.78, "Steady-State Vibration Monitoring of Safety-Related and High-Energy Piping"; RG 1.68, Appendix A, Item 1.a.(3).
- Preoperational test abstract 14.2.8.1.80, "Automatic Depressurization System"; RG 1.68, Appendix A, Item 1.a.(2)(d).
- Preoperational test abstract 14.2.8.1.82, "Dynamic Response"; RG 1.68, Appendix A, Items 1.a.(1) and (3).
- Preoperational test abstract 14.2.8.1.85, "Passive Core Cooling System"; RG 1.79, Preoperational Testing of Emergency Core Cooling Systems For Pressurized Water Reactors" and RG 1.68, Appendix A, Item 1.h.
- Preoperational test abstract 14.2.8.1.87, "Passive Residual Heat Removal System"; RG 1.139, "Guidance For Residual Heat Removal," RP C.5; and RG 1.68, Appendix A, Items 1.d.(5), and 1.d.(8).
- Preoperational test abstract 14.2.8.1.94, "Remote Shutdown"; RG 1.68.2, "Initial Startup Test Program To Demonstrate Remote Shutdown Capability For Water-Cooled Nuclear Power Plants," RP C.3 and 4.
- Preoperational test abstract 14.2.8.1.97, "Passive Containment Cooling System"; RG 1.68, Appendix A, Item 1.h.(3).
- Preoperational test abstract 14.2.8.1.100, "Main Control Room Habitability System"; RG 1.68, Appendix A, Item 1.n.(14)(f).
- Startup test abstract 14.2.8.2.20, "Dynamic Response"; RG 1.68, Appendix A, Item 5.o.o.

- Startup test abstract 14.2.8.2.34, "Natural Circulation"; RG 1.68, Appendix A, Item 4.t.

- Startup test abstract 14.2.8.2.38, "Process Measurement Accuracy Verification"; RG 1.68, Appendix A, Items 5.b and y.

- Startup test abstract 14.2.8.2.41, "Loss of Offsite Power"; RG 1.68, Appendix A, Item 5.j.j.

- Startup test abstract 14.2.8.2.47, "Rod Cluster Control Assembly Out of Bank Measurements"; RG 1.68, Appendix A, Items 4.e, 5.f, and 5.i.

- Startup test abstract 14.2.8.2.51, "100 Percent Load Rejection"; RG 1.68, Appendix A, Item 5.n.n.

- Startup test abstract 14.2.8.2.52, "Load Follow Demonstration"; RG 1.68, Appendix A, Item 5.h.h.

- Startup test abstract 14.2.8.2.55, "Plant Trip from 100 Percent Power"; RG 1.68, Appendix A, Item 5.l.l.

Response: The response to each testing abstract follows:

- Preoperational test abstract 14.2.8.1.77, "Reactor Internals and Reactor Coolant System Vibration Test"; RG 1.20, "Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational And Initial Startup Testing," RP C.3.4, Non-Prototype, Category II.

  This test, involving vibration measuring instrumentation installed for monitoring system and component vibration, is conducted for the first AP600. An inspection program as defined in Section 3.1.3 of RG 1.20 is also conducted on the first AP600. This provides the basis for performing an inspection program (Section 3.4.3 of RG 1.20) for succeeding AP600 plants. The inspections for these plants are implemented under Hot Functional Testing, subsection 14.2.8.1.67.

  This abstract will not be amended to include all plants.

- Preoperational test abstract 14.2.8.1.78, "Steady-State Vibration Monitoring of Safety-Related and High-Energy Piping"; RG 1.68, Appendix A, Item 1.a.(3).

  This abstract will be modified to be performed on each startup per the response to RAI210.53.

- Preoperational test abstract 14.2.8.1.80, "Automatic Depressurization System"; RG 1.68, Appendix A, Item 1.a.(2)(d).

  This test, performed on the first plant system, proves the depressurization capability of the Automatic Depressurization system. Each succeeding standard AP600 automatic depressurization system design will meet or exceed the plant depressurization criteria as

demonstrated by satisfying their appropriate ITAAC. Component functionality is proven by performing testing identified in abstract 14.2.8.1.79.

This abstract will not be amended to include all plants.

- Preoperational test abstract 14.2.8.1.82, "Dynamic Response"; RG 1.68, Appendix A, Items 1.a.(1) and (3).

  This abstract will be modified to be performed on each startup per the response to RAI210.53.

- Preoperational test abstract 14.2.8.1.85, "Passive Core Cooling System"; RG 1.79, Preoperational Testing of Emergency Core Cooling Systems For Pressurized Water Reactors" and RG 1.68, Appendix A, Item 1.h.

  This test, performed on the first plant system, proves the thermodynamic capabilities of the Passive Core Cooling (PXS) system, i.e., to meet or exceed performance criteria.

  Each succeeding standard AP600 PXS will meet or exceed the performance criteria as demonstrated by satisfying their appropriate ITAAC. Component functionality is proven by performing testing identified in abstract 14.2.8.1.84.

  This abstract will not be amended to include all plants.

- Preoperational test abstract 14.2.8.1.87, "Passive Residual Heat Removal System"; RG 1.139, "Guidance For Residual Heat Removal," RP C.5; and RG 1.68, Appendix A, Items 1.d.(5), and 1.d.(8).

  This test, performed on the first plant, proves the thermodynamic capabilities of the PRHR system; i.e., to meet or exceed performance criteria.

  Each succeeding AP600 PRHR subsystem will meet or exceed the performance criteria as demonstrated by satisfying their appropriate ITAAC. Component functionality is proven by performing testing identified in abstract 14.2.8.1.86, Passive Residual Heat Removal system.

  This abstract will not be amended to include all plants.

- Preoperational test abstract 14.2.8.1.94, "Remote Shutdown"; RG 1.68.2, "Initial Startup Test Program To Demonstrate Remote Shutdown Capability For Water-Cooled Nuclear Power Plants," RP C.3 and 4.

  This test, performed on the first plant, proves the remote shutdown design capability. The test proves remote manual control of the required valves and functions to effect cool down from hot standby and cooldown to safe shutdown conditions with safety related systems.

  For each succeeding standard AP600 remote shutdown work station, control functionality is proven by testing of components (switches) in systems required to be checked per plant technical specifications and as permitted by RG 1.68.2 RP C.3 and C4.

  This abstract will not be amended to include all plants.

- Preoperational test abstract 14.2.8.1.97, "Passive Containment Cooling System"; RG 1.68, Appendix A, Item 1.h.(3).

  This test is performed on the first plant and proves the thermodynamic capabilities of the Passive Containment Cooling System (PCS); i.e., to meet or exceed the performance criteria.

  Each succeeding AP600 PCS will meet or exceed the performance criteria, as demonstrated by satisfying their appropriate ITAAC . Component functionality is proven by performing testing identified in abstract 14.2.8.1.96.

  This abstract will not be amended to include all plants.

- Preoperational test abstract 14.2.8.1.100, "Main Control Room Habitability System"; RG 1.68, Appendix A, Item 1.n.(14)(f).

  This test, performed on the first plant, proves the main control room habitability design by extensive testing and control of environmental conditions.

  Each succeeding standard AP600 will meet the habitability criteria as demonstrated by satisfying their appropriate ITAAC. Component functionality (control room leak rate) is proven by conducting testing for each succeeding plant per testing identified in abstract 14.2.8.1.99.

  This abstract will not be amended to include all plants.

- Startup test abstract 14.2.8.2.20, "Dynamic Response"; RG 1.68, Appendix A, Item 5.o.o.

  This abstract will be modified to be performed on each startup per the response to RAI210.53.

- Startup test abstract 14.2.8.2.34, "Natural Circulation"; RG 1.68, Appendix A, Item 4.t.

  Exception to RG 1.68 is taken for this test as has been done for current PWR plants. The justification for this exception is that the performance of a natural circulation test is not necessary to demonstrate flow characteristics of the plant. The physical layout of the plant and key components (steam generators, pumps, piping, and reactor vessel) is identical for each unit. Typical manufacturing and construction variations in these parameters will have no significant impact on the natural circulation flow. Since the design and layout is fixed between each AP600 plant, no changes in the natural circulation characteristics will occur. Other system flow and performance measurements taken during the hot functional and power ascension testing provide assurances that the overall flow characteristics of the plant are equivalent to the reference plant. Therefore, demonstration of the natural circulation characteristics on the first AP600 plant is sufficient to validate the design characteristics. The Natural Circulation test is prototypical.

- Startup test abstract 14.2.8.2.38, "Process Measurement Accuracy Verification"; RG 1.68, Appendix A, Items 5.b and y.

  The purpose of this test is to confirm that excore detector uncertainties are enveloped by assumptions in Safety Analysis. The results of this test will apply to each succeeding standard AP600.

- Startup test abstract 14.2.8.2.41, "Loss of Offsite Power"; RG 1.68, Appendix A, Item 5.j.j.

  This test is performed to prove the electrical systems' design response to loss of offsite power; i.e., to meet or exceed the performance criteria.

  Each succeeding standard AP600 will meet the criteria as demonstrated by satisfying their appropriate ITAAC.

- Startup test abstract 14.2.8.2.47, "Rod Cluster Control Assembly Out of Bank Measurements"; RG 1.68, Appendix A, Items 4.e (5.e), 5.f, and 5.i.

  The purpose of this test is to prove core and instrumentation design meets the performance criteria of rod misalignment per the requirements of Items 5.e and 5.f. Furthermore, the sensitivity of the nuclear instrumentation to rod misalignments will be demonstrated per the requirements of 5.i. The results of this test will apply to each succeeding standard AP600.

- Startup test abstract 14.2.8.2.51, "100 Percent Load Rejection"; RG 1.68, Appendix A, Item 5.n.n.

  The purpose of this test is to prove the plant design dynamic response to 100% load rejection by the test method of Items 5.n.n. The results of this test will apply to each succeeding standard AP600.

- Startup test abstract 14.2.8.2.52, "Load Follow Demonstration"; RG 1.68, Appendix A, Item 5.h.h.

  The purpose of this test is to prove the plant design performance during load follow maneuvers. The results of this test will apply to each succeeding standard AP600.

