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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 19, 1994, April 27, 1994, April 29, 1994, May 11, 1994, May 26, 1994, June 1, 1994 and June 8, 1994. In addition, revisions of responses previously submitted is provided. This completes the response to the letter dated April 7, 1994.

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-94-4236
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JULY 25, 1994

RAI No.	Issue
210.048	SSAR section 3.7.3.5
210.063	Method of combination of dynamic responses
210.064	Use of elastic-plastic method of analysis
210.068	Stress criteria for active component supports
210.069	Section 3.9.3.4.3, snubber operability
210.088	Seismic qualification report
440.001R01	SPES test, check valve testing & PRHR testing
440.011R01	ADS testing
440.052	CMT Scaling Report
440.133	RNS pump suction
440.171	HF analysis inclusion of high point vent operation
440.245	Battery bank inconsistency between SSAR & PRA
480.049	Provisions for Type C testing, Table 6.2.3-1
480.050	Type C testing of service air
480.052	SSAR Table 6.2.3-1 & Figure 9.2.4-1
480.056	Relief valves as containment isolation barriers
480.060	manual vs remote manual
480.066	Margin between max calculated & design cont press
480.068	Postulated break size for subcompartment analyses
480.069	Use of TMD code for M&E releases
492.005	Fixed incore detector monitor
952.082	SPES-2 pipe schedule changes

Question 210.48

The following requests are relative to Section 3.7.3.5 of the SSAR, "Equivalent Static Load Method of Analysis:"

a. The second paragraph in this section states that single degree of freedom subsystems are designed for accelerations associated with their natural frequency. The staff's position as stated in Section 3.9.2.II.2.a(2) of the SRP is that for equipment that can be modeled adequately as a one-degree-of-freedom system, only the use of a static load equivalent to the peak of the floor response spectra is acceptable. Either revise this paragraph to be consistent with the staff position, or provide the basis for the use of accelerations associated with the natural frequency.

b. The third paragraph in this section states that, for multi-degree-of-freedom systems, in lieu of using the peak acceleration value, the actual frequency may be calculated and the corresponding acceleration value may be used. It is not clear whether or not the 1.5 factor is also included in this corresponding acceleration value. Revise this paragraph to provide a clarification of this alternative.

c. The fifth paragraph in this section states that the equivalent static load method of analysis can also be used for small-bore piping. The staff's position as stated in Section 3.9.2.II.2.a(2) of the SRP is that an equivalent static load method is acceptable if justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Furthermore, Section 3.9.2.II.2.a (2) of the SRP states that the design and the associated simplified analysis account for the relative motion between support points and a factor of 1.5 is applied to the peak acceleration of the floor response spectrum. Alternatively, the use of a static load equivalent to the peak of the floor response spectra is acceptable for piping supported at only two points. Revise this paragraph to be consistent with the staff position, or provide the basis for the use of the equivalent static load method of analysis for small-bore piping.

Response:

a. The static equivalent load method is applied to equipment that can be adequately modeled as a one-degree -of-freedom system.

b. When the equipment frequency is calculated the spectral acceleration at this frequency is used in the equivalent static load method. The factor of 1.5 is not applied to this acceleration. This is documented in Reference 210.48-1.

c. The amplification factor of 1.5 will be used for piping analysis with the equivalent static load method, except that a factor of 1.0 is used when there is an axial support. The relative motion between supports is also considered when significant.

References:

210.48-1 WCAP-9903, "Justification Of The Westinghouse Equivalent Static Analysis Method For Seismic Qualification Of Nuclear Power Plant Auxiliary Mechanical Equipment," August, 1980.

SSAR Revision:

3.7.3.5 Equivalent Static Load Method of Analysis

The equivalent static load method involves equivalent horizontal and vertical static forces applied at the center of gravity of various masses. The equivalent force at a mass location is computed as the product of the mass and the seismic acceleration value applicable to that mass location.

The magnitude of the seismic acceleration is established based on the dynamic response characteristics of the subsystem. Subsystems that can be characterized as a single degree of freedom system are designed for accelerations associated with their natural frequency. Seismic acceleration values ~~used for design of multi-degree of freedom systems, which may be in the resonance region of the amplified response spectra curves,~~ are the peak acceleration values from the applicable floor response spectra multiplied by a factor of 1.5, unless a lower factor is justified.

In lieu of using the peak acceleration value, the actual frequency may be calculated and the corresponding acceleration value may be used **without amplification**. In this case, the calculated frequency must be higher than that frequency related to the peak acceleration. Otherwise, the peak acceleration value is used in design. For subsystems and components having fundamental frequencies of 33 hertz or greater, the zero period acceleration is taken as the seismic acceleration value. **This is documented in Reference 23.**

The equivalent static load method of analysis may be used for design of platforms, electrical cable trays and supports, conduits and supports, HVAC ducts and supports, instrumentation tubing and supports, **piping systems**, and other substructures. **This analysis is based on single span models.**

The equivalent static load method of analysis can also be used for design of piping systems, **and instrumentation tubing and supports**, with significant responses at several vibrational frequencies. In this case, a static load factor of 1.5 is applied to the peak accelerations of the applicable floor response spectra. For piping runs with axial supports the acceleration value of the mass of piping in its axial direction may be limited to 1.0 times its calculated spectral acceleration value. The spectral acceleration value is based on the frequency of the piping system along the axial direction. This frequency is determined from the piping mass and the axial support stiffness value. Using this method for piping systems is limited to small-bore (two inches or less in nominal pipe size) piping. **The relative motion between support points is also considered when significant.**

Add Reference to Subsection 3.7.5 as follows:

23. WCAP-9903, "Justification Of The Westinghouse Equivalent Static Analysis Method For Seismic Qualification Of Nuclear Power Plant Auxiliary Mechanical Equipment," August, 1980.

Question 210.63

In Tables 3.9.3-5, 3.9.3-6, 3.9.3-7, and 3.9.3-8 of the SSAR, add a note to state that the method of combination of dynamic responses to loads is in accordance with the recommendations in NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, dated May 1980. In addition, explain how Note 6 in Table 3.9.3-5 and Note 4 in Tables 3.9.3-6, 3.9.3-7, and 3.9.3-8 relate to these recommendations.

Response:

The dynamic loads are combined by the square-root-sum-of-the-squares. This is consistent with industry practice and NUREG-0484.

Dynamic events are postulated initiating events or consequential events that contribute to the design basis mechanical loads for systems, structures, and components. These events result from starting and stopping of pumps, changes in valve position, pipe breaks, and steam/water interactions. Each fluid system is evaluated against the design basis initiating events to determine if an initiating event is a dynamic event for that particular fluid system, or whether the initiating event causes a dynamic event in that particular fluid system. This evaluation determines whether an initiating or consequential dynamic event can be identified for a particular system for an initiating event. The following screening criteria are used to identify the significant dynamic events:

Pumps:
Pump starts and stops resulting from an initiating event, or postulated to occur during an initiating event, shall be identified as consequential dynamic events.

Valves:
Fast-opening or fast-closing valves that change position shall be identified. Fast-opening and fast-closing valves are defined as: remotely-operated valve with a minimum total stroke time of 10 seconds or less; spring-loaded relief valve; check valve. These events shall be identified as initiating or consequential dynamic events.

Pipe Breaks:
High-energy pipe breaks, for those lines not qualified to Leak-Before-Break criteria, shall be identified. For ASME Class 1, 2, and 3 and seismically analyzed B31.1 piping, these events are postulated as initiating events only. For non-seismically analyzed B31.1 piping, these events are postulated as initiating or consequential dynamic events.

Steam/Water Interactions:
Potential interaction or mixing of cold water and hot steam in a line or component shall be evaluated for its potential to cause a dynamic load. This does not include actuation of pressurizer spray.

Dynamic loads that are expected to result from an initiating event are considered in combination with the loads resulting from that event, depending on time phasing of the consequential event and initiating event. Loads resulting from dynamic events will only be combined with loads resulting from an initiating event if the loads can mechanistically and realistically occur simultaneously.

Consequential dynamic loads from an SSE will be combined with SSE depending on the time phasing of the consequential event and the SSE. SSE loads are combined with consequential dynamic loads that can occur as a result of a single pipe break in a nonseismically analyzed piping system.

SSAR Revision: NONE

Question 210.64

If an elastic-plastic method of analysis will be used in the design of any safety-related system, component, or support, identify each applicable item and revise either Section 3.9.1 or 3.9.3 of the SSAR to provide information consistent with the guidelines in Section 3.9.1.II.4 of the SRP.

Response:

For systems where service level D limits are specified for safety-related piping and supports, the method of analysis used to calculate the stresses and deformations shall conform to the methods outlined in ASME Section III, Subsection NF and Appendix F. The inelastic analysis criteria included in Appendix F will be used as an alternative to the procedures of NB-3652 of the ASME Section III code when the inelastic analysis option will provide a more cost effective design. No particular system, component, or support is presently identified for this type of design evaluation.

When an elastic analysis is performed for SSE and an inelastic analysis is performed for pipe break loadings, the following conditions will be satisfied:

a. The stress-strain relationship for the actual type of material undergoing plastic deformation will be used in the analysis.
b. The ultimate strength value at service temperature is not important for this evaluation, because only small strains (< < ultimate strain) will be permitted.
c. The analytical procedures used in the inelastic analysis will be those associated with the Westinghouse proprietary code, WECAN.
d. The applicability and validity of the WECAN code for use in inelastic analysis will be provided to meet the requirements of 10 CFR Part 50, Appendix B and GDC 1.
e. The appropriate interaction of elastic and inelastic components will be demonstrated to provide assurance that system displacements and support deformations do not violate assumptions on which the system analysis is based. This should not be a problem because only piping will be allowed to go plastic, and therefore, the system displacements are the analytically correct displacements.
f. The support load results from an inelastic analysis for Design Basis Pipe Break (DBPB) will be combined with SSE results using Table 3.9-8.
g. The piping stress results from an inelastic analysis will be combined as stated in Tables 3.9-6 and 3.9-7 for the elastic piping elements. For the plastic piping elements, the element strain associated with the SSE condition will be added absolutely with the strain associated with the DBPB condition. The strains will be limited to 1% for strains averaged through the thickness; 2% for strains at the surface, due to an equivalent linear distribution of strain through the thickness; and 5% for local strains at any point.

