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NTD-NRC-94-4237
DCP/NRC0163
Docket No.: STN-52-003

July 27, 1994

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

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of April 28, 1994, April 29, 1994, May 2, 1994, May 5, 1994, May 16, 1994, and May 26, 1994. In addition, revisions of responses previously submitted is 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.

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-94-4237
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JULY 27, 1994

RAI No.	Issue
210.059	SSAR sections 3.9.1.1 & 3.9.3.1.2
210.076	SSAR section 3.6.2
210.079	WCAP-13054, SSAR Tables 3.9-5, 3.9-6, 3.9-7, 3.9-8
220.017R01	Dynamic soil bearing capacity
220.027R01	Potential sources of missiles in containment
220.083R01	Design information per SRP format
220.092	Containment structural calculations
230.009R01	ASCE standard 4-86
230.015R02	Soil-specific analyses
230.035R01	Results of 2D SSI & 3D response spectrum analyses
230.058R02	High frequency modes of structures
230.059R01	Comparison between SRSS and 1,.4,.4 method
230.082R01	Method of analysis used to calculate seismic force
410.138	Main turbine interface requirements
410.203	COL flood anaysis
410.209	Protection of safety-related SSCs
435.081	Modeling of battery unavailability in PRA
440.068	Vulnerabilities for shutdown/midloop operation
440.111	Justification for small break LOCA size
440.165	Startup feedwater isolation valves
480.058	Airlock seal testing as reduced pressure
480.061	Chilled water return isolation valve size
480.070	Containment pressure analyses for ECCS performance
720.272	Shutdown PRA - Maintenance Unavailability
720.273	Shutdown PRA - Loss of NRHR initiator
720.274	Shutdown PRA - Overdraining of Reactor Vessel

NTD-NRC-94-4237
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JULY 27, 1994

RAI No.	Issue
720.278	HEP for ATWS

Question 210.59

Sections 3.9.1.1 and 3.9.3.1.2 of the SSAR each contain the same brief discussion that states that the design of piping and component nozzles in the AP600 will minimize the potential for and the effects of thermal stratification and cycling. In Section 3.9.3.1.2, provide a description of the confirmation process to be implemented by the COL licensee to determine whether these effects have been minimized to an acceptable level. If this cannot be verified, describe the analyses and testing required to assure that the design has accounted for these effects, including the method and procedures necessary to define the stratified thermal profile.

Response:

See SSAR revision below for implentation of NRC Bulletins 88-08 and 88-11.

SSAR Revision:

Add the following to the end of Subsection 3.9.3.1.2.

NRC Bulletin 88-08 (Reference 14) requires that continuing assurance of piping integrity is provided for systems that are connected to the reactor coolant system (RCS) and which may experience isolation valve leakage that could result in adverse thermal stresses and fatigue cracks. This assurance may be provided by designing the system to withstand the stresses resulting from valve leakage, instrumenting the piping to detect adverse temperature distributions and establishing appropriate limits on these temperature distributions, or providing means that pressure upstream from isolation valves which might leak into the reactor coolant system is monitored and does not exceed reactor coolant system pressure. In addition to leakage into the reactor coolant system, leakage out of the reactor coolant system through valve packing glands is also considered as described below.

For the susceptible ASME Class 1 piping systems in the AP600, analyses are performed to determine the impact on piping and component integrity. The systems will be analyzed to determine if the piping and components can withstand the stresses resulting from postulated valve leakage. Design transient conditions are also included in this analysis. Valve leakage through two or more normally closed valves which do not have leakoff lines, or backflow through two or more check valves is considered to be insignificant, and therefore such piping systems are not considered as susceptible to adverse stresses from valve leakage. The unisolable portions of systems which are part of the reactor coolant pressure boundary are considered for susceptibility to valve leakage, as described in NRC Bulletin 88-08.

The first step in this analysis is the definition of the isolation valve leakage transient. Leakage flow rate is varied from zero to an upper bound based on valve design in order to identify a "worst case" leakage with respect to stress and fatigue. Methodology to determine stratification interface height and heat transfer of stratified flow, contained in Electric Power Research Institute (EPRI) Report TR-103581 (Reference 15) will be used. It is assumed that the isolation valve will leak continuously at the worst case flow rate during normal 100% power operation throughout the plant life.