- Startup test abstract 14.2.8.2.55, "Plant Trip from 100 Percent Power"; RG 1.68, Appendix A, Item 5.l.l.

  The purpose of this test is to prove the plant design performance following plant trip. The results of this test will apply to each succeeding standard AP600.

SSAR Revision: As noted above in referenced RAI responses

Question 410.108

Section 9.2.1.4 of the SSAR, "Tests and Inspections," states that the performance, structure, and leaktight integrity of system components is demonstrated by operation of the system. It appears to the staff that the "operation of the system" does not provide any commitment for operational tests or inspections. However, Section C14.2.3 and Table C14-5 of the PRA states that it is **assumed** that most of components (pumps, heat exchangers, and valves involved in train operation change) are tested quarterly, and other components are assumed to be tested every 24 months at plant shutdown for refueling.

Confirm that the tests assumed for the PRA are a design requirement for COL applicants to carry out. If that is the case, revise Section 9.2.1.4 of the SSAR to delineate the testing and inspection program. In addition, because there are no technical specifications for the SWS, where will the test requirements in terms of the test frequency and acceptance criteria be specified?

Response:

The service water system provides no safety-related function to the AP600. The system does not require any testing or inspection plan. As stated in Section C14.2.3 of the PRA, it is "assumed that most of train components are tested quarterly because of the periodic change in train operation when the standby train is manually actuated and the other one is stopped." Operational testing of service water system (SWS) components through switch over of pumps, heat exchangers, fans and associated components is consistent with the SSAR statement that the performance of system components is demonstrated by operation of the system. However, there is no specific design requirement regarding frequency of operation or testing of SWS components. The SWS is a defense-in-depth system and is considered to be available in the probabilistic risk assessment. As discussed in SSAR Subsection 3.2.2.6, the reliability and maintenance plans for such systems include provisions to check for operability, including appropriate testing and inspection, and to repair out-of-service structures, systems, and components. These provisions are documented and administered in the plant reliability assurance plan and operating and maintenance procedures.

The SWS function to supply cooling flow to the component cooling water system heat exchangers during reduced coolant inventory operations is identified in Reference 410.108-1 as an RTNSS-significant function. Reference 410.108-1 also provides short-term availability recommendations for the equipment used to support this function.

Reference:

410.108-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

AP600

PRA Revision: Revise the first paragraph of Subsection C14.2.3, Test and Maintenance

There are no technical specification for this system. However, it is assumed that most of train components are ~~tested~~operated at least quarterly because of the periodic change in train operation when the standby train is manually actuated and the other one is stopped.

Question 410.109

In WCAP-13856, the AP600 RTNSS evaluation identified that the service water system (SWS) provides defense-in-depth functions during shutdown when in reduced inventory operations. Demonstrate that the following criteria are met by the system, or justify the deviation, if any.

a. Does the SWS have an electric supply from both normal station ac and on-site non-safety-related ac power supplies that is separated, to the extent practicable?

b. Is the SWS designed and arranged for conditions or an environment anticipated during and after events to ensure functional operability, maintenance accessibility, and plant recovery?

c. Is the SWS protected against internal flooding and other in-plant hazards, such as the effects of pipe ruptures, jet impingement, fires, and missiles?

d. Can the SWS withstand the effects of natural phenomena that have a reasonable likelihood? Important systems and components should be designed to remain functional after a natural phenomena, such as a seismic event, that is of reasonable likelihood or may persist longer than 72 hours.

e. Is there a quality assurance program applied to the SWS that follows guidelines comparable to those of Generic Letter 85-06 for ATWS, and Appendices A and B of Regulatory Guide 1.155, "Station Blackout," for station blackout non-safety-related equipment?

f. Is the SWS included in the reliability assurance and maintenance programs for proper maintenance, surveillance, and inservice inspection and testing to ensure the system's reliability is consistent with the determined goals for this system?

g. Does the SWS have availability control mechanisms, including allowable outage time and surveillance requirements?

h. Does the SWS have proper administrative controls for shutdown configurations?

i. Does the SWS have sufficient redundancy to ensure defense-in-depth functions, assuming a single active failure of equipment or unavailability due to maintenance.

Response:

The service water system (SWS) performs no safety-related functions and need not meet the listed criteria which are applicable to safety-related systems. However, the following provides service water design information in response to the listed requests.

a. Each of the two service water system (SWS) pumps and associated active components are supplied from independent, permanent, nonsafety-related electrical buses. Each bus is capable of being supplied from one of two onsite standby diesel generators. The onsite standby diesel generators and permanent onsite 4160 volt buses are physically separated. No separation is required or provided for the 480 volt load centers or the supply cables for the system electrical loads.

b. The design of the SWS does not ensure functional operability, maintenance access or support plant recovery following design basis events. Maintenance accessibility is provided consistent with the pump nonsafety-related functions and plant availability goals.

c. Protection from internal hazards is neither required or provided for the SWS.

d. The SWS is not protected from natural phenomenon and is not required to remain functional after a natural phenomenon. There is no requirement for SWS functionality after 72 hours following an event.

e. As a defense-in-depth system, the SWS is classified as an AP600 Class D system. As discussed in SSAR Subsection 3.2.2.6, this classification invokes industrial quality assurance and industry design standards.

f. The extent of SWS inclusion in reliability assurance and maintenance programs is discussed in SSAR Subsection 3.2.2.6 for Class D structures, systems and components. The Reliability Assurance Program is further described in SSAR Section 16.2 and includes a discussion of the applicability to the nonsafety-related defense-in-depth systems, which includes the SWS.

g. The SWS does not have technical specification availability control mechanisms (i.e., limiting conditions for operation) nor allowable outage times or surveillance requirements. This system is not safety-related and not required for plant shutdown, and therefore not required to have technical specifications. The SWS function of supplying cooling flow to the component cooling water system heat exchangers during reduced reactor coolant system inventory, midloop operations, is identified as an RTNSS-significant function in Reference 410.109-1. This reference also provides short-term availability recommendations for the equipment used to support this function.

h. Reference 410.109-1 provides recommended availability controls for those portions of the SWS that perform RTNSS-significant functions during reduced reactor coolant inventory operations.

i. Appropriate redundancy is provided such that the SWS can support normal operation and defense-in-depth functions assuming a single active component failure.

Reference:

410.109-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

Question 410.115

Describe how AP600 SWS is designed to minimize the potential for water hammer.

Response:

Prior to system startup, the service water system (SWS) is filled with water and vented as described in SSAR Subsection 9.2.1.3.1. During normal operation, the SWS pumps water from the basin at the SWS cooling tower, through piping and equipment, to a high point located at the SWS cooling tower riser; the cooling water is then discharged to atmosphere in a spray fashion above the cooling tower basin. The system arrangement is such that there are no high points in the system piping that can lead to formation of vapor pressure voids upon loss of system pumping.

When the pumps are stopped, check valves located at the discharge of each pump minimize reverse flow of system fluid through the pumps and into the basin; also, cooling tower blowdown is isolated when the pumps are stopped. Therefore, drain down of system fluid is minimized when the system is shut down. Drainage that might occur, such as through the small SWS motor cooling lines, is replaced by air. No vapor cavities will form. Therefore, the potential for water hammer due to water column rejoining upon pump re-start is minimized.

Motor operated valves at the discharge of each SWS pump are interlocked to close prior to pump start. These valves then open at a controlled rate following pump start to slowly admit water to the system. This feature results in reduced fluid velocities during system start and minimizes transient effects that may occur as the system sweeps out any air and obtains a water solid condition. Temperatures in the system are moderate and the pressure of the SWS fluid is kept above its saturation pressure at all locations. Therefore, the potential for water hammer due to thermodynamic voiding and subsequent vapor collapse is minimized. There are no fast acting power-operated valves in the system, and the only check valves in the normal process flow path are in a standard configuration at the discharge of each SWS pump. Therefore, the design of the system minimizes water hammer potential due to rapid valve actuation.

SSAR Revision: NONE

Question 410.139

Provide in the design bases of the turbine-generator section the information identified in RG 1.70 regarding the functional limitations imposed by the design or operational characteristics of the reactor coolant system (the rate at which the electrical load may be increased or decreased with and without reactor control rod motion or steam bypass).

Response:

SSAR Subsection 10.2.2.7 provides the constraints imposed on the AP600 turbine generator control system by referring to the loading and load following characteristics of the NSSS described in Section 7.7. Section 7.7 states that the function of the plant control system is to establish and maintain the plant operating conditions within prescribed limits. SSAR Subsection 7.2.1 provides the considerations that define the safe operating region of the plant. The AP600 reactor and generator controls are designed to keep plant parameters within the safe operating region.

SSAR Subsection 7.7.1 provides the specific ramp rates allowed by the power control system to enable the plant to respond to load change transients without steam bypass. This ramp rate is dependent on control rod motion. Control rod motion is limited by stops to prevent abnormal power conditions resulting from excessive control rod withdrawal. An automatic turbine load runback is initiated by overpower thermal conditions (Subsection 7.7.1.5). Subsection 7.7.2 lists reference transients that the system can accommodate in the automatic mode.

SSAR Revision: NONE

Question 410.143

WCAP-13054 indicates that the turbine-generator design for the AP600 design meets Criteria 1 through 7 of Section 10.2 of the SRP, with a few identified exceptions. Describe in Section 10.2 how each of the seven criteria is met.