When an inelastic analysis is performed for SSE and pipe break loadings the method in Table 3.9-6, note 19 is used.

See response to RAI 210.68 for revisions to Tables 3.9-9 and 3.9-10. See response to RAI 210.79 for revisions to Tables 3.9-6, 3.9-7, and 3.9-8.

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SSAR Revision:

3.9.3.1.5 ASME Classes 1, 2, and 3 Piping

The loads for ASME Code Classes 1, 2, and 3 piping are listed in Tables 3.9-3 and 3.9-4. Tables 3.9-6 and 3.9-7 lists the loading combinations. Tables 3.9-9, 3.9-10, and 3.9-11 presents the stress limits.

Piping systems are designed and analyzed for Levels A, B, and C service conditions, and corresponding service level requirements to the rules of the ASME Code, Section III. The analysis or test methods and associated stress or load allowable limits that are used in evaluation of Level D service conditions are those that are defined in Appendix F of the ASME Code, Section III.

Subsection 3.7.3. summarizes seismic analysis methods and criteria. Subsection 3.6.2 summarizes pipe break analysis methods.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Alternate methods for support stiffnesses representation is provided in Subsection 3.9.3.4. Shock suppressors that resist rapid motions are also included in the analysis. The ~~p2~~
solution for the seismic disturbance uses the response spectra method. This method uses the lumped mass technique, linear elastic properties, and the principle of modal superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions. (See Subsection 3.7.3). The stresses resulting from the anchor motions are considered to be secondary and are evaluated to the limits in Table 3.9-11.

The mathematical models used in the seismic analyses of the Class 1, 2, and 3 piping systems lines are also used for pipe rupture effect analysis. To obtain the dynamic solution for auxiliary lines with active valves, the time-history deflections from the analysis of the reactor coolant loop are applied at nozzle connections. For other lines that must maintain structural integrity or that have no active valves, the motion of the reactor coolant loop is applied statically.

When an elastic analysis is performed for SSE and an inelastic analysis is performed for pipe break loadings, the following conditions are satisfied:

a. The stress-strain relationship for the actual type of material undergoing plastic deformation is used in the analysis.
b. The ultimate strength value at service temperature is not important for this evaluation, because only small strains (< < ultimate strain) will be permitted. The ultimate strength value is obtained from the ASME III Code, Appendix I..
c. The analytical procedures used in the inelastic analysis are those associated with the WECAN computer code.
d. The applicability and validity of the WECAN code for use in inelastic analysis meet the requirements of 10 CFR Part 50, Appendix B and GDC 1.
e. The appropriate interaction of elastic and inelastic components will be demonstrated to provide assurance that system displacements and support deformations do not violate assumptions on which the system analysis is based. This is accomplished by allowing only the piping to go plastic and maintaining pipe supports as elastic elements that meet the limits of ASME Section III, Subsection NF and Appendix F. Therefore, the system displacements are the analytically correct displacements.
f. The support load results for DBPB are combined with SSE results using Table 3.9-8, for either elastic or inelastic system analysis.

g. The piping stress results from an inelastic analysis are combined as stated in Tables 3.9-6 and 3.9-7 for the elastic piping elements. For the plastic piping elements, the element strain associated with the SSE condition are added absolutely with the strain associated with the DBPB condition. The strains are limited to 1% for strains averaged through the thickness; 2% for strains at the surface, due to an equivalent linear distribution of strain through the thickness; and 5% for local strains at any point.

When an inelastic analysis is performed for SSE and pipe break loadings the method in Table 3.9-6, note 19 is used.

A thermal transient heat transfer analysis is performed for each different piping component on the Class 1 branch lines larger than 1-inch nominal diameter. The following discussion on the evaluation of cyclic fatigue is not applicable to Class 2 and 3 pipe.

The Level A and B service condition and test condition transients identified in Subsection 3.9.1.1 are included in the fatigue evaluation. For each thermal transient, two load-sets are defined representing the maximum and minimum stress states for that transient.

The primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors, and cumulative usage factors are calculated for the possible load-set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load-sets yielding the highest alternating stress intensity range is determined, and the incremental usage factor is calculated. Likewise, the next most severe combination is then determined, and the incremental usage factor is calculated. This procedure is repeated until the combinations having an allowable cycle of less than 10^{11} are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

Question 210.68

Section 3.9.3.4 of the SSAR does not appear to specifically address allowable stress criteria for active component supports, where active is as defined in Section 3.9.3.2.2. The staff's position is that the stresses and associated deformations in such supports should be low enough to allow operability of the supported component. In Appendix 1A of the SSAR and Revision 1 to WCAP-13054 (under exceptions to Section 3.9.3 of the SRP), exceptions are taken to Position C8 in RG 1.124, and Paragraph B.5 in RG 1.30, which are the bases for the staff's position on this issue. The exceptions in Appendix 1A state that ASME Level C and D Service Limits are acceptable, however, when they are used, any significant deformation that might occur will be considered in the evaluation of equipment operability. Revise Section 3.9.3.4 to reference this exception and provide a more detailed discussion on how this significant deformation will be evaluated for the AP600 to meet the guidelines in Section 3.9.3.II.3 of the SRP. Appropriate revisions should also be made to Tables 3.9-9 and 3.9-10, and Appendix 1A of the SSAR, and to the exception to Section 3.9.2.II.3.a of the SRP in WCAP-13054.

Response:

In order to assure operability for active equipment, including valves, ASME limits for Service Level C loadings will be met for the supports.There will not be significant support deflections for these stress levels since only the outer fibers are permitted to yield.

For pipe supports on lines with active valves the support in the vicinity of the valves will meet either of the following to assure that support deflections are not significant:

a) ASME limits for Service level C loadings

b) Inelastically calculated support displacement will not exceed 1/8 inches

Stress criteria for piping is shown in Tables 3.9-6 and 3.9-7(see response to question 210.79)

WCAP 13054 will be revised to delete exception to SRP 3.9.3.II.3.a.

SSAR Revisions:

3.9.3.4 Component and Piping Supports

The supports for ASME Code, Section III, Class 1, 2, and 3 components including pipe supports satisfy the requirements of the ASME Code, Section III, Subsection NF. The boundary between the supports and the building structure is based on the rules found in Subsection NF. Tables 3.9-3 and 3.9-4 present the loading conditions. Table 3.9-8 summarizes the load combinations. The stress limits are presented in Table 3.9-9 and 3.9-10 for the various service levels.

The criteria of Appendix F of the ASME Code Section III is used for the evaluation of Level D service conditions. When supports for components not built to ASME Code, Section III criteria are evaluated for the effect of

Level D service conditions, the allowable stress levels are based on tests or accepted industry standards comparable to those in Appendix F of ASME Code, Section III.

~~The function of pipe supports for Service Level D conditions is to provide the integrity and operability of the piping system including valves. For certain Service Level D conditions, such as pipe rupture, the system integrity and operability may be demonstrated by allowing the supports to fail. When this is done, the consequences of the support failures are evaluated. This approach is not used for component supports.~~

In order to provide for operability of active equipment, including valves, ASME limits for Service Level C loadings will be met for the supports of these items. When elastic system analysis is used for pipe supports on lines with active valves the supports in the vicinity of the valves will meet either of the following to provide that deflections do not have a significant effect on the elastic piping system analysis.

a) ASME limits for Service Level C loadings

b) Inelastically calculated support displacement will not exceed 1/8 inches

The vicinity is defined as supports that are within a distance from the valve of one-half the standard deadweight span in ASME III, NF.

If an elastic system analysis is not able to satisfy (a) or (b) above to ensure operability, then an inelastic system analysis will be performed which accounts for the inelastic behavior of the piping and supports. This analysis is performed in such a manner that valves and equipment, which are required for operability, remain elastic and satisfies other operability requirements.

Dynamic loads for components loaded in the elastic range are calculated using dynamic load factors, time-history analysis, or any other method that accounts for elastic behavior of the component. A component is assumed to be in the elastic range if yielding across a section does not occur. Local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F for elastically analyzed components are used for Code components.

Dynamic loads for component supports loaded in the inelastic range are calculated using dynamic load factors, time history analysis, or other methods that account for the inelastic behavior. The analysis requirements of paragraph F-1322 of Appendix F of the ASME Code, Section III are satisfied.

The stiffness of the pipe support miscellaneous steel is controlled by one of the following methods so that component nozzle loads are not adversely affected by support deformation:

Pipe support miscellaneous steel deflections are limited for dynamic loading to 1/8 inch in each restrained direction. These deflections are defined with respect to the structure to which the miscellaneous steel is attached. These deflection limits, provide adequate stiffness for seismic analysis and are small enough so that nozzle loads are not affected by pipe support deformation. In this case, the pipe support and miscellaneous steel are represented by a generic stiffness value in the piping system analysis. Rigid stiffness values are used for fabricated supports, and vendor stiffness values are used for standard supports such as snubbers, rigid gapped supports, and energy-absorbing supports. The mass of the pipe support miscellaneous steel is evaluated as a self-weight excitation loading on the steel and the structures supporting the steel.