Once the isolation valve leakage transient is defined, stress analysis is performed to determine the global effect on the piping resulting from the stratification profile, as well as the local effect resulting from the through wall thermal gradients. These effects are combined to determine the total stress state in the piping. Fatigue analysis is performed to determine the fatigue usage factor resulting from the postulated isolation valve leakage transient and

design transients. Methods from the EPRI report for determining the location of turbulent penetration thermal cycling, the number of leakage cycles and thermal striping fatigue usage are used. If the calculated fatigue usage is greater than 1.0, the system is redesigned or temperature monitoring is implemented with criteria based upon the analysis described above. The following guidelines will be used, if required based on the fatigue usage factor calculation.

Monitoring Guidelines for Thermal Stratification

Type and location of sensors.

- Temperature sensors should preferably be resistance temperature detectors (RTDs).
- RTDs should be located between the first elbow or bend (closest to the reactor coolant system), and the first check valve (check valve closest to the reactor coolant system).
- For the auxiliary pressurizer spray line, RTDs should be installed near the "tee" connection to the main pressurizer spray line on the cold portion (ambient temperature) of the line.
- RTDs should be located within six inches of the welds.
- At each pipe cross section, one RTD should be positioned on the top of the pipe and another RTD on the bottom of the pipe.

Determination of baseline temperature histories.

After RTD installation, temperature should be recorded by the Combined License holder during normal plant operation at every location over a period of 24 hours. The resulting temperature versus time records represent the baseline temperature histories at these locations. Baseline temperature histories should meet the following criteria:

- The maximum top-to-bottom temperature difference should not exceed 50°F.
- Top and bottom temperature time histories should be in-phase.
- Peak-to-peak temperature fluctuations should not exceed 60°F.

Monitoring time intervals.

- Monitoring should be performed at the following times:

  - At the beginning of power operation, after startup from a refueling shutdown

  - At six-month intervals thereafter, between refueling outages

- During each monitoring period, temperature readings should be recorded continuously for a 24-hour period.

Exceedance Criteria.

Actions will be taken to modify piping sections or to correct valve leakage if the following conditions occur:

- The maximum temperature difference between the top and the bottom of the pipe exceeds 50°F.
- Top and bottom temperature histories are in-phase but the peak-to-peak fluctuations of the top or bottom temperatures exceed 60°F.
- Top and bottom temperature histories are out-of -phase and the bottom peak-to-peak temperature fluctuations exceed 50°F.
- Temperature histories do not correspond to the initially recorded baseline histories.

A plant specific analysis of the AP600 surge line is performed to demonstrate that all applicable requirements of the ASME Section III Code 1989 Edition are met for the 60 year life of the plant. This analysis will include consideration of plant operation, thermal stratification and thermal striping, using temperature distributions and transients which are developed from experience on existing plant monitoring programs. A monitoring program will be implemented by the Combined License holder at the first AP600 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters such as pressurizer temperature and level, hot leg temperature, reactor coolant pump status, etc. Monitoring will be performed during hot functional testing and during the first fuel cycle. The resulting monitoring data will be evaluated to show that it is within the bounds of the analytical temperature distributions and displacements.

Add to Subsection 3.9.8:

14. NRC BULLETIN NO.88-08: Thermal Stresses in Piping Connected to Reactor Coolant Systems, June 22, 1988, including Supplements 1, 2, and 3, dated: June 24, 1988; August 4, 1988; and April 11, 1989.

15. Electric Power Research Institute (EPRI) Report TR-103581, "Thermal Stratification, Cycling and Striping (TASCS)", Research Project 3153-02, March 1994.

16. NRC BULLETIN NO. 88-11: Pressurizer Surge Line Thermal Stratification, December 20, 1988.

Question 210.76

Section 3.6.2 of the SSAR does not appear to address the guidelines in Section B.1.c(4) of BTP MEB 3-1 in Section 3.6.2 of the SRP relative to structures that separate high-energy lines from essential components. Revision 1 to WCAP-13054 takes exception to this criteria and states that separating structures are designed for postulated terminal end breaks and high stress locations. This exception is not completely acceptable. The staff's position, as stated in Section 3.6.2 of the SRP, is that such structures should be designed to withstand the consequences of the pipe break on the high-energy line that produces the greatest effect on the structure irrespective of the fact that the pipe break criteria of Section 3.6.2 of the SRP might not require such a break location to be postulated. Revise Section 3.6.2 of the SSAR to add a commitment to this position, and delete the exception to this guideline in WCAP 13054.