Response:

Below is a discussion of how the AP600 turbine-generator design meets the seven Acceptance Criteria as listed in the Standard Review Plan, Section 10.2 - Turbine Generator (Rev. 2, 7/81):

Criterion 1:

*A turbine control and overspeed protection system should be provided to control turbine action under all normal and abnormal operating conditions, and to assure that a full load turbine trip will not cause the turbine to overspeed beyond acceptable limits. Under these conditions, the control and protection system should permit an orderly reactor shutdown either by use of the turbine bypass system and main steam relief or other engineered safety systems. The overspeed protection system should meet the single failure criterion and should be testable when the turbine is in operation.*

Compliance with Criterion 1:

The automatic turbine control system and the turbine overspeed protection system is described in SSAR Subsections 10.2.2.3, 10.2.2.4 and 10.2.2.5. Two separate systems have been provided to protect the turbine against overspeed - the normal overspeed protection system (which is integral to the speed control unit) and the emergency trip system. SSAR Table 10.2-2 provides a sequence of events following a full load rejection. As discussed in SSAR Subsection 10.4.4.1.2, the turbine bypass system, in conjunction with reactor coolant system, supports a turbine trip without a reactor trip and without challenging the main steam power-operated relief valves, main steam safety valves or pressurizer safety valves. Redundancy in the overspeed protection system and the ability of the overspeed protection system to be tested when the turbine is in operation are described in SSAR Subsections 10.2.2.3.6 and 10.2.2.5.

Criterion 2:

*Turbine main steam stop and control valves and reheat steam stop and intercept valves should be provided to protect the turbine from exceeding set speeds and to protect the reactor system from abnormal surges. The reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to assure that turbine overspeed is controlled within acceptable limits. The valve arrangement and valve closure times should be such that a failure in any single valve to close will not result in excessive turbine overspeed in the event of a TGS trip signal.*

Compliance with Criterion 2:

The main steam stop and control valves and reheat steam stop and intercept valves are described in SSAR Subsection 10.2.2.3.4. These valves protect the turbine from exceeding set speeds as discussed in SSAR Subsection 10.2.2.3.6. The closure times for the main steam stop and control valves and reheat steam stop and intercept valves are listed in SSAR Subsection 10.2.2.3.6. The sequence of valve operation is described in SSAR Table 10.2-2. Redundancies in the overspeed protection systems, as described in SSAR Subsection 10.2.2.3.6, ensure that the failure of a single valve will not disable the trip systems.

Criterion 3:

*The extraction steam check valves provided at extraction connections shall be capable of closing within an appropriate time limit to maintain stable turbine speeds in the event of a TGS trip signal.*

Compliance with Criterion 3:

As listed in SSAR Subsection 10.2.2.3.6, the extraction nonreturn valves close to prevent turbine overspeed.

Criterion 4:

*The TGS should be provided with the capability to permit periodic testing of components important to safety while the unit is operating at rated load.*

Compliance with Criterion 4:

The emergency trip system can be tested when the turbine is in operation as described in SSAR Subsection 10.2.2.5.

Criterion 5:

*An inservice inspection program for main steam and reheat valves should be provided and it should include...*

Compliance with Criterion 5:

Main steam and reheat valves have no safety-related function. The inservice inspection program for the main steam and reheat valves is described in SSAR Subsection 10.2.3.6 and is based on recommendations of the turbine vendor.

Criterion 6:

*Unlimited access to all levels of the turbine area under all operating conditions should be provided. Radiation shielding should be provided as necessary to permit access.*

Compliance with Criterion 6:

As stated in SSAR Subsection 10.2.4, unlimited access to the turbine generator and associated components is provided and radiation shielding is not required.

Criterion 7:

*Connection joints between the low pressure turbine exhaust and the main condenser should be arranged to prevent adverse effects on any safety-related equipment in the turbine room in the event of a rupture (it is preferable not to locate safety-related equipment in the turbine room).*

Compliance with Criterion 7:

As stated is SSAR Subsection 1.2.8, there is no safety-related equipment in the AP600 turbine building.

SSAR Revision: NONE

Question 410.171

In accordance with WCAP-13856, the onsite standby power system is classified as a "Defense-In-Depth (DID)" system. Therefore, the diesel generator support systems should also be classified as DID systems. Section 9.5.4 of the SSAR only discusses the diesel generator fuel oil storage and transfer system and Section 8.3.1.1.2.1 only lists the titles of all of the diesel generator support systems. Provide more detailed information in the SSAR on the following other diesel generator support systems:

a. diesel engine cooling subsystem,
b. diesel engine starting subsystem,
c. diesel engine lubrication subsystem, and
d. diesel engine combustion air intake and exhaust subsystem.

Provide the following information with appropriate justification to demonstrate that the criteria identified in the questions are met by these subsystems, or justify the deviation, if any.

a. Does the system have an electric supply from both normal station ac and on-site non-safety-related ac power supplies that is separated, to the extent practicable?

b. Is the system designed and arranged for conditions or an environment anticipated during and after events to ensure functional operability, maintenance accessibility, and plant recovery?

c. Is the system protected against internal flooding and other in-plant hazards, such as the effects of pipe ruptures, jet impingement, fires, and missiles?

d. Can the system withstand the effects of natural phenomena that have a reasonable likelihood? Important systems and components should be designed to remain functional after a natural phenomena, such as a seismic event, that is of reasonable likelihood or may persist longer than 72 hours.

e. Is there a quality assurance program applied to the system that follows guidelines comparable to those of Generic Letter 85-06 for ATWS, and Appendices A and B of Regulatory Guide 1.155, "Station Blackout," for station blackout non-safety-related equipment?

f. Is the system included in the reliability assurance and maintenance programs for proper maintenance, surveillance, and inservice inspection and testing to ensure the system's reliability is consistent with the determined goals for this system?

g. Does the system have availability control mechanisms, including allowable outage time and surveillance requirements?

h. Does the system have proper administrative controls for shutdown configurations?

i. Does the system have sufficient redundancy to ensure defense-in-depth functions, assuming a single active failure of equipment or unavailability due to maintenance.

Provide detailed rationale regarding conformance with the above criteria for the staff to use to evaluate the defense-in-depth capabilities of the diesel generator support systems. Revise the SSAR accordingly to reflect the above rationale to categorize these systems as "DID" systems.

Response:

The SSAR will be revised to include more detailed information on the following diesel generator support systems:

a. diesel engine cooling subsystem
b. diesel engine starting subsystem
c. diesel engine lubrication subsystem, and
d. diesel engine combustion air intake and exhaust subsystem

In addition, information will be provided on the diesel engine fuel oil system to clarify the interface between the fuel oil storage and transfer (DOS) system and the diesel engine fuel oil system.

The onsite standby power system, including its support subsystems, perform no safety-related functions and need not meet the listed criteria which are applicable to safety-related systems. However, the following provides service water design information in response to the listed requests.

a. The onsite standby power system (ZOS) consists of two independent diesel generator (D/G) units, each furnished with its own support subsystems. Power supplies to each diesel generator subsystem components are separated to maintain reliability and operability of the onsite standby power system.

b. The design of the ZOS does not ensure functional operability, maintenance access or support plant recovery following design basis events. Maintenance accessibility is provided consistent with the system nonsafety-related functions and plant availability goals.

c. Protection from internal hazards is neither required for the ZOS system, however, the diesel generators and their associated auxiliary systems are located in different fire zones such that a fire within one diesel generator does not affect the other.

d. The ZOS is not protected from natural phenomenon and is not required to remain functional after a natural phenomenon. There is no requirement for ZOS functionality after 72 hours following an event.

e. As a defense-in-depth system, the ZOS is classified as an AP600 Class D system. As discussed in SSAR Subsection 3.2.2.6, this classification invokes industrial quality assurance and industry design standards.

f. The extent of ZOS inclusion in reliability assurance and maintenance programs is discussed in SSAR Subsection 3.2.2.6 for Class D structures, systems and components. The Reliability Assurance Program is further described

in SSAR Section 16.2 and includes a discussion of the applicability to the nonsafety-related defense-in-depth systems, which includes the ZOS.

g. The ZOS, including supporting subsystems, does not have Technical Specification availability control mechanisms (i.e., limiting conditions for operation) nor allowable outage times or surveillance requirements. This system is not safety-related and not required for plant shutdown, and therefore not required to have technical specifications. The ZOS function to provide a backup source of electrical power to onsite equipment needed to support decay heat removal operation during reduced reactor coolant system inventory, midloop operations, is identified as an RTNSS-significant function in Reference 410.171-1. This reference also provides short-term availability recommendations for the equipment used to support this function.

h. Reference 410.171-1 provides recommended availability controls for those portions of the ZOS that perform RTNSS-significant functions during reduced reactor coolant inventory operations.

i. There are two identical standby diesel generator units each complete with its supporting subsystems available for the same standby service, thereby providing system redundancy. The support subsystems for each diesel do not require redundancy to perform their defense-in-depth function.

Reference:

410.171-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: Subsection 8.3.1.1.2.1 of the SSAR will be revised as follows:

Two onsite standby diesel generator units provide power to the selected plant non-safety-related ac loads. These onsite standby diesel generator units and their associated support systems are classified as AP600 Class D, defense-in-depth systems.

Each of the generators is directly coupled to the diesel engine. The engine horsepower rating is slightly greater than the generator rating, such that the unit is load limited by the generator and not by the engine. Each diesel generator unit is an independent self-contained system complete with necessary support subsystems that include:

- Diesel Engine Starting Subsystem
- Combustion Air Intake and Engine Exhaust Subsystem
- Engine Cooling Subsystem
- Engine Lubricating Oil Subsystem
- Engine Speed Control Subsystem
- Static exciter, generator protection, monitoring instruments and controls subsystems.

The diesel-generator starting air subsystem consists of a multistage ac motor driven air cooled compressor, an air cooled aftercooler, refrigerant dryer, and an air receiver with sufficient storage capacity for three diesel engine starts. The interconnecting stainless steel piping from the compressor to the diesel engine dual air starter system includes

all necessary air filters, moisture drainers, and pressure regulators to provide clean dry compressed air for engine starting.

The diesel-generator combustion air intake and engine exhaust subsystem provides combustion air directly to the diesel engine and discharges exhaust gases from the engine to the outside of the diesel generator building. The combustion air circuit includes weather protected dry type inlet air filters piped directly to the inlet connections of the diesel engine mounted turbochargers. The engine exhaust gas circuit consists of the engine exhaust gas discharge pipes from the turbocharger outlets to a single vertically mounted outdoor silencer which discharges to the atmosphere.

The diesel-generator engine cooling system is an independent closed loop cooling system, rejecting engine heat thru two separate roof mounted fan cooled radiators. The system consists of two separate cooling loops each maintained at a temperature required for optimum engine performance. One circuit cools the engine cylinder block, jacket and head area, while the other circuit cools the oil cooler and turbocharger aftercooler.