Table 3.9-9

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

Design/Service Level	Vessels/Tanks Pumps	Piping	Core Supports	Valves, Disk & Seats	Components Supports [c][d]
Design and service level A	ASME Code, Section III NB-3221, 3222	~~ASME Code, Section III NB-3652, 3653~~ See Table 3.9-6	ASME Code, Section III NG-3221, 3222, 3231, 3232	ASME Code, Section III NB-3520, 3525	ASME Code, Section III Subsection NF ~~NF-3221, 3222 NF-3231.1(a)NF-3240~~ (e)
Service level B (Upset)	ASME Code, Section III NB-3223	~~ASME Code, Section III NB-3654~~ See Table 3.9-6	ASME Code, Section III NG-3223, 3233	ASME Code, Section III NB-3525	ASME Code, Section III Subsection NF (e) ~~NF-3223, 3231.1(a) NF-3240~~
Service level C (Emergency	ASME Code, Section III NB-3224	~~ASME Code, Section III NB-3655~~ See Table 3.9-6	ASME Code, Section III NG-3224, 3324	ASME Code, Section III NB-3526	ASME Code, Section III Subsection NF (e) ~~NF-3224, 3231.1 (b) NF-3240~~
Service level D (Faulted)	ASME Code, Section III (see § 3.9.1.4) NB-3225 (No active Class 1 pumps used)	~~ASME Code, Section III~~ See Table 3.9-6 (see § 3.9.1.4) ~~NB-3656~~	ASME Code, Section III (see § 3.9.1) NG-3225, 3335	(b)	ASME Code, Section III Subsection NF (e), (see § 3.9.1) ~~NF-3225, 3231.1(c) NF-3240~~ (f)

Notes to Table 3.9-9:

a. A test of the components may be performed in lieu of analysis.

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

c. Including pipe supports.

d. In instances where the determination of allowable stress values utilizes S_u (ultimate tensile stress) at temperatures not included in ASME Code Section III, S_u shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.

e. ASME Table 3131(a)-1.

f. See subsection 3.9.3.4 for supports for active equipment, valves, and piping with active valves.

Table 3.9-10

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

Design/Service Level	Vessels/Tanks	Piping	Pumps	Valves, Disks, Seats	Component Supports(a)(b)
Design and service level A	ASME Code Section III NC-3217 NC/ND-3310, 3320	~~ASME Code Section III NC/ND-3652, 3653~~See Table 3.9-7	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3510	ASME Code Section III(c) ~~NF-3321 NF-3231 NF-3260~~
Service level B (Upset)	ASME Code Section III NC/ND-3310, 3320	~~ASME Code Section III NC/ND-3653~~See Table 3.9-7	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	ASME Code Section III(c) ~~NF-3321 NF-3231 NF-3260~~
Service level C (Emergency)	ASME Code Section III NC/ND-3310, 3320	~~ASME Code Section III NC-3654~~See Table 3.9-7	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	ASME Code Section III(c) ~~NF-3321 NF-3231 NF-3260~~
Service level D (Faulted)	ASME Code Section III NC/D-3310, 3320	~~ASME Code Section III NC3655~~See Table 3.9-7	ASME Code Section III NC/ND-3400	ASME Code Section III NC/ND-3520	ASME Code Section III(c)(d) ~~NF3321 NF-3231 NF-3260~~

Notes to Table 3.9-10:

a. Including pipe supports.

b. In instances where the determination of allowable stress values utilizes S_u (ultimate tensile stress) at temperatures not included in ASME Code Section III, S_u shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.

c. ASME Table 3131(a)-1.

d. See subsection 3.9.3.4 for supports for active equipment, valves, and piping with active valves.

Question 210.69

Section 3.9.3.4.3 of the SSAR does not provide sufficient information for the staff to conclude that snubber operability will be assured. Revise this section to provide a more detailed discussion which incorporates the guidelines in Section 3.9.3.II.3 of the SRP. In addition, if applicable, provide a commitment to dynamically qualify all large bore hydraulic snubbers.

The discussion of Generic Safety Issue A-13, "Snubber Operability Assurance," in Section 1.9.4.2 of the SSAR should also be revised to reference the revised Section 3.9.3.4.3.

Response:

The SSAR Subsection 3.9.3.4.3 will be revised as shown below to address the SRP guidelines and provide a more detailed discussion of the production and qualification tests. The discussion of Generic Safety Issue A-13, Snubber Operability Assurance, in Subsection 1.9.4.2 of the SSAR already references Subsection 3.9.3.4.3.

SSAR Revision:

3.9.3.4.3 Snubbers Used as Component and Piping Supports

The location and size of the snubbers are determined by stress analysis. Access for the testing, inspection, and maintenance of snubbers is considered in the AP600 layout. The location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment. Snubbers are chosen in lieu of rigid supports where restricting thermal growth would induce excessive thermal stresses in the piping or nozzle loads or equipment. Snubbers that are designed to lock up at a given velocity are specified with lock-up velocities sufficiently large to envelope the highest thermal growth rates of the pipe or equipment for design thermal transients. The snubbers are constructed to ASME Code, Section III, Subsection NF standards.

In the piping system seismic stress analysis, the snubbers are modeled as stiffness elements. The stiffness value is based on vendor stiffness data for the snubber, snubber extension, and pipe clamp assembly. Supports for active valves are included in the overall design and qualification of the valve.

The elimination of the analysis of dynamic effects of pipe breaks due to leak-before-break considerations, as outlined in Subsection 3.6.3, permits the use of fewer snubbers than in plants that were designed without considering leak before break. Also, the AP600 design makes use of gapped support devices to minimize the use of snubbers. The evaluation of those snubbers used as supports is outlined below.

Design specifications for snubbers include:

- Seismic requirements
- Normal environmental parameters
- Accident/post-accident environmental parameters
- Full-scale performance test to measure pertinent performance requirements
- Instructions for periodic maintenance (in technical manuals)

Two types of tests are performed on the snubber to verify proper operation.

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- Production tests on every unit to verify proper operability
- Qualification tests on randomly selected production models to demonstrate the required load performance (load rating)

The production operability tests for large hydraulic snubbers (i.e., those with capacities of 1000 kips or greater) include a) a full Level D load test to verify sufficient load capacity, b) testing at full load to verify proper bleed with the control valve closed, c) testing to verify the control valve closes within the specified velocity range, and d) testing to demonstrate that breakaway and drag loads are within the design limits.

The operability of essential snubbers is verified by the COL holder by verifying the proper installation of the snubbers, and performing visual inspections and measurements of the cold and hot positions of the snubbers as required during plant heatup to verify the snubbers are performing as intended.

Question 210.88

Revision 1 to WCAP-13054 lists an exception to Section 5c of Section 3.10 of the SRP, that states that Westinghouse does not prepare a Seismic Qualification Report (SQR), and that, in lieu of such a report, seismic qualification of equipment is documented in test reports, analysis reports, calculation notes, etc. contained in Westinghouse files. The staff's position is that an SQR should be prepared, and included in the documentation provided by the COL (see Q210.86). Revise this exception to state that the SQR should be submitted by the COL applicant.

In addition, verify the existence of design and analysis documentations of reactor internals, and provide a summary of the analysis results in conjunction with design limits.

Response:

The information recommended to be included in a Seismic Qualification Report is included as part of an equipment qualification data package for each piece of equipment considered. Details of the documentation process are described in AP600 SSAR Section 3.10 and Section 3D.7 (Appendix 3D). The Combined License applicant will maintain this information as the equipment is selected and procured. The data packages will be available for review and audit by the NRC.

The design and analysis documentation for the reactor internals will be available as part of the ASME Code Design Report when construction of the reactor internals is complete.

The position for Section 5c of Section 3.10 of the Standard Review Plan will be revised in the next revision of WCAP-13054.

The verification of the existence of design and analysis documentation of reactor internals and a summary of the analysis results in conjunction with design limits is provided as an attachment to this response for information. The summary includes the components of reactor internals which are not similar or can not be enveloped by previously designed Westinghouse plants. See attachment to this RAI for the summary of stresses.

SSAR Revision:

See the response to RAI 210.86 for the SSAR Revision.

ATTACHMENT TO RESPONSE TO RAI 210.88
SUMMARY OF REACTOR INTERNALS ANALYSIS RESULTS

RADIAL KEY AND CLEVIS INSERT MARGINS OF SAFETY(1) TABLE

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Key Base	Level A+B	Pm	0.18	
		Pm + Pb	0.195	
		Pm+Pb+Q	Large*	
	ΣU	Fatigue	---	0.026< 1.0 Allowable
	Level D	Pm	0.265	
Key Base 45°	Level A+B	Pm	0.013	
Key Bearing	Level A+B	Pm	0.227	
Weld	Level A+B	Pm	0.77	
		Pm+Pb+Q	0.203	
	ΣU	Fatigue	----	0.449 < 1.0 Allowable
	Level D	Pm	1.25	
Clevis Insert Flange	Level A+B	Pm	0.074	
		Pm + Pb	>0.074	
	ΣU	Fatigue	----	Approx.0.00 < 1.0 Allowable
Clevis Insert	Level A+B	Pm	0.29	
		Pm + Pb	Large*	
	ΣU	Fatigue	----	0.80 < 1.0 Allowable
	Level D	Pm	0.326	

Note : (1) Margin of Safety = Allowable Stress Intensity / Calculated Stress Intensity - 1

* = Greater than 10

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Clevis Fasteners	Level A+B	Pm	1.52 (Bearing)	
			4.52 (Shear)	
		Pm + Pb	Large*	
		Pm+Pb+Q	0.816	
	ΣU	Fatigue	----	Approx. 0.00 < 1.0 Allowable
Dowel Pin	Level A+B	Pm	Large*	
		Pm+Pb	Large*	
		Pm+Pb+Q	1.28	
	ΣU	Fatigue	----	Approx. 0.00 < 1.00

* = Greater than 10

UPPER SUPPORT ASSEMBLY MARGIN OF SAFETY TABLE

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Perforated Region	Level A+B	Pm	Large*	
		Pm + Pb	1.746	
		Pm+Pb+Q	1.55	
	ΣU	Fatigue	---	0.0013 < 1.00 Allowable
	Level D	Pm	2.12	
		Pm+Pb	3.67	
Skirt / Flange	Level A+B	Pm	Large*	
		Pm + Pb	0.81	
	Level D	Pm	12.59	
		Pm+Pb	12.67	
Skirt/Plate	Level A+B	Pm	8.0	
Upper Support Flange	Level A+B	Pm	4.77	
		Pm+Pb	3.42	
		Pm+Pb+Q'	3.52	
	ΣU	Fatigue	----	Approx. 0.00 < 1.00 Allowable
	Level D	Pm	2.22	
		Pm +Pb	3.12	