Response:

Structures in the main steam and feedwater break exclusion zones are evaluated for subcompartment pressurization effects due to one square foot ruptures in the main steam or main feedwater piping. These are the same ruptures that NRC Branch Technical Position SPLB 3-1, section B.1.a.(1), defines for evaluation of environmental effects. This position is similar to positions previously approved by the staff on operating plants. Structures in the steam generator blowdown break exclusion zone are evaluated for subcompartment pressurization effects due to worst case double ended pipe rupture in the four inch steam generator blowdown piping. Pipe whip and jet impingement are not evaluated for structures in the break exclusion zones per NRC Branch Technical Position MEB 3-1, section B.1.b. In addition to the subcompartment pressurization loads, the wall between the main steam line isolation valve compartment and the main control room is evaluated for jet impingement load from the 1.0 square foot break as a longitudinal break in either the main steam line or the main feedwater line.

Structures not related to the break exclusion zones are designed for subcompartment pressurization effects for the piping that does not satisfy the leak-before break criteria as supplemented by the following. These structures are designed for pipe whip effects for specific terminal end and high stress intermediate break locations in piping that does not satisfy the leak-before-break criteria. Revision 1 of the response to RAI 220.27 identifies the terminal end pipe ruptures that are used for subcompartments inside containment. In order to account for high stress break locations and the additional pressure boundary leakages from manways and flanges, pressurization loads on compartments inside containment enclosing high energy piping, other than the upper reactor cavity, are designed for a 3 inch diameter pipe break in the reactor coolant system. There are no manways or flanges in the upper reactor vessel cavity to cause leakage beyond the 5 gallon per minute leakage crack in the primary coolant loop piping.

Also, see response to RAI 480.68.

WCAP 13054 will not be revised.

SSAR Revisions:

Revise the last three paragraphs of Subsection 3.6.1 as follows:

The pressurization loads on structures and components are evaluated for circumferential breaks and longitudinal breaks in piping that does not meet leak-before-break requirements and for leakage cracks in piping that meets the leak-before-break requirements. In addition, structures inside containment containing high energy piping are evaluated for the pressurization loads due to a break area equivalent to a three-inch (nominal) diameter primary system pipe.

The in-containment refueling water storage tank ~~and the reactor vessel annulus, which do not include any pipes less than three-inch diameter subject to failure, are~~ is evaluated for pressurization with different criteria. The evaluation ~~of these areas for pressurization~~ is described ~~later~~ in Subsection 3.6.1.2.1.

Pressurization loads for pipe failures in the main steam and feedwater break exclusion zones for high-energy lines in the vicinity of containment penetrations are evaluated for a 1.0 square foot break. Structures in the steam generator blowdown break exclusion zone are evaluated for subcompartment pressurization effects due to worst case double ended pipe rupture in the four inch steam generator blowdown piping. Pipe whip and jet impingement are not evaluated for structures in the break exclusion zones per NRC Branch Technical Position MEB 3-1, section B.1.b, except that the wall between the main steam line isolation valve compartment and the main control room is evaluated for jet impingement from the 1.0 square foot break as a longitudinal break in either the main steam line or the main feedwater line. See Subsection 3.6.2.1.1.4.

Revise Subsection 3.6.1.2.1 as follows:

3.6.1.2.1 Pressurization Response

Pressure response analyses are performed for subcompartments containing high-energy piping for which break locations are defined by Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3 or postulated leakage flaws are defined based on Subsection 3.6.3.3. ~~In subcompartments inside containment containing no lines greater than three-inch nominal diameter subject to pipe rupture, based on mechanistic pipe break requirements, the subcompartment pressurization analysis and evaluation of venting provisions are based on a break area equivalent to a three-inch diameter primary system pipe break. The three-inch break is applied to subcompartments inside containment, except for the in-containment refueling water storage tank and the reactor vessel annulus.~~ In order to account for high stress break locations and the additional pressure boundary leakages from manways and flanges, pressurization loads on compartments inside containment, other than the upper reactor cavity, enclosing high energy piping are designed for a 3 inch diameter pipe break in the reactor coolant system. There are no manways or flanges in the upper reactor vessel cavity (reactor vessel annulus) to cause leakage beyond the 5 gallon per minute leakage crack in the primary coolant loop piping.