The diesel-generator engine lubrication system is contained on the engine skid and includes an engine oil sump, a main engine driven oil pump and a continuous engine prelube system consisting of an ac and dc motor driven prelube pump and electric heater. The prelube system maintains the engine lubrication system in service when the diesel engine is in standby mode. The lube oil is circulated through the engine and various filters and coolers to maintain the lube oil properties suitable for engine lubrication.

The diesel generator engine fuel oil system consists of an engine mounted, engine driven fuel oil pump which takes fuel from the fuel oil day tank, and pumps through inline oil filters to the engine fuel injectors and a separate recirculation circuit with a fuel oil cooler. The recirculation circuit discharges back to the fuel oil day tank which is maintained at the proper fuel level by the diesel fuel oil storage and transfer system."

The piping and instrumentation diagrams for the onsite standby diesel generator units and the associated subsystems are shown on Figure 8.3.1-5.

The onsite standby power supply system is shown schematically on one line diagram, Figure 8.3.1-1.

Question 410.188

Section 10.4.9.1.2 of the SSAR states, in part, that the startup feedwater system (SFS) is a non-safety system serving as a first-line of defense for loss of feedwater events, but the passive core cooling system is a safety system which provides safety grade protection for such events. Provide the following information with appropriate justification to demonstrate that the criteria identified in the questions are met by this system, or justify the deviation, if any.

a. Does the system have an electric supply from both normal station ac and on-site non-safety-related ac power supplies that is separated, to the extent practicable?

b. Is the system designed and arranged for conditions or an environment anticipated during and after events to ensure functional operability, maintenance accessibility, and plant recovery?

c. Is the system protected against internal flooding and other in-plant hazards, such as the effects of pipe ruptures, jet impingement, fires, and missiles?

d. Can the system withstand the effects of natural phenomena that have a reasonable likelihood? Important systems and components should be designed to remain functional after a natural phenomena, such as a seismic event, that is of reasonable likelihood or may persist longer than 72 hours.

e. Is there a quality assurance program applied to the system that follows guidelines comparable to those of Generic Letter 85-06 for ATWS, and Appendices A and B of Regulatory Guide 1.155, "Station Blackout," for station blackout non-safety-related equipment?

f. Is the system included in the reliability assurance and maintenance programs for proper maintenance, surveillance, and inservice inspection and testing to ensure the system's reliability is consistent with the determined goals for this system?

g. Does the system have availability control mechanisms, including allowable outage time and surveillance requirements?

h. Does the system have proper administrative controls for shutdown configurations?

i. Does the system have sufficient redundancy to ensure defense-in-depth functions, assuming a single active failure of equipment or unavailability due to maintenance.

Response:

The startup feedwater system performs no safety-related functions and need not meet the criteria listed which are applicable to safety-related systems. However, the following provides startup feedwater design information in response to the listed requests.

a. Each of the two startup feedwater pumps and associated active electrical components are supplied from independent, permanent, nonsafety-related electrical buses. Each bus is capable of being supplied from one of two onsite standby diesel generators. The onsite standby diesel generators and permanent onsite 4160 volt buses are physically separated. No separation is required or provided for the 480 volt load centers or the supply cables for the system electrical loads.

b. The design of the startup feedwater system does not ensure functional operability, maintenance access, or support plant recovery following design basis events. Maintenance accessibility is provided consistent with the pump nonsafety-related functions and plant availability goals.

c. Protection from internal hazards is neither required or provided for the startup feedwater system.

d. The startup feedwater system is not protected from natural phenomenon and is not required to remain functional after a natural phenomenon. There is no requirement for startup feedwater system functionality after 72 hours following an event.

e. As a defense-in-depth system, the startup feedwater system is classified as an AP600 Class D system. As discussed in SSAR Subsection 3.2.2.6, this classification invokes industrial quality assurance and industry design standards.

f. The extent of startup feedwater system inclusion in reliability assurance and maintenance programs is discussed in SSAR Subsection 3.2.2.6 for Class D structures, systems and components. The Reliability Assurance Program is further described in SSAR Section 16.2 and includes a discussion of the applicability to the nonsafety-related defense-in-depth systems, which includes the startup feedwater system.

g. The startup feedwater system does not have technical specification availability control mechanisms (i.e., limiting conditions for operation) nor allowable outage times or surveillance requirements. This system is not safety-related and not required for plant shutdown, and therefore not required to have technical specifications.

h. No administrative controls for plant shutdown configurations are required.

i. Appropriate redundancy is provided such that the startup feedwater system can support normal operation assuming a single active component failure.

SSAR Revision: NONE

Question 410.255

Section 10.4.1.2.1 of the SSAR states "refer to Table 10.3.5-1 for permissible cooling water inleakage and time of operation for maintaining the required condensate/feedwater quality." Describe how the information in this table provides the above information. Where is the permissible cooling water leakage? Where is the information of length of time that the condenser may operate with degraded conditions without affecting the condensate/feedwater quality for safe operation? What are the definition of the action levels (1, 2, and 3) listed in Table 10.3.5-1? Also, provide information in the SSAR regarding the procedure to repair condensate leaks in accordance with Section 10.4.1 of RG 1.70.

Response:

The condensate polishing system design basis for plant operation with either a "chronic" cooling water inleakage of 0.001 gpm or a "faulted" cooling water inleakage of 0.1 gpm is provided in section 10.4.6.2. Condensate polishing minimizes the effect of cooling water inleakage on condensate/feedwater quality until either repairs are completed or an orderly plant shutdown is achieved (Reference section 10.4.6.3, paragraph 3). Table 10.3.5-1 will be revised to include a reference to the cooling water inleakage flow rates of section 10.4.6.2.

The length of time that the plant may operate, with a "chronic" or "faulted" inleakage condition, is determined by the value of the control parameters in Table 10.3.5-1. The table provides the recommended corrective measure (Action Levels) according to the value of the control parameter during the inleakage condition. The specific corrective measures associated with each of the three action levels is defined in section 10.3.5.5 and includes the length of time plant operation may continue during an inleakage condition before an orderly plant shutdown is to be initiated. Table 10.3.5-1 will be revised to include a reference to the definitions of the action levels.

SSAR Revision:

Revise next to the last paragraph of the SSAR Subsection 10.4.1.2.1 and Table 10.3.5-1 as follows.

Leakage at the connections of the tubes to the tube sheets, should it occur, can be detected at either end of each tube bundle by the collection troughs and conductivity cells. These conductivity measurements are indicated and alarmed. This information permits determination of which tube bundle has sustained the leakage. Steps may be taken to repair or plug the leaking tubes. This is performed by isolating the circulating water system from the affected water box while at reduced plant power. This will temporarily reduce condenser capacity by approximately 50 %. The water box is then drained and the affected tubes are either repaired or plugged. Leakage occurring in tube locations other than at the tube ends is detected and alarmed by monitoring the condensate leaving the hotwell. Detection, isolation, and repair are performed as above. Refer to Table 10.3.5-1 for permissible cooling water inleakage and time of operation for maintaining the required condensate/feedwater quality.

Table 10.3.5-1 (Sheet 1 of 3)

## Guidelines for Condensate During Power Operation [d]

| Parameters | Normal Value | Action Levels [c] | | |
|---|---|---|---|---|
| | | 1 | 2 | 3 |
| **Control** | | | | |
| Cation conductivity due to strong acid anions at 25°C, μS/cm | ≤ 0.15 | > 0.15 | > 0.3 | > 0.5 |
| Total cation conductivity at 25°C, μS/cm | ≤ 0.3 | > 0.3 | > 0.5 | > 1.0 |
| Dissolved oxygen, ppb[a] | ≤ 10 | > 10 | > 30 | |
| **Diagnostic** | | | | |
| Total organic carbon, ppb | ≤ 100 | | | |
| Sodium, ppb | < 1 | | | |
| pH at 25°C | > 9.0 | | | |
| Specific conductivity at 25°C, μS/cm | 2 - 6 | | | |
| Morpholine, ppb | (b) | | | |

(a) Air leakage should be reduced until total air ejected flow rate is less than 6 SCFM.
(b) pH, morpholine, and specific conductivity should correlate.
(c) Action Levels are defined in Section 10.3.5.5.
(d) Includes operation during cooling water inleakage; allowable inleakage rates provided in Section 10.4.6.2.

Table 10.3.5-1 (Sheet 2 of 3)

## Guidelines for Feedwater During Power Operation [e]

| Parameters | Normal Value | Action Levels [d] | | |
|---|---|---|---|---|
| | | 1 | 2 | 3 |
| **Control** | | | | |
| pH at 25°C[a] | > 9.5 | < 9.3[b] | | |
| Hydrazine, ppb[c] | ≥ 100 | < 50 | | |
| Total iron, ppb | ≤ 20 | > 20 | | |
| **Diagnostic** | | | | |
| Dissolved oxygen, ppb | ≤ 2 | > 5 | | |
| Cation conductivity due to strong acid anions at 25°C, μS/cm | ≤ 0.2 | | | |
| Specific conductivity at 25°C, μS/cm | 4.0 - 12.0 | | | |
| Morpholine, ppb | (a) | | | |

(a) pH, morpholine and specific conductivity should correlate.
(b) When operating with condensate polishers, the pH of an all-ferrous system can be controlled to a lower value of 9.2, with action required when pH < 9.2.
(c) Values apply if hydrazine is used for oxygen scavenging.
(d) Action levels are defined in Section 10.3.5.5.
(e) Includes operation during cooling water inleakage; allowable inleakage rates provided in Section 10.4.6.2.