* = Greater than 10

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Skirt to USP Weld	Level A+B	Pm	7.24	
		Pm+Pb	4.62	
		Pm+Pb+Q	0.22	
	ΣU	Fatigue	----	.0143 < 1.00 Allowable
	Level D	Pm	4.19	
		Pm+Pb	2.89	
Outer G/T Location	Steady State	Rotation, θ,rad	0.00131 < 0.0016	

LOWER CORE SUPPORT PLATE ASSEMBLY MARGIN OF SAFETY TABLE

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Inner Perforated Region	Level A+B	Pm	Large*	
		Pm + Pb	0.25	
	ΣU	Fatigue	----	0.255 < 1.00 Allowable
	Level D	Pm	Large*	
		Pm+Pb	0.91	
Transition Rim Region	Level A+B	Pm	10.7	
		Pm+Pb	2.9	
	ΣU	Fatigue	----	0.299 < 1.00 Allowable
	Level D	Pm	16.0	
		Pm+Pb	12.0	
	Level D	Pm	11.37	
		Pm+Pb	17.56	
Center LCP	Steady state	Defl., δ,inch	0.032 < 0.060	

* = Greater than 10

VORTEX RING ASSEMBLY MARGIN OF SAFETY TABLE

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Vortex Ring	Level A+B	Pm	Large*	
		Pm + Pb	21.6	
		Pm+Pb+Q	0.4457	
	ΣU	Fatigue	-----	0.0017 < 1.00 Allowable
	Level D	Pm	Large*	
		Pm+Pb	5.9	
Sec. Core Supt Col.	Level A+B	Pm	81.7	
		Pm + Pb	4.23	
	ΣU	Fatigue	-----	0.018 < 1.00 Allowable
	Level D	Pm	41.8	
		Pm+Pb	1.11	
Core Drop	Level D	Buckling	0.206	
Inner Column Flange	Level A+B	Pm	0.084	
		Pm + Pb	0.352	
		Pm+Pb+Q	3.15	
	ΣU	Fatigue	----	Approx.0.00 < 1.00 Allowable
	Level D	Pm	4.8	
		Pm+Pb	6.29	

* = Greater than 10

CORE BARREL (UPPER AND LOWER) MARGIN OF SAFETY TABLE

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Lower Core Barrel	Level A+B	Pm	1.03	
		Pm + Pb	2.04	
	ΣU	Fatigue	----	0.03 < 1.00 Allowable
Upper Core Barrel	Level A+B	Pm	0.67	
		Pm+Pb	0.92	
	ΣU	Fatigue	-----	0.095 < 1.00 Allowable
	Level D	Pm	1.51	
		Pm+Pb	2.44	
Core Barrel Flange	Level A+B	Pm	0.87	
		Pm+Pb	0.04	
	ΣU	Fatigue	------	0.029 < 1.00 Allowable
	Level D	Pm	1.26	
		Pm+Pb	0.53	
Core Barrel Nozzle	Level A+B	Pm	35	
	ΣU	Fatigue	----	0.894 < 1.00 Allowable
	Level D	Pm	41	

REFLECTOR ASSEMBLY

Section	Load Category	Stress Type	Margin of Safety	Fatigue Usage
Bottom Block	Level A + B	Pm	> 1.18	
		Pm + Pb	> 1.18	
	Usage Factor	Fatigue	-----	0.044 < 1.00 Allowable
	Level D	Pm	> 1.62	
		Pm + Pb	> 1.62	
Center Block	Level A + B	Pm	> 1.18	
		Pm + Pb	> 1.18	
	Usage Factor	Fatigue	-----	0.014 < 1.00 Allowable
	Level D	Pm	> 1.62	
		Pm + Pb	> 1.62	
Top Block	Level A + B	Pm	> 1.18	
		Pm + Pb	> 1.18	
	Usage Factor	Fatigue	-----	0.156 < 1.00 Allowable
Top Block	Level D	Pm	> 1.62	
		Pm + Pb	> 1.62	
Top Flange	Level A + B	Pm	> 1.18	
		Pm + Pb	> 1.18	
	Usage Factor	Fatigue	-----	0.156 < 1.00 Allowable
	Level D	Pm	> 1.62	
		Pm + Pb	> 1.62	

Tie Rod	Level A + B	Pm	4.9	
	Usage Factor	Fatigue	-----	0.0016 < 1.00 Allowable
	Level D	Pm	0.45	
Alignment Bar	Level A + B	Pm	6.39	
		Pm + Pb	2.71	
	Usage Factor	Fatigue	-----	0.013 < 1.00 Allowable
	Level D	Pm	0.32	
		Pm + Pb	7.86	

Question 440.1

a. Provide updated or revised topical reports on the AP600 test program, including WCAP-13277, "Scaling, Design, and Verification of the SPES-2, the Italian Experimental Facility Simulator of the AP600 Plant," and WCAP-13234, "AP600 Long Term Cooling Test Specification." An accurate representation of the facility design, a scaling analysis reflecting that design, a detailed test matrix, and an analysis plan should be provided in the reports.

b. Provide topical reports detailing planned testing for the long-term check valve testing program and the departure from nucleate boiling testing program. Indicate whether the "biased-open" check valves will be tested and, if so, a topical report detailing the test specification should be provided. If the "biased-open" check valves are not to be tested, provide a detailed explanation why such testing is not required.

c. Provide WCAP-12980, "AP600 Passive Residual Heat Exchanger Test Final Report," that is referenced in the SSAR.

Response (Revision 1):

a. WCAP-13277, "Scaling, Design, and Verification of SPES-2, the Italian Experimental Facility Simulator of the AP600 Plant," Revision 1, was provided to the NRC via Westinghouse letter ET-NRC-93-3883, dated May 11, 1993.

WCAP-13234, "AP600 Long Term Cooling Test Specification", Revision 1, was provided to the NRC via Westinghouse letter ET-NRC-93-3883, dated May 11, 1993.

~~WCAP 13277 and 13234 will be updated and forwarded by January, 1993.~~

b. Check Valve Test Documentation

~~A topical report detailing the planned testing for the "long-term" check valve testing program has not yet been prepared. Westinghouse has initiated a review of existing utility information to assess check valve opening performance after being closed at high delta P for a long time, i.e. conditions similar to those which would be experienced by the gravity drain check valves, in order to assure that the test program will address relevant factors.~~ Westinghouse is assessing check valve opening performance for valves that have been closed at high delta P for a long time. These tests are being performed at existing PWR sites during refueling outages. The report on the first of these in-situ tests is documented in WCAP-14045, "In Situ Check Valve Test Report (Fall Outage '93)," which was transmitted to the NRC via Westinghouse letter NTD-NRC-94-4120, dated May 4, 1994.

"Biased-open" check valves will not be tested in this program. These valves are open and are not exposed to high differential pressure.

Additional information on check valve testing is provided in WCAP-13560, "Advanced Plant Check Valve Study", Rev. 0, which was provided to the NRC in letter ET-NRC-93-3801, "AP600 Testing Reports (WCAP-13567, WCAP-13566 and WCAP-13560), from N. J. Liparulo to Dr. Thomas Murley, January 25, 1993.

~~Information on DNB Testing will be provided in January 1993.~~

DNB Test Documentation

DNB testing was performed in accordance with WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" which was provided to the NRC as a topical report for review under letter NS-NRC-90-3482, to V. H. Wilson from W. J. Johnson, April 2, 1990. Further information has been provided in the following submittals:

ET-NRC-92-3702, "Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process", from N. J. Liparulo to R. C. Jones, June 8, 1992.

ET-NRC-92-3723, "Supplement to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process", from N. J. Liparulo to R. C. Jones, July 17, 1992.

ET-NRC-93-3819, "Final Responses to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process", from N. J. Liparulo to R. C. Jones, February 8, 1993.

ET-NRC-93-3984, "Test Matrix for AP600 Departure from Nucleate Boiling Tests," from N. J. Liparulo to R. W. Borchardt, October 8, 1993.

The final test report on the AP600 DNBR tests will be transmitted to the NRC in August 1994.

c. PRHR Test Documentation

WCAP-12980, AP600 PRHR HX Test Final Report", was provided to the NRC in letter ET-NRC-92-3779, "Submittal of AP600 Design Certification Material required by 10 CFR 52.47", from N. J. Liparulo to Dr. Thomas Murley, December 15, 1992. ~~"AP600 Passive Residual Heat Exchanger Test Final Report," will be forwarded by January, 1993.~~

SSAR Revision: NONE

Question 440.11

Revision 0 of WCAP-13342, "AP600 Automatic Depressurization System Test," is dated January 1991. An updated version of this test specification should be provided, incorporating any changes in the design or test plans for the test articles in Phase A and Phase B, particularly as a result of changes in the AP600 plant design.

Response (Revision 1):

~~WCAP-13342, AP600 Automatic Depressurization System Test is currently being updated, specifically to incorporate additional or revised information for the Phase B of the test program. This revision will also include changes to the Phase A test program. The revised WCAP will be provided in January, 1993.~~

WCAP-14112, Revision 0, "AP600 Automatic Depressurization System - Test Specification (Phase B1)" was provided to the NRC via Westinghouse letter NTD-NRC-94-4178.

WCAP-13891, Revision 0, "AP600 Automatic Depressurization System Phase A Test Data Report," was provided to the NRC via Westinghouse letter NTD-NRC-94-4176, dated June 20, 1994.