The pressurization loads for the in-containment refueling water storage tank are based on the pressure and hydrodynamic loads due to the maximum discharge through the first, second, and third stages of the automatic depressurization system valves.

The pressurization loads for the reactor vessel annulus for the evaluation of asymmetric compartment pressurization are based on a five-gallon per minute leakage crack in the primary loop piping. The internal reactor pressure vessel asymmetric pressurization loads are based on a break in the largest pipe connected to the reactor coolant system that does not qualify for the application of mechanistic pipe break

The pressurization loads for the steam generator blowdown break exclusion zone are based on a double ended rupture of the 4 inch blowdown piping.

For a detailed discussion of the criteria and analysis methods for subcompartment pressurization analysis, see Subsection 6.2.1.2. The analytical methods for transient mass distribution, used for pressure response analysis, are described in WCAP-8077 (Reference 2).

Revise the third paragraph of Subsection 3.6.1.2.2 as follows:

Consistent with the criteria for evaluation of leaks in the break exclusion area, the subcompartment, including the walls, is evaluated for the effects of flooding, spray wetting and subcompartment pressurization from a 1.0 square foot break from either main steam or feedwater line within the respective break exclusion areas. The wall ~~closest to the break~~ between the main steam line isolation valve compartment and the main control room and the floor slab between the main steam line isolation valve compartment and the safety related electrical equipment room are ~~is~~ also evaluated for jet impingement from the 1.0 square foot break as a longitudinal break in either the main steam line or the main feedwater line.

Revise the second paragraph of Subsection 3.6.2.1.1.4 as follows:

Areas of system piping where no breaks, except as noted in subsections 3.6.1.2 , 3.6.1.2.1, and 3.6.1.2.2, are postulated are as follows:

- The main steam piping, from the containment penetration flued head outboard weld, to the upstream weld of the auxiliary building anchor downstream of the main steam isolation valves, including the main steam safety valves and the connecting branch piping

- The main feedwater piping from the containment penetration to the auxiliary building anchor upstream of the isolation valve, including branch connections

- The steam generator blowdown piping from the containment to auxiliary building anchor downstream of the isolation valve

Question 210.79

Revision 1 to WCAP-13054 lists an exception to Section C.1.3.4(a) of Appendix A to Section 3.9.3 of the SRP, that states that SSE loads are not combined with "non-LOCA" pipe ruptures. The staff does not agree with this exception. Design basis pipe breaks (DBPB) as defined in Section 3.9.3 of the SRP are non-LOCA loads and should be combined with SSE loads and designed to Service Level D limits. Revise WCAP-13054 to delete this exception and revise Tables 3.9-5, 3.9-6, 3.9-7, and 3.9-8 of the SSAR to include Sustained Loads + DBPB + SSE under "Level D Service."

Response:

The Service Level D load combinations for ASME piping in SSAR Tables 3.9-6 and 3.9-7 are being revised to include SSE plus pipe rupture, where pipe rupture is either a LOCA or non-LOCA event. These Tables are also revised to incorporate new criteria for reversing and nonreversing dynamic loads.

Tables 3.9-3 and 3.9-4 are revised to reflect elimination of the piping loads which are shown in Table 3.9-16.

The Service Level D load combination for components, supports and piping supports in SSAR Tables 3.9-5 and 3.9-8 are revised to include SSE plus pipe rupture, where pipe rupture is either a LOCA or non-LOCA event.

The load combinations and stress limits are revised based on Attachment for RAI 210.60 "Staff Position on the Use of a Single-earthquake Design for Systems, Structures, and Components in the AP600 Standard Plant"

The current staff position on functionality of piping systems as documented in NUREG-1367, "Functional Capability of Piping Systems," dated November 1992, will be incorporated in SSAR Table 3.9-11.

Section 3.7 is revised to incorporate 5 percent damping in lieu of Code Case N411 damping values.

See response to RAI 210.68 for revisions to Tables 3.9-9 and 3.9-10.

The following SSAR revisions include revisions due to responses to RAIs 210.60, 210.62, 210.65, 210.68 and 210.80.