Table 10.3.5-1 (Sheet 3 of 3)

## Guidelines for Steam Generator Blowdown During Power Operation [(e)]

| Parameters | Normal Value | Action Levels [(d)] | | |
|---|---|---|---|---|
| | | 1 | 2 | 3 |
| **Control** | | | | |
| pH at 25°C[(a)] | 9.0 - 9.5[(b)] | < 9.0[(b)] | | |
| Total cation conductivity | ≤ 0.8[(c)] | > 0.8[(c)] | > 2 | > 7 |
| Sodium, ppb | ≤ 20 | > 20 | > 100 | > 500 |
| Chloride, ppb | ≤ 20 | > 20 | > 100 | |
| Sulfate, ppb | ≤ 20 | > 20 | | |
| Silica, ppb | ≤ 300 | > 300 | | |
| **Diagnostic** | | | | |
| Cation conductivity due to strong acid anions at 25°C, μS/cm | ≤ 0.5 | | | |
| Suspended solids, ppb | < 1000 | | | |
| Specific conductivity at 25°C, μS/cm | < 3.0 | | | |
| Morpholine, ppb | (a) | | | |

(a) pH, morpholine and specific conductivity should correlate.
(b) When operating with condensate polishers, the pH of an all-ferrous system can be controlled to a value of > 8.8.
(c) Based on concentrations of total anionic species present, any inconsistencies between theoretical and measured values should be investigated.
(d) Action levels are defined in Section 10.3.5.5.
(e) Includes operation during cooling water inleakage; allowable inleakage rates provided in Section 10.4.6.2.

Question 410.257

WCAP-13054 indicates that the AP600 design meets the guidance in Regulatory Guide (RG) 1.26 as related to Section 10.4.2 of the SRP. Demonstrate how the AP600 design meets RG 1.26 regarding the quality group classification for the main condenser evacuation system.

Response:

SSAR Section 3.2 provides the methodology used for classification of AP600 structures, components and systems. Subsection 3.2.2.6 provides a discussion of equipment Class D, and Subsection 3.2.2.7 provides a discussion of equipment Class E. As stated in Subsection 3.2.2.6, some structures, systems and components that have the potential to be contaminated with radioactive fluids but do not normally contain radioactive fluids are classified as Class E. The criteria for classifying such systems as Class E versus Class D are listed in Subsection 3.2.2.7. These design criteria apply to the AP600 main condenser evacuation system (Section 10.4.2). This system is classified as Class E.

SSAR Revision: NONE

Question 410.258

WCAP-13054 indicates that the AP600 design meets the guidance in RG 1.26 as related to Section 10.4.3 of the SRP. However, the designed Quality Group E as stated in Section 10.4.3.2.1 of the SSAR does not meet RG 1.26, which recommends Quality Group D for the system. Address this issue, and correct the SSAR, as appropriate.

Response:

SSAR Section 3.2 provides the methodology used for classification of AP600 structures, components and systems. Subsection 3.2.2.6 provides a discussion of equipment Class D, and Subsection 3.2.2.7 provides a discussion of equipment Class E. As stated in Subsection 3.2.2.6, some structures, systems and components that have the potential to be contaminated with radioactive fluids but do not normally contain radioactive fluids are classified as Class E. The criteria for classifying such systems as Class E versus Class D are listed in Subsection 3.2.2.7. These design criteria apply to the AP600 turbine steam sealing system (Section 10.4.3). This system is classified as Class E.

SSAR Revision: NONE

Question 410.259

Section 10.4.3 of the SSAR does not provide a system flow diagram, a piping and instrument diagram, and a table for the design parameters of the system components. Provide the above information for the turbine steam sealing system.

Response:

The turbine steam seal serves no safety-related or defense-in-depth functions. Further design information is not required for design certification, but is available for review at the Westinghouse offices.

SSAR Revision: NONE

Question 410.260

Section 10.4.5 of the SSAR states that the auxiliary steam system is provided with the necessary controls and indicators for local or remote monitoring of the operation of the system. What are the "necessary controls and indicators"?

Response:

The RAI references SSAR Section 10.4.5. The correct reference is Subsection 10.4.10.5.

The remote and local controls and indicators necessary for monitoring operation of the auxiliary steam system are schematically illustrated on the Piping and Instrumentation Diagram for the auxiliary steam system (SSAR Figure 10.4.10-1).

SSAR Revision: NONE

Question 435.76

The AP600 RTNSS report (WCAP-13856) indicates that the initiating event frequencies used are the same as in the baseline PRA. Section 3.2 of the AP600 baseline PRA states that the loss of offsite power frequency used is for passive plants with full load rejection capability. Although the full load rejection capability has been used in the establishment of the loss of offsite power initiating event frequency, the AP600 RTNSS report does not identify it as a candidate for regulatory oversight. This is not consistent with SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. The staff believes that the availability of the full load rejection capability should be controlled by technical specifications that would then also require some type of surveillances to demonstrate its operability. In lieu of that, the second offsite power source that is now designated as a maintenance source could be redesignated as an alternate offsite power source for all plant modes (including power operation) with appropriate technical specifications to control its availability (in addition to the technical specifications necessary to control the availability of the normal offsite power source). Address this concern.

Response:

The AP600 does not require ac power sources for plant safety. Neither full load rejection capability nor an independent offsite power source are needed for plant safety. No technical specifications are required.

Revision 2 of the PRA will eliminate credit for full load rejection capability in establishing the loss of offsite power initiating event frequency. The effect of this modification on core melt frequency is small. The Loss of Offsite Power Initiating Event (IEV-TE) frequency will increase from 0.082 to 0.12 per year. The importance of IEV-TE is 0.87 percent in the baseline PRA. Therefore, the core melt frequency will increase less than half a percent as a result of this change.

SSAR Revision: NONE

PRA Revision: Revision 2 of the PRA will eliminate credit for full load rejection capability. Revision 2 will be completed by December 31, 1994.

Question 440.57

Provide a description of the Westinghouse emergency procedures guidelines (EPG) for the AP600 for the development of emergency operating procedures (EOP) for conditions including shutdown and mid-loop operations. Specifically, address the adequacy of the EPGs (existing or to be proposed) for shutdown conditions when many systems will be out for maintenance and the plant is in a configuration different from the normal plant operation.

Response:

Subsection 18.9.8 provides the high-level operator action strategies for emergency operations and describes the process that will be used to develop AP600-specific Emergency Response Guidelines. As stated in Subsection 18.9.8.1.1, the development of the AP600-specific Emergency Response Guidelines is based on the Westinghouse low-pressure Emergency Response Guidelines. Reference 440.57-1 provides a comparison of the low-pressure ERG reference plant system designs to the AP600 system designs.

A set of matrices and figures depicting event mitigation strategies and levels of defense-in-depth for a number of events starting from full power and from various shutdown conditions are provided in Reference 440.57-2.

RAI 440.063 discusses the AP600 response to events that occur at shutdown. The AP600 has passive safety-related features that protect the plant during all modes of operation including shutdown and refueling. Shutdown events and midloop operations were evaluated in the AP600 PRA and the results indicate a low contribution to core damage from shutdown events. Shutdown events were included in the RTNSS implementation as described in Reference 440.57-3.

The AP600 Technical Specifications in SSAR Section 16.1 and as discussed in RAI 440.58, require availability of passive safety-related features during shutdown conditions. For events that occur at hot shutdown or hot standby, the full compliment of passive safety-related systems is available to mitigate an event. For events that occur at cold shutdown conditions with the RCS pressure boundary intact, the passive safety-related features are available except for the accumulators and the containment. When the RCS pressure boundary is open, the passive RHR heat exchangers, accumulators, and core makeup tanks are not effective. Prior to and during reduced inventory operations, ADS stage 1/2/3 valves must to be opened. The IRWST gravity injection lines, containment recirculation lines and containment closure capability must be available. For events that occur during refueling conditions, the refueling cavity inventory provides for at least 72 hours of heat removal, and the design includes a safety-related connection for makeup to the refueling cavity if needed in the long term.

The ERG's will provide guidance for responding to events at shutdown conditions by making use of normally operating nonsafety-related systems when available and by use of the safety-related systems when the normal systems are unavailable. References 2 and 3 and RAI 440.063 provide description of how this use of plant systems addresses the plant configurations at shutdown conditions.

References:

440.57-1 WCAP-14075, "AP600 Design Differences Document for Development of Emergency Operating Guidelines Report," May 1994.

440.57-2 WCAP-13793, "AP600 System / Event Matrix," June 1994.

440.57-3 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

SSAR Revision: NONE

Question 440.66

Transient and accident analyses presented in safety analysis reports typically concentrate on power operation. The recent experience from the events in operating reactors indicate that further evaluation for the plant lower modes is needed. Confirm whether each of the transient and accident analyses included in the SSAR for the AP600 is applicable to modes 1 through 6, or provide a discussion of any plans, if any, with respect to the transient and accident analysis at lower operation modes.

Response:

The analyses provided in SSAR Chapter 15 are specifically applicable to power operation. The analyses provided in Chapter 15 are expected to bound those that could occur at other operation modes.

No additional analyses are planned beyond that provided in Chapter 15 and the PRA.

See the response to RAI 440.63 for information pertaining to the passive system capabilities during shutdown modes of plant operation and the PRA analyses performed for these modes.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.90

The PXS is designed to supply the core cooling flow rates to the RCS during accidents. Section 6.3.2.1 of the SSAR states that the Chapter 15 accident analyses flow rates and heat removal rates are calculated by assuming a range of component parameters, including best estimate and conservatively high and low values. Clarify when the best estimate values are used in the licensing design basis analyses, and provide the bases for their use.

Response:

Chapter 15 accident analyses are performed assuming the limiting single active failure(s) occurs in the PXS, where appropriate. The design of the AP600 is such that these potential failures cannot defeat the function of the PXS system by preventing flow from occurring in any path; this statement does not apply to the ADS system, where some flow paths can be lost. Furthermore, one PRHR heat exchanger is presumed to be out of service (as permitted by the Technical Specifications) in Chapter 15 accident analyses where it is conservative.

Postulated possible single active (valve) failures serve to increase the resistance in the core makeup tank and PRHR flow paths. Each of these flow paths is modeled in the small and large break LOCA analyses of Chapter 15 under the conservative assumption that one of the parallel flow path valves has failed to open, which increases the resistance of the affected PXS flow paths thereby decreasing the flow delivery. Within the context of this assumption, best-estimate piping resistance values are then applied.