SSAR Revision: NONE

Question 440.52

The staff has determined that the scaling report on the core makeu tank (CMT) does not provide sufficient information to demonstrate that the CMT separate-effects test will represent the processes occurring in the AP600 plant during operation of the CMTs. In addition, there appear to be inconsistencies in the report. Address the following concerns:

a. The description of the operation of the CMT described in Section 1-2 of the report appears to be inconsistent with Figure 1-1. On page 4, it states that "the discharge line isolation valves are normally open." This is inconsistent with the drawing referenced in the discussion (Fig. 1-1), which shows the valves closed. In addition, this description appears to be incorrect, since with the CMT at reactor coolant system (RCS) pressure, leaving the discharge valves open would establish a circulation path from the reactor vessel through the pressurizer to the CMTs, and the tanks would drain slowly. From the AP600 SSAR, it has been understood that all CMT isolation valves are normally closed (but fail open), and open on the various CMT actuation signals. Clarify this inconsistency.

b. Clarify the nomenclature in Chapter 2 of the report (from page 141) used for the conservation equations for the CMT. Specifically, in the momentum equation, as shown in Equations 2.2 and 2.7, and in the dimensionless parameters derived therefrom, the terms a_c and ℓ appear, which are defined as core area and subchannel length, respectively. While the equation *per se* is appropriate, insofar as consistency with the stated assumptions (1-D steady-state momentum) is concerned, the reference to core parameters in the CMT equation does not appear appropriate. The symbology or the nomenclature should be corrected.

c. In Section 2-2 of the report, in the discussion of convective heat transfer during the recirculation mode of CMT operation, it is stated (on page 30) that the Reynolds numbers for both the plant's CMT and the test article are about 6000-7000, and that this is a "turbulent flow regime such that a turbulent heat transfer convective correlation such as Dittus-Boelter...is applicable." The staff concludes that a Re value of 6000-7000 places the system in the laminar-to-turbulent transition regime, not a fully turbulent one. There are very few heat transfer correlations developed for this regime, and this situation is further complicated by the fact that the flow is natural convection, in which the velocity/temperature profiles can be considerably distorted from those which would exist in turbulent forced convection. In addition, for Dittus-Boelter specifically--and all other similar heat transfer correlations (Colburn, McAdams, Seider-Tate)--the applicable range given is $Re > 10,000$ (and, it should be noted, the aspect ratio range is $L/D > 60$ for Dittus-Boelter, which is also much larger than exists in either the plant or the test article), so that the use of this correlation for the comparison/scaling of convective heat transfer appears to be incorrect. As a practical matter, however, this effect may be second order in many cases, and errors in its scaling may be of low importance overall. However, it is important that correlations with appropriate thermal-hydraulic and geometric ranges be employed in this study, because there may in fact be instances where the effects are not second-order. Address these concerns.

The following is additional information and clarification to Part c provided in a letter from the NRC dated March 24, 1994.

1. There appear to be errors in Revision 0 of WCAP-13963, "Scaling Logic for the Core Makeup Tank Test" The text on page 2-12 still refers to use of the Dittus-Boelter equation, though that equation is not employed in the discussion that follows. There is also an inconsistency in the exponent on the Rayleigh (not Raleigh) number in Equations 2-37 and 2-39, and the term R in the denominator of Equation 2-37 is not defined (it appears to be a typographical error). Address these inconsistencies.

2. The staff questions the technical aspects of the approach used, both with regard to consistency of definitions and appropriateness of the heat transfer correlation employed. The Reynolds number based on diameter is employed to demonstrate that both the plant core makeup tank (CMT) and the test article are in turbulent flow. While the relatively small diameter of the test article may make it look like a pipe, the use of a diameter- based Reynolds number for the plant CMT appears to be inappropriate, due to its very large diameter and extremely low flow rate; a length-based Reynolds number may be more appropriate, which also changes the criterion for laminar-to-turbulent transition. Further, the staff concludes that both may not be in turbulent flow because (a) the value of Re (based on diameter) for the test article falls within what is generally considered to be a transition regime from laminar to turbulent flow, and (b) the very small aspect ratios of both the CMT (1.8/1) and the test article (about 6/1) would not allow the flow to become fully developed in any event. In light of the above reasoning, the use of length-based Grashof and Nusselt numbers (as in the correlation cited in Equation 2-37) appears to be appropriate, but it appears to be inconsistent with the use of diameter-based Reynolds numbers (including calculation of the Gr/Re^2 ratio) to determine the single-phase flow regimes. In addition, no information is included in the report to allow the staff to determine whether the correlation cited is appropriate for use over the range of thermal-hydraulic and geometric parameters characteristic of both the plant and test CMTs. The staff recommends that Westinghouse review this section of the report and reevaluate its approach to the question of wall-to-fluid heat transfer.

d. The approach taken in the scaling analyses presented in Chapters 2 and 3 of the report mirrors that used by Oregon State University (OSU) in the scaling of the APEX facility. However, there do not seem to be any real conclusions drawn about the applicability of the test facility results to plant behavior, and, further, there is little or no discussion about what has been left out of the analyses at the start. For example, the scaling of both recirculatory and draining behavior is based on a one-dimensional momentum equation. This is certainly applicable for the test facility, since multi-dimensional effects are suppressed by the small diameter of the test article. However, it may not be the case for the full-size CMT, where the diameter is large both in absolute terms and as compared to the tank length. The report does not address distortions due to suppression of multi-dimensional behavior in the test facility, nor is there an order-of-magnitude analysis showing the relative importance (or lack thereof) of multi-dimensional effects. In addition, for many of those aspects of CMT operation not specifically analyzed, qualitative or intuitive arguments are used, with no supporting justification. An example is found on page 32, in the discussion of liquid mixing and flashing effects during recirculation. The fact that things "seem like" they should behave similarly is not a substitute for a quantitative scaling study. Distortions and the uncertainties that they introduce into the scaling analysis should be dealt with in a quantitative manner, where possible. Address these concerns.

e. The scaling analysis for draining behavior presented in Chapter 3 of the report does not provide an order-of-magnitude analysis to separate the important phenomena/scaling parameters from the unimportant ones. In addition, the scaling parameters derived for this mode of operation appear to vary over a very wide range (factors of 6 or more) during a given experiment, but the significance of such a variation is not addressed. Also, the scaling of steam jet behavior should be updated to address the incorporation of the steam "distributor" at the CMT inlet. Dependence of the mixing length on steam diffuser design, scaling of the mixing length between the test facility and the CMT, behavior of steam when the diffuser is uncovered, and the means and impact of "scaling up" the diffuser should be addressed. The potential effects of non-condensible gases on condensation behavior and the resultant impact on CMT behavior should be addressed in a quantitative fashion. Furthermore, it appears that the mass conservation equation employed in Chapter 3 includes the assumption that all steam that enters the CMT condenses. Were this to be the case indefinitely, the CMT would not drain, because some vapor (steam or non-condensible gas) will fill the space vacated by the water as the tank empties. As the top layer of water in the CMT is heated, condensation will slow, and some steam (or other gas) will remain at the top of the tank, accumulating as the CMT drains. The equations used to assess draining behavior should explicitly reflect the physical processes that occur during that period of CMT operation; this includes not only condensation, but also the development, growth, and subsequent behavior of a thermally stratified layer of liquid as draining continues, including the possible flashing of the hot fluid as the system is depressurized. The proper scaling of these effects on CMT draining behavior should be addressed.

f. Provide a discussion of the significance of the plots of predicted plant and model behavior, and dimensionless parameters and their ratios, that are presented in Chapter 3 of the report. It appears that for certain conditions at certain points in a given type of experiment, the model will represent approximately the behavior expected in the plant, but it is not clear that such a conclusion can be extended to the range of conditions under which the CMTs are expected to operate, that the assumptions made (e.g., mixing depth) are realistic, nor that the idealized test conditions used for the analyses adequately represent the conditions that would exist during periods when CMT operation is most important. Address these concerns, and provide justification for these assumptions and for the selection of the test conditions.

g. Discuss how the CMT test results will be related to CMT operation in other integral test facilities, and how the overall results will be implemented in the codes.

Response:

a. In the original SSAR AP600 design, the isolation valves on the CMT cold leg balance line and on the CMT discharge lines were closed and would open on a CMT activation signal. This is what is stated on page 1-1 of the scaling report (Reference 440.52-1). Figure 1-1 of the scaling report also shows the valves closed. In the AP600 design, presented to the NRC in April 1994, the pressurizer-to-CMT balance line was eliminated and the cold leg isolation valves are normally open. These changes have been documented in a design report to the NRC. Isolation of the CMT is achieved by keeping the CMT discharge valves closed for normal operation. These valves open on a CMT actuation signal. These changes in the CMT design and logic will be included in a revision to the CMT scaling report. This revision will be issued by December, 1994.

b. The nomenclature for the equations given in Section 2 is presented in Section 7-1. The term a_c is defined as the CMT area and ℓ is the heated length of the CMT.

c. Response (including questions 1 and 2)

There is a typographical error in the exponent of Equation 2-39. The power should be 0.4 as given in Equation 2-37.

As suggested by the question, the heat transfer coefficients during the recirculation mode have been re-evaluated. A forced convection heat transfer coefficient was evaluated assuming flow over a flat plate with the hot CMT liquid layer being the characteristic dimension. The correlation used is from Kreith and Bohn (Reference 440.52-2), and is given as

$$Nu_x = 0.332 Re_x^{1/2} Pr^{1/3}$$

for the local heat transfer and

$$Nu_L = 0.664 Re_L^{1/2} PR^{1/3}$$

for the surface average heat transfer coefficient. Using the length or depth of the heated thermal layer as the characteristic dimension, the Reynolds and Grashoff numbers can be calculated and the ratio of Gr/Re^2 can be compared for different thermal layer thicknesses. These calculations were performed for a range of CMT draining velocities which were calculated from the AP600 SSAR analysis as well as the draining velocities that are simulated in the CMT test. The SSAR CMT draining velocities are small, and range from 0.008 to 0.016 ft/sec, while the CMT test draining velocities will range up to a velocity of 0.127 ft/sec. The Gr/Re^2 ratio was calculated using the largest drain velocity of 0.127 ft/sec. The calculated ratios are given in a table below for different assumed thermal layer thicknesses. As the results indicate, the natural convection flow will dominate for all heated layer thicknesses such that the analysis presented on pages 2-11 to 2-13 of WCAP-13963 remains correct. These revised calculations using the flat plate forced convection heat transfer coefficient will be included in a revision to WCAP-13963.