WCAP 13054 will be revised to delete exception to Section C.1.3.4(a) of Appendix A to Section 3.9.3 of the SRP. In response to RAI 210.62, the exception to Section C.1.2 will be revised in the next revision of WCAP 13054 to reflect the SRSS combination of appropriate dynamic loads.

SSAR Revision:

Revise the first item after the third paragraph of Subsection 3.9.3.1.1 as follows:

- Structures, ASME Code ~~Class 1~~ components, and supports for these components are designed for the safe shutdown earthquake combined by the square root of the sum of the squares method, with short-term dynamic

loads due to postulated pipe ruptures. The pipe ruptures included in this combination are those not excluded by application of mechanistic pipe rupture criteria. (See Subsection 3.6.3.) This combination is used for structures, components, and supports that are required to mitigate the effects of the postulated pipe rupture.

Revise the first paragraph of Subsection 3.9.3.1.2 as follows:

The loads used in the analysis of the Class 1 components(**excluding piping**), core supports, and component supports are described ~~in detail~~ in the following paragraphs. The loads are listed in Tables 3.9-3 and 3.9-4.

Revise Subsection 3.9.3.1.3 as follows:

3.9.3.1.3 ASME Code Class 1 Components and Supports and Class CS Core Support Loading Combinations and Stress Limits

Tables 3.9-5 and 3.9-8 list loading combinations for ASME Class 1 components(**excluding piping**) and component supports(**including piping**) and Class CS core support structures. Table 3.9-9 lists the stress limits for these components. Table 3.9-3 ~~and 3.9-4~~ lists the loads included in the loading combinations.

The stress limits for Service Level D that allow inelastic deformation are supplemented with the requirements of "Rules for Evaluation of Service Loadings with Level D Service Limits," Appendix F of ASME Code, Section III. The limits and rules of Appendix F confirm that pressure boundary integrity and core support structural integrity are maintained but do not confirm operability. The limits and rules of Appendix F do not apply to the portion of the component or support in which the failure has been postulated. Subsection 3.9.1 provides a ~~detailed~~ discussion of design transients used in the analysis of cyclic fatigue.

The structural stress analyses performed on the ASME Code Class 1 components(**excluding piping**) and supports(**including piping**) and Class CS core support structures consider the loadings specified, as shown in Table**s** 3.9-3 ~~and 3.9-4~~. These loads result from thermal expansion, pressure, weight, earthquake, pipe rupture, and plant operational thermal and pressure transients. Dynamic effects of pipe rupture, including the loss of coolant accident, are not included in loading combinations when the leak-before-break criteria are satisfied. The methods and acceptance standard for leak-before-break analyses are described in Subsection 3.6.3.

Although the system is designed for seismic loads, the combination of safe shutdown earthquake plus pipe rupture (those breaks not excluded by mechanistic pipe break criteria) loads by square-root-sum-of the squares is considered. This loading combination is evaluated for ASME Code ~~Class 1~~ components **and piping** and the supports for those components ~~and the cold leg and hot leg primary loop piping~~.

Revise the first five paragraph**s** of Subsection 3.9.3.1.5 as follows:

The loads for ASME Code Classes 1, 2, and 3 piping are listed in Table**s** 3.9-16~~3 and 3.9-4~~. Tables 3.9-6 and 3.9-7 list**s** the loading combinations **and stress limits**. ~~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 and limit stop supports with gaps 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. Alternatively, the time history method may be used for the solution of the seismic disturbance.

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 Tables 3.9-~~6~~~~11~~ and 3.9-7.

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.

The functional capability requirements for ASME piping systems that must maintain an adequate fluid flow path to mitigate a Level C or Level D plant event are shown in Table 3.9-11.