This approach obtains the most representative prediction of the AP600 plant behavior during a postulated large break LOCA event without skewing the prediction arbitrarily. It is consistent with the philosophy and use of WCOBRA/TRAC to perform a best-estimate analysis of the AP600 large break LOCA. The requirement to consider a single active failure is met in the 10CFR50 Appendix K small break LOCA analysis.

The small break analysis, using the NOTRUMP code, uses the same assumptions on the flow path resistances as the large break analysis. The nominal, or best estimate piping resistance is also used. However, in the small break analysis, the Appendix K requirement of using the ANS 1971 plus 20% decay power curve is also used for conservatism. The worst single failure is also assumed in these calculations.

None of the Non-LOCA or steam generator tube rupture analyses of Chapter 15 use best estimate values for PXS component parameters affecting flow rates and heat removal rates. These analyses use conservatively high or low values.

SSAR Revision: None
PRA Revision: None

Question 440.121

Generic Issue 122.2 deals with the adequacy of emergency procedures, operator training, and available monitoring systems for determining the need to initiate feed-and-bleed cooling following loss of the steam generator heat sink. The AP600 design relies on feed (from the IRWST, CMT, and accumulator) and bleed (through the ADS) operation as backup to the startup feedwater and PRHR heat exchanger. Provide a discussion of the emergency procedure guidelines, operator training, parameters, and instrumentation and control systems relevant to the initiation of feed and bleed.

Response:

In case of a beyond design basis event where the nonsafety-related startup feedwater (SFW) pumps and the safety-related passive RHR (PRHR) heat exchanger both fail following a transient, core cooling would be automatically actuated as follows:

- With the loss the SFW and the PRHR the initial steam generator (SG) secondary side inventory would effectively remove decay heat for about 1.5 hours.
- After the SG inventory was depleted, the reactor coolant system (RCS) would heat up. The combination of low SG water level and high RCS temperature automatically actuates the core makeup tanks (CMT)s.
- If the SFW and the PRHR are not recovered, then decay heat would be removed by steaming of reactor coolant through the pressurizer safety valves.
- The loss of reactor coolant through the pressurizer safety valves would eventually result in the CMT level dropping to the automatic depressurization system (ADS) actuation setpoint.

The operator can also manually actuate the CMTs and ADS based on the same signals that would automatically actuate them (CMTs on low SG water level plus high RCS temperature, ADS on low CMT water level). Refer to SSAR Chapter 18 and the AP600 ITAAC for additional man-machine information.

SSAR Revision: NONE

Question 440.145

Section 28-2 of the Code Qualification Document (CQD) for WCOBRA/TRAC, "Compliance with Regulatory Guide 1.157, REGULATORY POSITION 1," states that application of WCOBRA/TRAC to the AP600 design is considered acceptable, based on information in the CQD and confirmatory tests and comparisons currently being performed on the unique features of the AP600 design, the results of which will be provided in other reports. Describe the specific features of the AP600 design that will be evaluated with these tests, and show how the results will be used to meet the requirements of the Regulatory Position. Are the results to be incorporated into a later edition of the CQD?

Response:

A Code Applicability document will be submitted by September 30, 1994 which will describe the application of the WCOBRA/TRAC code to the AP600. This document will contain data comparisons to experiments which model AP600 features. These results will supplement the data comparisons for the code provided in the WCOBRA/TRAC code qualification document (CQD).

SSAR Revision: None
PRA Revision: None

Question 440.146

Justify the capability of WCOBRA/TRAC to adequately predict downcomer ECC bypass and CCFL phenomena for the UPTF tests. Section 14-4 of the CQD presents the comparison of code calculations to UPTF Tests 6 and 25, to evaluate the ability of the code to predict ECC bypass in the downcomer. Test 6 comprised five steady state runs with steam flows to establish points on a flooding curve for the downcomer.

a. Clarify the presentation and comparison of the code calculations to the test data. The interpretation of the comparisons would be facilitated if the data and code traces were plotted together. When different parameters are plotted together, i.e., test temperature and calculated enthalpy, a second (right side) y-axis would be appropriate. Also, the axis limits should represent the range of the data, where possible (e.g., Figures 14-4-26 and -27, -28 and -29, -30 and -31, etc.). For example, on Figure 14-4-37, if the data and the calculated results cannot be on the same plot, at least the vertical scale for the calculated result should be the same as for the data.

b. Test 6-131: As mentioned in Section 14-4-5 of the CQD, the code calculates the upper plenum/downcomer pressure rise to a higher pressure than shown by the data. Explain the reasons for the differences in responses. Also, it is stated that there is good agreement between calculated and test results after 70 seconds; however, the test pressure is about 1100 kPa at that time period and the code is still up at ~1350 kPa. Good agreement is delayed until ~100 seconds. Explain why the code overpredicts system pressure for the first 100 seconds. The timing of the calculated pressure decay following the end of steam injection is about 20 seconds slow. This timing delay is also present in the axial and azimuthal downcomer dp (Figures 14-4-30 and -31, -32 and -33, and -34 and -35). Is this delay related to prolonged lateral bypass flow in the calculated results? Was the steam flow boundary condition applied with the correct timing or was there a time shift? What is the mechanism responsible for the delay in the calculated lateral bypass flow? Does this indicate a deficiency in either the liquid entrainment or CCFL models? Explain the comparison in more detail. Note also that the code does not calculate the correct lower plenum and downcomer water levels, particularly at the end of this test (see Figures 14-4-36, -37, -38, and -39). Similar differences between the calculated and the measured water levels are also noted in Runs 132 and 135. Explain the reasons for the differences.

c. Tests 6-132 and 6-133: These tests are repeats of Run 6-131 with variations in injection flow rates and pressure; the comments identified for Run 131 apply to these comparisons as well. The CQD provides a nearly identical narrative for all of the runs of Test 6. Emphasize what the response differences were and how the code predictions serve to verify the adequacy of the CCFL model. The stated purpose of Test 6 was to establish points on a flooding curve for the downcomer. Provide analysis results detailing how well WCOBRA/TRAC predicted the points on the flooding curve.

d. Test 6-135: This run was with slightly higher containment backpressure - 360 vs. ~290 kPa. Explain why the code predicts more bypass high in the downcomer than for the low backpressure tests. Is this trend consistent with the test results? Justify the applicability of the CCFL models

in the code and what it implies in terms of the ability of the code to successfully predict ECC bypass.

Response:

Additional comparisons for ECC by-pass will be provided in the WCOBRA/TRAC Code Applicability Document for the AP600 as indicated in the response to RAI 440.145. These comparisons will include tests in the Upper Plenum Test Facility which utilized direct vessel injection similar to that used for the AP600 design. A comparison to that data will be provided in the Code Applicability Document.

Additional clarification to this specific question will be included in the response to the generic WCOBRA/TRAC questions on volume 2 of the WCOBRA/TRAC Code Qualification Document. A revision to this RAI, which responds to the specific questions of this RAI, will be provided following submittal of the responses to the generic WCOBRA/TRAC questions in September 1994.

SSAR Revision: None
PRA Revision: None

Question 440.147

Demonstrate the applicability of WCOBRA/TRAC to the calculation of the ECC bypass phase for the AP600 design. The AP600 downcomer is not typical of a current generation PWR. It has a significantly larger annular gap width, and there is no thermal shield in the annular region. Discuss the applicability of the test results to this design. The design includes a reflector: a large metal mass occupying most of the barrel-baffle region (between the fuel bundle and the inside of the core barrel). The AP600 design also employs direct injection of the accumulator liquid into the downcomer. The lower plenum is significantly more open than for a current generation PWR; there are no instrument penetrations, there is a single lower core support which also accomplishes any bundle inlet flow distribution.

a. Explain how these differences will affect the penetration of ECC liquid. What test comparisons demonstrate the capability of WCOBRA/TRAC to correctly calculate ECC penetration and bypass in the AP600 downcomer?

b. Address the influence of thermal contact between the reflector and the core barrel on hot wall ECC delay.

c. Address the influence of the reduced resistance of the lower plenum structures during blowdown. How will it affect the expected duration of reverse steam flow and how will this impact the duration of the ECC bypass period?

Response:

The applicability of WCOBRA/TRAC to predict the ECC by-pass effects for the AP600 vessel design will be discussed in the WCOBRA/TRAC code applicability document, as indicated in the response to RAI440.145. There will be additional comparisons to Upper Plenum Test Facility data which includes direct vessel injection. The specific design features of the AP600 vessel are discussed in the following:

a. The differences in the design of the AP600 vessel and downcomer are accounted for in the input to the plant model. The larger metal mass of the reflector is specifically modeled as a separate channel as shown in the SSAR WCOBRA/TRAC vessel nodding. The heat transfer from the metal mass to the core barrel is accounted for in the heat structure calculation. Also, the heat that is removed through the reflector flow holes is modeled and is accounted for in the calculation. The heat transfer to the liquid will act to decrease its subcooling and thereby reduce its condensation potential. This will penalize the ECC penetration into the lower plenum. The larger annular gap is directly modeled and will act to reduce the upward steam velocity which will aid the ECC penetration into the lower plenum. The single lower support plate is similar to several W designed plants, and is not a difference for the AP600.

b. The thermal contact of the reflector will act to provide a heat source to the core barrel. However, the heat transfer from the reflector and core barrel quickly becomes conduction limited because of the low conductivity of the stainless steel structures (conductivity is approximately 10 Btu/hr-ft-°F). While the heat

source is larger, it has a negligible effect on the ECC delivery because of the conduction limited heat transfer in the core barrel and reflector walls.

c. The more open lower plenum has two effects on the flow in that region. There is almost no cross flow resistance or drag on the steam so the pressure drop is reduced and the steam velocity can be larger. Also the lateral flow area is larger, thereby reducing the cross flow velocities which will reduce the potential for liquid sweep-out of the lower plenum. These effects are modeled in the WCOBRA/TRAC input for the AP600.

SSAR Revision: None
PRA Revision: None

Question 440.149

Clarify the presentation and comparison of the code calculations to the test data for the LOFT tests that simulate reflood-related phenomena in the analysis of the LBLOCA for the AP600 design. See Q440.146(a).