CALCULATING THE Gr/Re^2 RATIO WAY THE TEST VELOCITIES

Heated Layer Thickness X	Gr	Re (.127 ft/sec)	Gr/Re^2
0.5	3.49×10^{10}	44354.7	17.7
1.0	27.92×10^{10}	88713.94	35.5
3.0	753.9×10^{10}	266142.78	106.34

Conclusion, Free convection dominates even for the fastest drain rates

d. With regard to the conclusions drawn from the scaling study; there were conclusions presented at the end of the major sections of the report. For the recirculation phase, the CMT test captured all the phenomena of interest and would model the plant CMT behavior. For the draining portion of the transient, there were limitations that were indicated in the report. There is a time scale difference in the test CMT relative to the plant CMT due to the wall thickness difference. This was specifically analyzed in the report. Also, the interfacial condensation correlation used tended to bias the results and introduce a scale effect. These points were also discussed in the report. The conclusion on the draining portion of the scaling analysis was that the CMT test will capture the phenomena of interest, for most transient time periods. There will be periods when the test will not accurately represent the plant CMT behavior due to the thinner walls used in the test. However, as indicated in the report, there are sufficient time periods in which the tests is accurately representing the plant CMT such that this data can be used to develop or verify the models or correlations needed to model the AP600 plant CMT.

The question on the use of a one-dimensional analysis for the CMT analysis and its validity is a separate issue. The inclusion of the CMT diffuser has reduced the occurrence of three-dimensional effects for the majority of the CMT tank. Since the draining velocities are very small, the axial velocities in the tank are typically 0.008 to .127 ft/sec. Therefore, the flow down through the tank will be very one-dimensional. The region in which the three-dimensional flow can occur is at the CMT diffuser at the very top of the tank. The flow area of the diffuser has been specifically sized to obtain a low liquid velocity (approximately 0.1 ft/sec) at the wall of the CMT both for the CMT test and the AP600 CMT. There is a recirculation flow pattern which is developed at the top of the CMT tank as either steam or liquid flow radially enters the tank. The recirculation flow pattern has been observed to be confined to the top of the tank. It continues until the liquid at the tank top is heated and the tank begins to drain. Once the CMT liquid is heated, steam condensation is reduced and the resulting steam velocity through the diffuser is reduced which reduces the mixing and recirculation. A more complete discussion of the recirculation and the scaling of the CMT diffuser will be included in the revision to the CMT scaling logic report WCAP-13963.

There was a question on the lack of discussion on the flashing effects between the test CMT and the AP600 CMT. In the recirculation mode of operation, the development of the heated thermal layer is very similar between the test and the plant as seen in Figures 2-2 to 2-9 of the scaling report. Since the fluid conditions are preserved in the test relative to the plant CMT, if the depressurization rates of the test are similar to the plant the, flashing effects will be captured in the tests. The depressurization rate is a test parameter in the CMT tests and is controlled over a range to approximate that in the plant. Therefore, the flashing effects will be similar between the test and the plant, and this phenomena will be captured in the CMT tests.

e. The CMT report will be revised to address the concerns expressed on the draining behavior or the CMT. The specific Questions raised in the question will be addressed and the CMT diffuser scaling analysis will also be included. The analysis that has been performed is quasi-steady in which a thermal layer thickness is specified and the mixing is assumed to be perfect within this layer. This is a reasonable approximation based on the CMT data obtained to date.

The flashing effects were not specifically developed for the CMT drain down analysis provided in Section 3 of the CMT scaling study. The mixing layer, which is what would flash once the CMT depressurized, was ranged over the thicknesses expected in the CMT tests and the plant. Since the fluid conditions are preserved in the tests relative to the plant, the flashing effects observed in the test, over the range of the mixing layers, would be expected to be similar to the plant behavior. This will be discussed in more detail in the revised scaling report.

f. The Figures presented in Section 3 of the report were sensitivity studies at conditions which were typical of those conditions that the plant CMT would be expected to operate. The pressure selected is typical of the system pressure for a small break LOCA when the primary system pressure has equilibrated with the secondary system pressure. Both the liquid level and the mixing depths were parameters that were selected to examine the sensitivity of the test CMT to the plant CMT. As the plots indicate, the test will capture the thermal hydraulic phenomena of interest for selected time periods of the transient. This will be sufficient to develop or verify the models or correlations used in the plant for the CMT. In particular Figures 3-34 and 3-37 shows the dimensionless π group ratios (test or model to plant) for wall condensation and liquid surface condensation. As figure 3-34 indicates, the model wall condensation is approximately the same as the plant for the first 700 seconds. After this time period the thinner walls of the model heat-up and the wall condensation significantly decreases. Figure 3-37 shows the dimensionless π ratios for the liquid surface condensation (model or test to plant). For very thin mixing depths, the ratio varies significantly as the water heats up faster in the plant calculation relative to the test calculation. This is caused by the choice of the condensation heat transfer correlation which is scale dependent as discussed in the report. However, for thicker more typical mixing depths, the agreement is much better and indicates that the test will simulate the surface condensation effects expected in the plant for a significant time period. The same behavior is observed in Figure 3-40 and 3-43 for a CMT tank level of 95%.

Since design changes have occurred in the CMT balance lines and the initiation of the PRHR, The conditions presented in the existing scaling report will be reviewed to investigate if they should be supplemented by additional calculations. The additional calculations, if needed, will be included in the revised CMT scaling report.

g. The CMT tests are separate effects experiments which concentrate and isolate particular phenomena which will also occur in the integral systems tests such as SPES and OSU. The CMT tests will be used to develop or verify existing models and correlations for wall and liquid surface condensation with the prescribed boundary conditions of the experiments. The CMT tests will be modeled with both NOTRUMP and WCOBRA/TRAC to examine the predictability of the existing models and correlations in these codes which are used in the SSAR analysis. If the agreement is inadequate, these models and correlations will be refined and improved to better predict the CMT tests. Once satisfaction is obtained with the CMT data, the same model will be used to predict the OSU and SPES tests. In this fashion, the CMT tests are used for model development while the integral tests are for model validation.

References:

440.52-1 WCAP-13963, " Scaling Logic for the Core Makeup Tank Test," transmitted to NRC via Westinghouse letter NTD-NRC-94-4068.

440.52-2 Kreith, F. and M. S. Bohn, Principles of Heat Transfer, 4th Edition, pg 216-217, (1986).

SSAR Revision: NONE

Question 440.133

As a part of design features to mitigate air binding of the RNS pump during mid-loop operation, Section 5.4.7.2.1 of the SSAR states that the RNS employs a step-nozzle connection to the RCS hot leg, that will (a) substantially lower the RCS hot leg level at which a vortex can occur in the RNS pump suction line due to the lower fluid velocity in the hot leg nozzle, and (b) limit the maximum air entrainment into the pump suction, if a vortex should occur, to no greater than 5 percent. This value has been demonstrated experimentally. Provide a discussion of the actual design configuration of the AP600 stepped nozzle connection, the experiment(s) and associated data, as well as any analysis that demonstrate the adequacy of this design to minimize vortex formation and air entrainment into the RNS pump suction.

Response:

Westinghouse test report, APWR-0452, "AP600 Vortex Mitigator Development Test for RCS Mid-Loop Operation" was provided to the NRC via Westinghouse letter NTD-NRC-94-4191, dated July 6, 1994. This report describes a comprehensive test program performed to investigate the vortex behavior at the RNS line/hot leg junction of the AP600 during mid-loop operation. The AP600 RNS line/hot leg junction employs a step nozzle design which conforms with the recommendations of the test report.

SSAR Revision: None
PRA Revision: None

Question 440.171

To address the concern of increasing the potential for operator error as a result of added displays and controls in the control room of the high point vent systems, Section 5.4.12 of the SRP states that a human-factor analysis should be performed taking into consideration the use of this information by an operator during both normal and abnormal plant conditions, integration into emergency procedures and operator training, and other alarms during emergency and need for prioritization of alarms. Confirm that the displays and controls of the high point vent will be included as part of this human-factor analysis in the design of the displays and controls in the control room.

Response:

The design of the control room displays and controls for the high point vent systems will be included as part of the human-factor analysis. Chapter 18 of the AP600 SSAR describes the human factor engineering design approach to be applied to the man-machine interface system design. Refer to sections 18.6.5, 18.6.7, 18.8.2.1.1.4 and 18.8.2.1.2 for descriptions of the human decision making model, the function based task analysis and their application to the design of the AP600 man-machine interface.

SSAR Revision: None

Question 440.245

Provide the following information regarding the onsite dc system:

a. Section 8.3.2.1.1.1 of SSAR states that class 1E divisions A and D have one battery bank each while divisions C and B have two. On the other hand, Figure C17-1 of the PRA shows that the 4 divisions have very similar structures. Explain the inconsistency.

b. Tables 8.3.2-1 to 8.3.2-4 of the SSAR list the loads on the class 1E buses. Are there similar tables for the non-class 1E dc buses?

Response:

a. The numbers of battery banks per class 1E division as stated in SSAR Subsection 8.3.2.1.1.1 are accurate. In the upcoming revision of the Level 1 AP600 PRA, the dc power distribution system modeling will reflect the current dc power system design.

b. The non-class 1E dc load assignments are shown on the drawings for the non-class 1E dc buses, SSAR Figure 8.3.2-3, Sheets 1 and 2.

SSAR Revision: NONE

PRA Revision: The PRA will be revised by December 31, 1994

Question 480.49

Note 12 of Sheet 1 of Figure 9.4.7-1 and Table 6.2.3-1 of the SSAR do not clearly describe the provisions for Type C testing for the metal-seated containment air filtration system and chilled water system isolation valves. Provide a description of the design and Type C testing proposed for these valves.