Response:

The response to this question and additional clarification of the LOFT comparisons will be included in the WCOBRA/TRAC code applicability document for the AP600, which will be submitted by September 30, 1994.

SSAR Revision: None
PRA Revision: None

Question 440.150

LOFT L2-5 data (Figures 14-1-26, -36, and -37) show a significantly higher intact hot leg liquid flow during core reflood (~40 to 60 seconds) than that calculated by WCOBRA/TRAC. Is this response related to entrainment during quench front advancement or to ECC nozzle condensation in the intact cold leg? Explain why WCOBRA/TRAC does not capture this response feature.

Response:

The response to this question will be included in the responses to the generic WCOBRA/TRAC review questions. A revision to this RAI, which responds to the specific questions of this RAI, will be provided following submittal of the responses to the generic WCOBRA/TRAC questions in September 1994.

SSAR Revision: None
PRA Revision: None

Question 440.151

The conclusion drawn in Section 14-2-6-5 of the CQD attributes poor predictions of core heat transfer in CCTF Run 75 to underprediction of system pressure. Provide further justification for this conclusion.

a. The code appears to be overpredicting core steam temperature, and therefore, underestimating post-dryout heat transfer from the heater rods. Specifically, the data showed more droplet entrainment (lower quality mixture) and, therefore, less superheat than that calculated by the code. The trend of the data for CCTF Run 62 (Figures 14-2-31 through 14-2-41) low in the core (below 10.0 ft) showed that the channel steam temperature drops to saturation coincident with increased fuel cladding surface cooling, thereby resulting in the turnover of cladding heatup. Provide additional justification for the interfacial heat transfer and droplet entrainment models. This may be because of underprediction of liquid droplet entrainment in the steam during core reflood, a conclusion that would be consistent with the underprediction of the LOFT L2-5 intact hot leg mass flow noted above. The explanation offered in the Section (14-2) of the CQD implies that the data are not realistic because of quenching of the steam temperature sensor probes. Justify this explanation.

b. In Volume I, page 5-8, it is stated that bubbles of superheated steam are unlikely to occur extensively in a LOCA transient because the large interfacial area will quickly drive the system to saturation. This statement seems justified according to the CCTF data but appears to contradict the calculated results. Explain the high superheat response predicted by the code.

c. In the description of the interfacial heat and mass transfer models (in Chapter 5), there is a preponderance of the number "278" for superheated vapor, subcooled vapor, and superheated liquid:

- $h_{i,SHV}$ = 2.78 (a constant is assumed)
- $h_{i,SCV}$ = 2780.0 (a large constant value is assumed, presumably to drive the mixture toward equilibrium)
- $h_{i,SHL}$ = 278.0 (a large value is assumed in order to drive the liquid towards saturation).

It appears that these heat transfer coefficients are arbitrary and have little or no physical basis. Substantiate the interfacial heat transfer models used by the code.

d. It appears that WCOBRA/TRAC does not include the major influences of structures on the flow regime transition. At the highest levels, calculated quench times for rods 2 (low power) at 10.0 ft and 3 (average power) at 11.68 ft were sooner than the data. Is this an early calculated top-down quench that is not supported by the data? Or is this because of the calculated $T_{min}$ of 700 to 900°F discussed in Section 14-2 of the CQD? It is stated on page 14-2-17 that the mass in the upper core was underpredicted by the code. It is further noted on page 14-2-19 that the prediction (measurement?) of substantial mass retention in the upper core region in the CCTF tests was attributed to a flow regime transition, possible resulting from the rewetting of structural members, the effects

of which are not simulated. Should the structure rewetting effect be included in the flow regime transition model? It appears to significantly affect the outcome of the calculation, primarily the predicted core inventory distribution and heater rod cooling. Explain whether and why this model produces an adequate representation of the phenomena present in the core during reflood.

e. Section 14-2-6-5 of the CQD (CCTF Run 75) states that the code predicts significantly lower pressure during this lower plenum injection test, and that this is responsible for the underprediction of reflood cooling. It is implied that this result is an anomaly because the scaled (FLECHT SET) results did not exhibit a similar pressure drop. However, the reference cited, Akimoto et. al.,[1] apparently does not support this conclusion. Instead, the response difference between full size and scaled results is attributed to flow area scaling. Provide further explanation for the overprediction of vessel-to-broken cold leg delta-p and broken cold leg steam flowrate.

f. In Section 14-2-7 of the CQD, "Overall Comparisons and Conclusions," Test 75 is singled out as poorly predicting the fuel temperature results, but some cladding surface temperature responses for the other CCTF tests (62, 63, and 67) appear to show the same trend. It is further noted in Reference 1 of the CQD that the maximum broken cold leg dp was ~50 kPa in the base case (62) and ~60 kPa in the FLECHT coupling test (75). Because the difference is small (only about ~1.5 psi), similar responses for the two tests should be expected, as shown. Therefore, provide further clarification on the conclusion that the Test 75 prediction is significantly poorer than the others.

Response:

The response to this question will be included in the responses to the generic WCOBRA/TRAC review questions. A revision to this RAI, which responds to the specific questions of this RAI, will be provided following submittal of the responses to the generic WCOBRA/TRAC questions in September 1994.

SSAR Revision: None
PRA Revision: None

[1] H. Akimoto, T. Iguchi, Y. Murao, *Pressure Drop through Broken Cold Leg during Reflood Phase of Loss-of-Coolant Accident of Pressurized Water Reactor*, Journal of Nuclear Science and Technology, 21(6), pp. 450-465 (June 1984).

Question 440.152

Provide additional justification to explain the comparison to SCF Test 604 results (Section 14-3-6-1 of the CQD).

a. This case resulted in underpredicted cladding temperatures in the region above the core midplane (Figure 14-3-11), and an earlier downward temperature slope of cladding temperature, that indicate overprediction of quench front advancement. Calculated steam temperatures are again overestimating the time superheat is present. As with the CCTF comparisons, provide additional explanation to justify whether and why the code is predicting the correct thermal-hydraulic responses.

b. As described on page 14-3-16, pressure oscillations in the calculated result indicated the prediction of gravity reflood oscillations. This phenomenon does not appear in the data. Are these oscillations responsible for the overprediction of the heat transfer at the quench front? The overprediction of quench front advancement may be related to the observation that fuel rewet onset temperature value (discussed in Section 6-2-6) is too high. Provide a more detailed explanation of this phenomenon, showing cause-effect relationships.

c. Provide additional justification for the statement that the instrumentation may not be adequate to capture the phenomena. Include evaluations of instrument sensitivity and time response characteristics. If the oscillations are present but damp out, state why. If the gravity reflood oscillations are not present in the test, clarify why WCOBRA/TRAC predicts them.

d. Inventory is underpredicted in the upper half of the core. The code apparently predicts an early quench low in the core but a late quench high in the core. Clarify the reasons for the differences between the calculated results and the data.

Response:

The response to this question will be included in the responses to the generic WCOBRA/TRAC review questions. A revision to this RAI, which responds to the specific questions of this RAI, will be provided following submittal of the responses to the generic WCOBRA/TRAC questions in September 1994.

SSAR Revision: None
PRA Revision: None

Question 440.153

Accumulator injection will be followed by injection of the $N_2$ pressurizing gas. The staff has raised an issue regarding the successful gravity draining of the CMT/IRWST in the presence of the pressurization effect of the cover gas. Identify applicable assessment data and demonstrate the acceptability of the LOCA Methodology to be used for evaluation of long term cooling response for the AP600 design.

Response:

The long term cooling model which will be used to demonstrate the performance of the passive safety systems uses the WCOBRA/TRAC code. This model will be verified against the Oregon State University tests which simulate the injection of the accumulator nitrogen when the accumulators completely drain. WCOBRA/TRAC will be compared to these tests to verify its applicability for CMT/IRWST gravity drain during AP600 long term cooling transients. These comparisons will be documented in the WCOBRA/TRAC V&V report to be provided to the NRC in May 1995. Once the WCOBRA/TRAC analysis is complete, a less sophisticated analysis method may be possible.

SSAR Revision: None
PRA Revision: None

Question 440.156

Long term cooling is an issue for gravity-drained systems with low driving heads with competing forces present. This is not addressed in the CQD. Describe the methodology to be used for verifying the adequacy of the ability of the AP600 to employ long term cooling, including the assessment data and a demonstration of its applicability.

Response:

The WCOBRA/TRAC code will be used for the AP600 long term cooling calculations. The application of the use of the WCOBRA/TRAC code for this application will be provided in the WCOBRA/TRAC code validation document which will be submitted in May 1995. The Oregon State University long term cooling experiments will provide the data to verify the use of the code for this purpose.

SSAR Revision: None
PRA Revision: None

Question 440.176

Section 8.1.1 of WCAP-13856 identifies general methods used for the evaluation of the functional interaction aspect of systems interactions between the active and passive systems in the AP600 design. These methods include (a) assuming the operation of a non-safety-related system if its actuation worsens the analytical result of the Chapter 15 analysis, (b) considering those integral tests involving both safety-related and non-safety-related systems in the AP600 test program that will validate the thermal-hydraulic codes used in Chapter 15, and (c) modelling the system interdependencies in the system fault trees and event trees in the PRA. This section also includes a list of safety-related functions of the active non-safety-related systems in the AP600 design to preclude the potential for non-safety-related systems to adversely interact with the safety system. However, it does not provide detailed information to demonstrate how these methods can capture all of the subtle interactions.