Response:

The containment air filtration system and chilled water system isolation valves subject valves are metal seated butterfly valves with a double seal around the periphery of the valves disk. Integral leakage connections to perform type C leakage testing are provided between the seals in order to pressurize between the seals and thus determine the valve leakage. The testing configuration is much like the leakage detection provisions provided for equipment hatches, fuel transfer blind flange or spare connections where double seals are provided with pressurization capabilities between the seals for leakage detection.

SSAR Revision: NONE

Question 480.50

Generally, the P&IDs in the SSAR depict the test, vent, and drain (TV&D) valves provided for Type C testing. However, the service air P&ID does not. If TV&D connections are not shown on a system P&ID, does that mean that they will not be installed and that Type C testing of the isolation valves is not planned?

Response:

The service air system P&ID is presently under revision and will incorporate the necessary features to perform Appendix J, Type C testing. The features will be similar to other typical penetrations with test vents outboard of the outer containment isolation valves, test connections inboard of the inner isolation valves and a second test connection between the isolation valves for penetrations that utilize a check valve as an inner containment isolation valve. SSAR Figure 9.3.1-1 "Compressed and Instrument Air System Piping and Instrumentation Diagram" will be revised and included in Revision 2 of the SSAR.

Question 480.52

Table 6.2.3-1 of the SSAR indicates that an 8-inch demineralized water transfer penetration is Type C tested with air in the forward direction. Figure 9.2.4-1 does not depict the necessary TV&D connections. Also, Table 6.2.3-1 indicates isolation on a "T" signal; however, the valve is indicated to be a manual valve. Clarify these tables or the drawing.

Response:

The demineraled water system P&ID is presently under revision and will incorporate the necessary features to perform Appendix J, Type C testing. The features will be similar to other typical penetrations with test vents outboard of the outer containment isolation valve, test connections inboard of the inner isolation check valve and a second test connection between the isolation valves. SSAR Figure 9.2.4-1 "Demineralized Water Transfer and Storage System Piping and Instrumentation Diagram" will be revised and included in Revision 2 of the SSAR.

The outboard containment isolation valve is a locked closed manual valve and will not recieve a containment isolation signal.

SSAR Revision:

See the response to RAI 480.61 for revisions to Table 6.2.3-1 to reflect the above changes

Question 480.56

An SRP criterion for use of relief valves as containment isolation barriers is that the setpoint be ≥ 105 percent of the containment design pressure. Confirm that the relief valves of Table 3.2-1 of the SSAR meet this criterion. Will these valves open under severe accident (Service Level C) conditions?

Response:

The following relief valves are identified as containment boundaries in SSAR Table 6.2.3-1:

CVS	Penetration P05	Valve V056
CVS	Penetration P06	Valve V042
RNS	Penetration P19	Valve V021
SGS	Penetration P23/P24	Valves V030A,B; V031A,B; V032A,B

The relief valve set pressures for each of the above valves will be in excess of both 105% of containment pressure and a containment pressure relating to ASME Service Level C stress limits. The valves will therefore not open under design basis accident conditions or severe accident design conditions.

SSAR Revision:

6.2.3.1.3 Additional Requirements

M. Relief valves that serves as a part of the containment boundary have a set pressure in excess of the 105 percent of containment design pressure.

PRA Revision: None

Question 480.60

The terms "manual" and "remote manual" are used in Table 6.2.3-1 of the SSAR to describe isolation device actuation modes. The staff understands that the term "manual," when used to describe the primary actuation mode, is used to mean manual operation from the control room. In addition, it is the staff's understanding that the term "remote manual," when used to describe the secondary mode of actuation, means manual actuation at a control station other than the main control room. Confirm or clarify this understanding.

Response:

The following clarification has been added to the "Explanation of Heading and Acronyms for Table 6.2.3-1"

Actuation Mode Primary/Secondary:

Primary closure mode of operation / Secondary closure mode of operation: Types:

manual:	manual manipulation at the valve (e.g. handwheel)
self:	self controlled valve (e.g. check or relief valve)
automatic:	power operated valve closes automatically on a safety related signal
remote manual:	power operated valve requiring remote operator action (e.g. from the MCR)
N/A:	isolation devices without manipulation capability (e.g. flange)

SSAR Revision:

A revised version of Table 6.2.3-1 is provided in response to RAI 480.61.

Question 480.66

This question pertains to Westinghouse's statement of conformance to paragraph 6.2.1.1.A of the Standard Review Plan, "PWR Dry Containments, Including Subatmospheric Containments," that is identified on page 6-6 of Revision 1 to WCAP-13054, "AP600 Compliance with SRP Acceptance Criteria."

a. What is Westinghouse's position relative to required margin between maximum calculated and design pressure? If none is considered, then how does Westinghouse ensure there will be no differences between the current design stage and the "as built" of the actual plant.

b. Provide justification for using best-estimate heat transfer coefficients in DBA calculations. Provide an uncertainty analysis for the worst case DBA accident that takes these heat transfer coefficients into account.

c. The staff understands that other parameters that are inputs to the DBA calculations are conservative, specifically boundary conditions and initial conditions such as input mass and energy release rates. Confirm this understanding.

Response:

a. The results of the containment integrity analyses presented in Chapter 6 of the AP600 SSAR indicate 8% margin to the design pressure limit for the limiting case. The limiting cases have been reanalyzed in reference 480.66-1 and indicate that there is approximately 19% margin to the containment design pressure limit. These results indicate ample margin is available in the design, particularly in view of the fact that the AP600 is a standardized design under Part 52.

b. The heat transfer correlations used in the design basis analysis calculations are taken from open literature and applied via the Westinghouse-GOTHIC code. The free convection component of the heat transfer correlation is calculated using the McAdams correlation. The Colburn correlation is used for natural convection heat transfer that is better represented by a forced convection heat transfer coefficient due to higher steam/air velocities. These correlations are applied directly; however, there is the capability to apply heat transfer coefficient multipliers to the calculation. To demonstrate the sensitivity to an uncertainty of 10%, the limiting loss of coolant accident case was reanalyzed. The heat and mass transfer coefficients were reduced to 90% of the calculated value and the transient was rerun. The reduction in the heat transfer coefficient should not affect the blowdown peak pressure, but should cause the containment pressure and temperature to increase later in the transient.

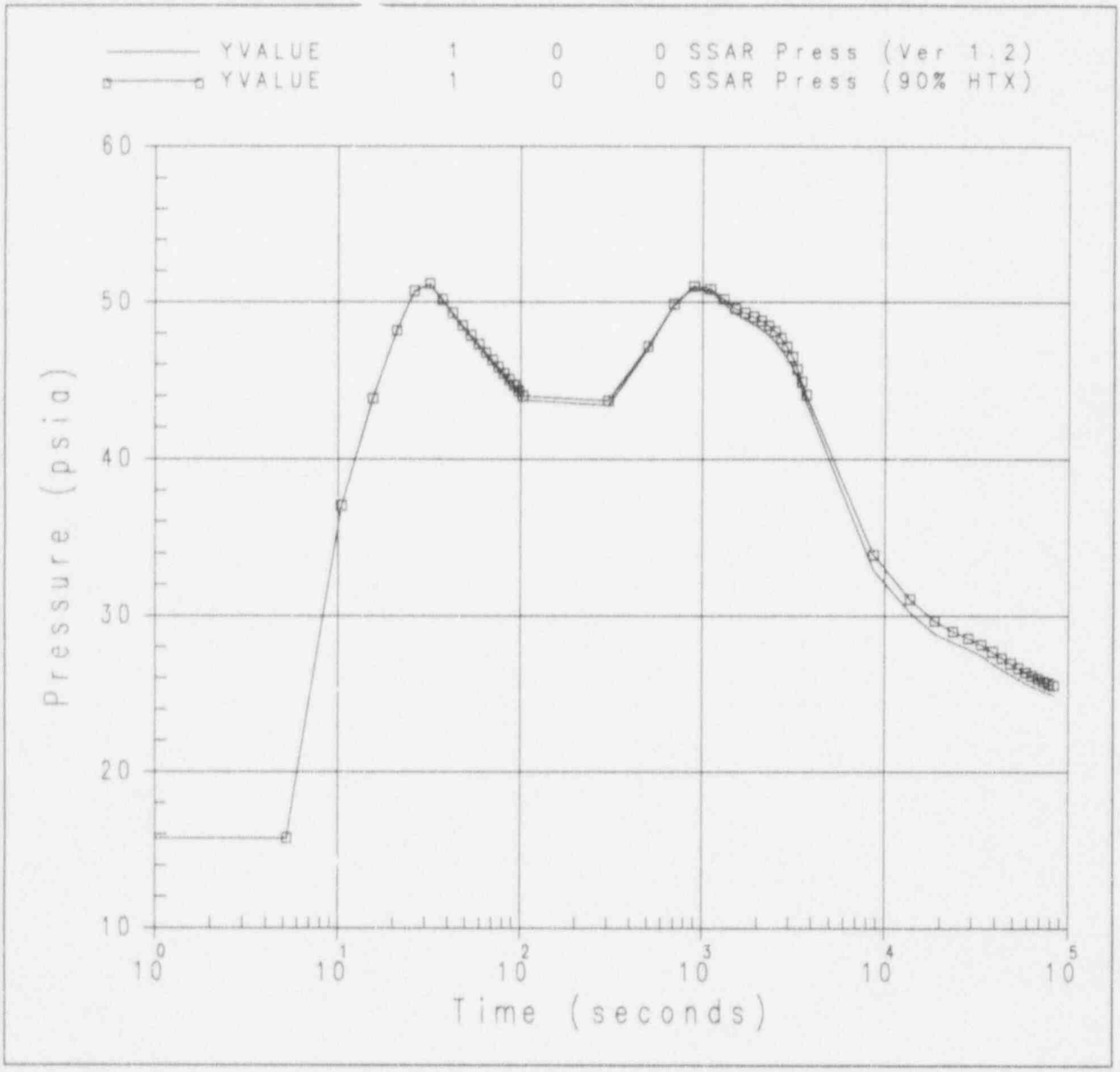
The containment pressure and temperature response calculated with the reduced heat transfer coefficients has been plotted and compared with the limiting loss of coolant accident case. The peak containment pressure (which occurs at the end of blowdown) increased by 0.05 psi and the containment pressure at 24 hours increased by 0.6 psi (and is still less than half the containment design pressure).

c. The inputs to the Design Basis Analysis calculations, specifically the boundary and initial conditions, are conservative.