Provide a discussion of the detailed methods used to identify possible adverse systems interactions (ASI) that could be hidden in the AP600 design, and proposed resolutions of these ASIs, including design modifications and operating procedures. In addition, provide the following information:

a. For some combinations of depressurization hardware, the success criteria analysis for medium- and small-break LOCA reported in the PRA (Appendix J) indicates that significantly higher peak clad temperatures (PCTs) are predicted for the operation of two accumulators as compared to one accumulator operation. It states that with two accumulators, the pressurization following accumulator injection causes an increase in the pressurizer water level. This results in a lower release of steam through the automatic depressurization system (ADS) valves, which leads to a delayed in-containment refueling water storage tank (IRWST) injection and a higher PCT because of the lower two phase reactor vessel water level. Confirm that no clad failure would take place. What are the boundary conditions, initial conditions, and timing of events that can be varied in the analysis of these events, such that a worst and bounding case can be determined? Can partial or brief operation of the normal residual heat removal system (RNS) in the injection mode have a similar effect?

b. For certain medium-break LOCA cases and the transient event trees, the status of the CVS is not queried. This implies that the ADS success criteria are identical whether or not the CVS worked. Can continuous or intermittent operation of the CVS result in the RCS pressure "hanging up" above the gravity injection pressure while not providing sufficient CVS flow to maintain core cooling?

c. Can CVS operation cause a situation where insufficient flow is provided to the RCS by CVS during a steam generator tube rupture (SGTR) event, but gravity injection cannot function because the ADS success state assumed cannot compensate for pressure effects of the CVS?

d. Given a small-break LOCA, the expected response of the plant is reactor coolant pump (RCP) trip, core makeup tank (CMT) actuation, passive residual heat removal system (PRHRS) actuation, ADS actuation, and manual actuation of RNS injection. In such a scenario, the low pressurizer level signal or a safety injection signal will also start the CVS pumps and keep them running and injecting to the PRHRS return line to the

SG channel head. How does the CVS injection affect the operation of the PRHRS and the plant's ability to mitigate the small-break LOCA and other transients that result in the CVS injection?

e. In an SGTR, what is the effect of SG pressure on the RCS pressure? Can SG pressure cause the RCS pressure to remain high, impeding gravity injection flow? How does the answer to this question depend on the number of tubes ruptured? How does the status of SG isolation affect RCS pressure for one- or multiple-tube rupture scenarios?

Response:

The response to RAI 440.95 provides a discussion of the methods used to identify possible adverse systems interactions.

a. Clad failure is not a PRA acceptance criteria. Peak clad temperatures below 2200°F is the acceptance criteria. The response to RAI 440.178 provides details of the automatic depressurization system success criteria analysis performed for the PRA. Possible adverse interactions with the normal residual heat removal system were investigated in tests at SPES-2 and at OSU. These tests provide data to confirm analytical methods that support successful core cooling with normal residual heat removal operation.

b. Chemical and volume control system injection flow is much smaller than the CMT or the accumulator, so any impact is expected to be small. Note that the chemical and volume control system can only affect reactor coolant system pressure indirectly since the automatic depressurization system provides large vent holes. The PRA success criteria analysis performed with different numbers of CMTs and accumulators bounds the chemical and volume control system effect since the chemical and volume control system flow is much smaller.

c. A steam generator tube rupture is mitigated without automatic depressurization system operation with chemical and volume control system injection and operator actions or with passive residual heat removal heat exchanger operation (with or without operator action). The core makeup tanks provide injection that is independent of chemical and volume control system operation. Normal CVS makeup operation will not adversely affect the SGTR mitigation. A malfunction of the CVS that results in excessive makeup is isolated by redundant safety-related valves.

   There is an alternate success path in the PRA that protects against failure of the operators and of passive residual heat removal or the CMTs. This path includes actuation of the automatic depressurization system and injection from the accumulators and the IRWST. See responses (b) and (e) for additional information.

d. Possible adverse interactions with the chemical and volume control were investigated in tests at SPES-2 and at OSU. These tests provide data to confirm analytical methods that support successful core cooling with chemical and volume control system operation. See item (b) for additional discussion.

e. For the AP600, the design basis steam generator tube rupture consists of a single tube. For this event, automatic depressurization system actuation does not occur as demonstrated in the safety analysis contained

in SSAR Subsection 15.6.3 and Section 4.2 of the AP600 Design Change Report dated February 15, 1994. In a design basis steam generator tube rupture, the reactor coolant system pressure decreases to the secondary side pressure, which terminates the reactor coolant system leak. The core makeup tanks effectively add makeup to the reactor coolant system without draining the core makeup tank level to the automatic depressurization system actuation setpoint.

Best estimate analysis indicates that automatic depressurization does not occur during a multiple steam generator tube rupture. Refer to Revision 1 of the response to RAI 440.27.

There will be a test performed at SPES-2 with automatic depressurization manually actuated during a single steam generator tube rupture. This test will provide data to confirm analytical methods that support successful automatic depressurization system operation in such a SGTR scenarios.

SSAR Revision: NONE

Question 440.202

In the PRHRS break tree, Figure F-19 of the PRA, does the RTNSS evaluation take credit for isolating the break? Does it reflect the possibility that one HX was isolated previously and the other one has the break in it now? Are there any missions in the PRA for which both heat exchangers are needed?

Response:

As documented in Reference 440.202-1, the RTNSS evaluation of the passive residual heat removal heat exchanger break event models the probability for isolating the failed passive residual heat removal heat exchanger. The passive residual heat removal fault tree model used in the RTNSS evaluation of the passive residual heat removal heat exchanger break event considers the probability that one of the passive residual heat removal trains is already isolated.

There are no missions in the PRA for which both passive residual heat removal heat exchangers are needed.

Reference:

410.202-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.203

How much makeup water (CVS, CMT) is needed during extended operation of the PRHRS for decay heat removal?

Response:

Extended operation of the passive residual heat removal heat exchanger provides decay heat removal for design basis non-LOCA events. The core makeup tanks provide sufficient reactor coolant system makeup to compensate for reactor coolant system cooldown shrink effects and for pre-existing leakage (within technical specification limits). The chemical and volume control system is not needed to support passive residual heat removal heat exchanger operation. Refer to the response to RAI 440.218 for additional discussion of passive residual heat removal heat exchanger operation and the reactor coolant system inventory.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.208

An issue for conventional PWRs during a large-break LOCA is potential precipitation of boron in the core region. The issue is addressed by switching to hot leg injection. How does the AP600 design address this issue? Describe the behavior of boron in the core during long-term recirculation with the RCS steaming to the containment and water returning through the sump recirculation screens. Does boron accumulate in the core?

Response:

The characteristics of the AP600 passive core cooling system (PXS) limits the build up of boron in the core. The long term injection to the core is from the in-containment refueling water storage tank and /or the containment recirculation sump. The automatic depressurization system is in operation to vent the steam generated in the core and maintain the reactor coolant system pressure low enough to permit this recirculation from the containment.

A mixture of steam and water is vented through the automatic depressurization system 4th stage paths on the hot legs. As decay heat drops, the water level in the reactor will rise. The amount of water being vented out the 4th stage then increases. The water that is vented with the steam carries boron from the reactor, which limits the build up of boron. Refer to the response to RAI 440.154 for additional discussion.

SSAR Revision: NONE

PRA Revision: NONE

Question 480.4 (Misidentified by NRC as 280.4)

Provide the test data for the passive containment cooling system that was generated from both the small and large scale test facilities that are related to dry shell testing conditions.

Response (Revision 1):

Four test data reports for the passive containment cooling system generated from small- and large-scale test facilities under dry shell test conditions are being reissued as WCAPs. The WCAPs ~~will be~~ were transmitted to the NRC under separate cover ~~by May 31, 1993 with Westinghouse proprietary 2 and 3 revisions~~ via Westinghouse letter ET-NRC-93-3903. The WCAP numbers and report titles are as follows:

- WCAP-13727 (Proprietary) and WCAP-13728 (Non-proprietary), "Heavy Water Reactor Facility Project (HWRF) Small Scale Containment Cooling System Test Final Report"
- WCAP-13732 (Proprietary) and WCAP-13733 (Non-proprietary), "Heavy Water Reactor Facility (HWRF) Small Scale Containment Cooling Test Preliminary Series 2 Test Results"
- WCAP-13742 (Proprietary) and WCAP-13743 (Non-proprietary), "Heavy Water Reactor Facility Project, Phase I AP600 Small Scale Passive Containment Cooling System Test 'Dry' Test Results Applicable to the HWRF Project"
- WCAP-13725 (Proprietary) and WCAP-13726 (Non-proprietary), "Heavy Water Facility (HWRF) Large Scale Passive Containment Cooling System Baseline Test Data Report"

SSAR Revision: NONE

Question 952.27

The staff is concerned that the full range of possible conditions under which ADS may operate has not yet been considered. Justify your selection of operating conditions. A "map" of RCS behavior of the AP600 vs. ADS test facility operation would be useful in making this assessment.

Response (Revision 1):

~~The requested map of RCS behavior of the AP600 vs. ADS test facility operation will be provided in a revision to this RAI by March 31, 1994.~~

The requested map was provided to the NRC via Westinghouse letter NTD-NRC-94-4240, dated 7/26/94.

SSAR Revision: NONE

Question 952.75

What are the steam generator recirculation ratios and riser inlet temperatures for the SPES-2 facility? Also, were the volumes for the steam generator bypasses accounted for in the second provided RELAP5 input model?***

Response:

The average steam generator recirculation ratios for the SPES-2 tests are 1.9 for steam generator A and 1.8 for steam genertor B.

Typical steam generator riser tempertures are:

| | |
|---|---|
| SGA | (T-A05S), hot side - 266.5 °C (511.7 °F) |
| | (T-A09S), cold side - 261.1 °C (502.0 °F) |
| SGA | (T-B05S), hot side - 263.7 °C (506.7 °F) |
| | (T-B09S), cold side - 259.5 °C (499.1 °F) |

Also, the observed tubular downcomer temperatures (riser inlet temperatures) are:

| | |
|---|---|
| SGA | (T-A02S) - 259.0 °C (4 8.2 °F) |
| | (T-A03S) - 258.8 °C (497.8 °F) |
| SGB | (T-B02S) - 259.0 °C (498.2 °F) |
| | (T-B03S) - 259.6 °C (499.3 °F) |

The referenced RELAP5 input model was not developed or provided to the NRC by Westinghouse. Therefore, we can not speak to the construction or content of the model.

SSAR Revision: None
PRA Revision: None

***Letter from A. Alemberti, ANSALDO S.p.A. to Marcos G. Ortiz, EG&G Idaho, Inc., INEL, "Revised RELAP5/Mod3.0 input deck 'spes2.1' of SPES-2 facility," June 14, 1993.