References

480.66-1. Westinghouse letter NTD-NRC-94-4174, "AP600 Passive Containment Cooling System Design Basis Analysis Models and Margin Assessment", June 30, 1994.

SSAR Revision: NONE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Time (seconds)
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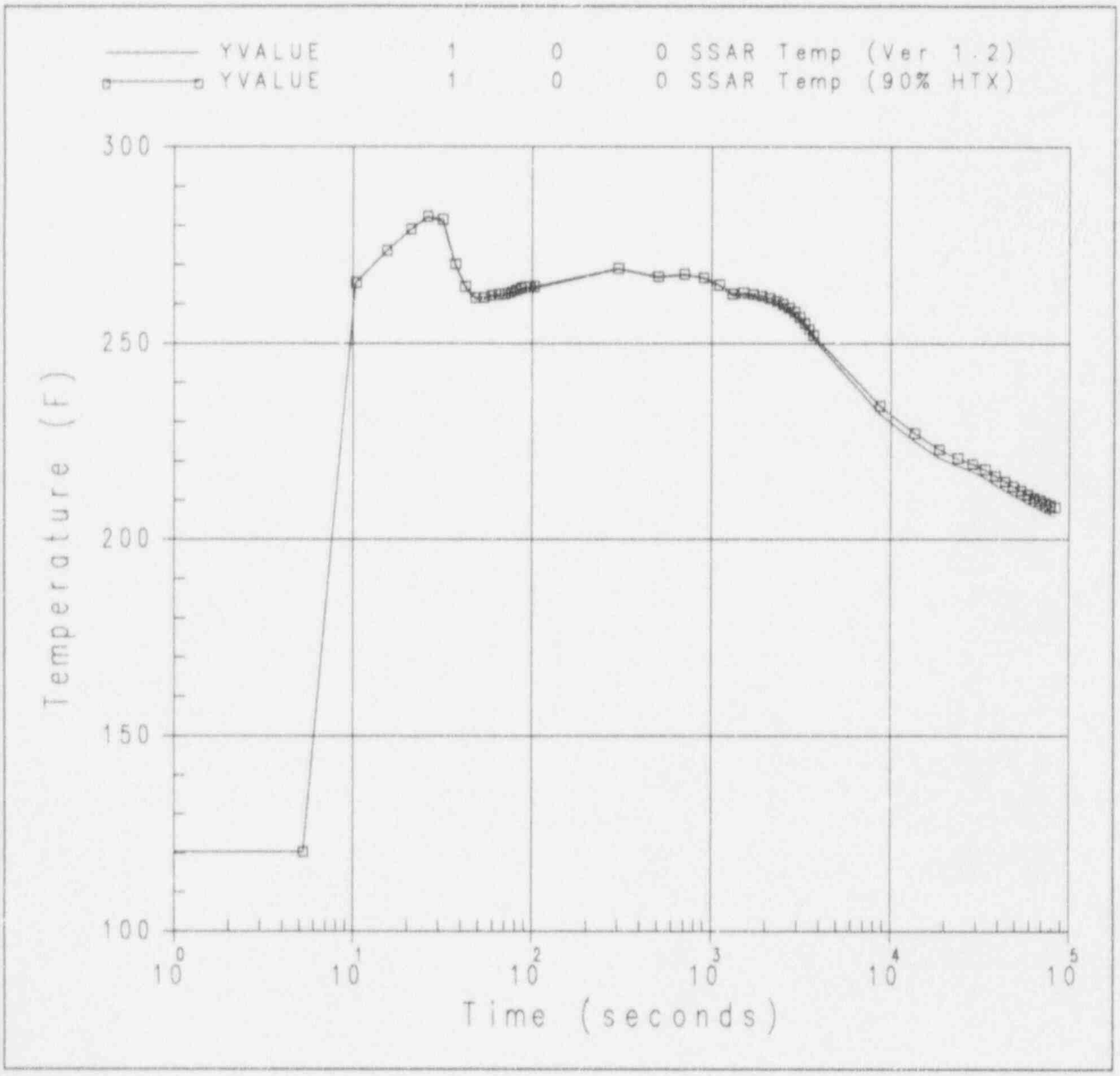
10^4
10^5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Temperature (F)
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10^1
10^2
10^3
10^4
10^5
Time (seconds)

Question 480.68

This question pertains to Westinghouse's statement of conformance to paragraph 6.2.1.2 of the Standard Review Plan, "Subcompartment Analysis," that is identified on page 6-8 of Revision 1 to WCAP-13054, "AP600 Compliance with SRP Acceptance Criteria."

The current design basis is based on the use of leak-before-break. Are the subcompartment analyses and the associated wall capacities to be established by postulating the break of a 3" high energy line in each subcompartment, regardless of whether that subcompartment has any such lines in it?

Response:

No, please see the response to RAI 210.76.

SSAR Revision: NONE

Question 480.69

This question pertains to Westinghouse's statement of conformance to paragraph 6.2.1.3 of the Standard Review Plan, "Mass and Energy Release Analysis for Postulated LOCAs," that is identified on pages 6-8 and 6-9 of Revision 1 to WCAP-13054, "AP600 Compliance with SRP Acceptance Criteria."

a. Provide the reference for the NRC-approved TMD code Westinghouse intends to use for M&E analyses.

b. For what reason is the TMD code being used rather than a more recognizable licensing code?

c. Document the specific assumptions used while performing this analysis.

d. Identify the experimental data that will be used in this analysis.

Response:

a. The TMD code is used to analyze the pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout containment. The TMD code is documented in WCAP-8077 (Proprietary) and WCAP-8078 (Non-proprietary). The TMD code received NRC approval 18 December 1973 and is documented in a letter from D. B. Vassallo (NRC) to Romano Salvatori (Westinghouse).

b. The TMD code is the NRC licensed code utilized by Westinghouse to perform subcompartment analyses.

c. The analysis specific assumptions are presented in Table 6.2.1.2 and are discussed throughout section 6.2.1.2 in the AP600 SSAR

d. No additional experimental data will be used in the subcompartment analyses of the AP600. The test basis for the TMD code is provided in WCAP-8077.

WCAP-13054 will be modified to delete the reference to TMD.

SSAR Revision: NONE

Question 492.5

Sections 4.3.1.6.2 and 4.4.6 of the SSAR provides a brief description of the fixed in-core detectors.

a. Provide a more detailed description of the fixed in-core detector monitoring system on a functional and operational basis.

b. How does the AP600 **fixed** in-core detecting system differ from the typical PWR **movable** detecting system? Provide technical justification for this design.

c. Is the design of the proposed fixed in-core detecting system approved by the NRC? If so, provide references.

Response:

a. The incore instrumentation system performs two basic functions:

First, it provides a means for monitoring, in an on-line real time mode, nuclear power distribution-related data from the reactor core. This data is used to provide a reliable inference of the actual three-dimensional nuclear power distribution in the core. Certain parameters, obtained by editing the resultant inferred nuclear power distribution, are then used both to calibrate the power range nuclear instrumentation for core protection purposes and to provide guidance to the plant operators to ensure continuing compliance with the plant Technical Specifications and for enhanced fuel management.

Second, it provides in-vessel mechanical support for the core exit thermocouples used in post-accident core cooling monitoring.

The signal output from each fixed incore detector is input to a signal processing device that converts the signals to digitized voltages and transfers them to a multiplexing device where they are put onto a data highway.

The incore instrument system data processor receives the transmitted digitized fixed incore detector signals from the signal processor and combines the measured data with analytically-derived constants, and certain other plant instrumentation sensor signals, to generate a full three-dimensional indication of nuclear power distribution in the reactor core. It also edits the three-dimensional indication of power distribution to extract pertinent power distribution parameters outputs for use by the plant operators and engineers. The data processor also generates hardcopy representations of the detailed three-dimensional nuclear power indications.

b. The AP600 fixed incore system differs from typical moving detector systems in that information pertaining to core power distribution is continuously monitored in the AP600 rather than the conventional method of monthly monitoring by means of incore flux mapping. With a movable detector system, incore power distributions measurements are taken once a month at a reference reactor operating condition. Between the time these measurements are performed, compliance to safety limits is assured by monitoring of a number of global

parameters such as axial flux difference, or quadrant power tilt ratio. With the fixed incore system, continuous monitoring of safety related core parameters such as $F_{\Delta H}$ and Fq is performed on line.

The in-core neutron detectors are fixed but removable and measure the neutron flux at representative locations throughout the core. The use of fixed detector assemblies capable of determining axial power profile eliminates the need for movable in-core detectors and accompanying mechanisms. This simplifies plant equipment and reduces the maintenance effort.

c. The hardware components, specifically rhodium fixed incore detectors, utilized in this system are functionally similar to components which are currently in use in operating plants. A Westinghouse designed incore instrumentation system using fixed incore detectors is not yet approved. Information on the employment of fixed incore detectors in conjunction with an online power distribution monitoring system will be provided to the NRC to support the Final Safety Evaluation Report.

SSAR Revision: NONE

Question 952.82

The February 22, 1994 submittal on the AP600 design changes indicates some piping changes for safety lines, including scheduling changes. Describe any changes that will be implemented in the SPES-2 facility.

Response:

All of the AP600 design changes from the February 22, 1994 report which impacted the design and operation of the SPES-2 facility were implemented for the SPES-2 facility starting with matrix test S00303.

SSAR Revision: None
PRA Revision: None