Revise Tables 3.9-3, 3.9-4, 3.9-5, 3.9-6, 3.9-7, 3.9-8, 3.9-9, 3.9-11, and 3.9-16 as follows:

Table 3.9-3

Loadings for ASME Class 1, 2, 3, CS and Supports ~~Primary Stress Producing Loads~~ (Excluding piping)

Load	Description
P	Internal design pressure
PMAX	Peak pressure
DW	Dead weight
XL	Other specified external loads associated with the various service conditions (e.g., piping loads on nozzles)
SSE	Safe shutdown earthquake (inertia portion)
E	Earthquake smaller than SSE (inertia portion) ~~This load is evaluated for continued operation; not to verify performance of a safety function.~~
FV	Fast valve closure
RVC	Relief/safety valve - closed system (transient)
RVOS	Relief/safety valve - open system (sustained)
RVOT	Relief/safety valve - open system (transient)
~~SRV~~	~~Dynamic effects due to safety relief valve discharge~~
DU	Other transient dynamic event associated with Level B (Upset) service conditions
DE	Transient dynamic event associated with Level C (Emergency) service conditions
DF	Transient dynamic event associated with Level D (Faulted) service conditions during which, or following which, the piping system being evaluated must remain intact. This includes postulated pipe rupture events.
SSES	Seismic anchor motion portion of SSE

ES	Seismic anchor motion of earthquake smaller than SSE
TH	Thermal loads for the various service conditions
TNU	Service Level A and B (Normal and Upset) plant condition thermal loads; including thermal stratification and thermal cycling.
TN	Service Level A (Normal) plant condition thermal loads
TU	Service Level B (Upset) plant condition thermal loads
TE	Service Level C (Emergency) plant condition thermal loads
TF	Service Level D (Faulted) plant condition thermal loads
HTDW	Hydrostatic test dead weight
DBPB	Design basis pipe break, includes LOCA and non-LOCA
LOCA	Loss of coolant accident

AP600

Table 3.9-4

(Deleted)

~~Additional Loadings for ASME Class 1, 2, 3, CS and Supports~~

~~These loads are treated as loads producing secondary stress in the analysis of piping and as loads producing primary stress in the analysis of components and supports.~~

~~Load~~	~~Description~~
~~SSES~~	~~Seismic anchor motion por[illegible] of SSE~~
~~ES~~	~~Seismic anchor motion of earthquake smaller than SSE~~
~~TH~~	~~Thermal loads for the various service conditions~~
~~TNU~~	~~Service Level A and B (Normal and Upset) plant condition thermal loads; including thermal stratification and thermal cycling~~
~~TN~~	~~Service Level A (Normal) plant condition thermal loads~~
~~TU~~	~~Service Level B (Upset) plant condition thermal loads~~
~~TE~~	~~Service Level C (Emergency) plant condition thermal loads~~
~~TF~~	~~Service Level D (Faulted) plant condition thermal loads~~

~~Additional Loadings for ASME Class 1, 2, 3 Supports~~

~~Load~~	~~Description~~
~~HTDW~~	~~Hydrostatic test dead weight~~
~~SWE~~	~~Self weight excitation~~ ~~(Effect of the acceleration of the support mass on the support)~~

Table 3.9-5

Minimum Design Loading Combinations for ASME Class 1, 2, 3(excluding piping) and CS Systems and Components

Condition	Design Loading Combinations[6]
Design	P + DW + XL
Level A Service	PMAX[1] + DW + XL
Level B Service	PMAX + DW + RVC + XL
	PMAX + DW + RVOS[7] + XL
	PMAX + DW + RVOT[7] + XL
	~~PMAX + DW + SRV[7] + XL~~
	PMAX + DW + DU + XL
	PMAX + DW + FV + XL
	PMAX + DW + ~~SRSS (~~E + ES~~)[2]~~[8] + XL[9]
Level C Service	PMAX + DW + DE + XL
	PMAX + DW + ~~SRSS (~~SSE + SSES~~)~~[8] + XL[10]
Level D Service[6]	PMAX + DW + ~~SRSS (~~SSE + SSES~~)~~ + XL
	PMAX + DW + SRSS (SSE + SSES + RVC)[2] + XL
	~~PMAX + DW + SRSS (SSE + SSES + SRV) + XL~~
	PMAX + DW + DF + XL
	PMAX + DW + SRSS (SSE + SSES + DBPB ~~DF~~)[2][5] + XL
	PMAX + DW + RVOS + ~~SRSS (~~SSE + SSES~~)~~ + XL
	PMAX + DW + SRSS (SSE + SSES + RVOT)[2] + XL

See following page for notes.

Notes for Table 3.9-5

1. The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100% power.

2. SRSS equals the square root of the sum of the squares.

3. Design mechanical loads, such as the nozzle reactions associated with thermal expansion of piping systems, shall be combined with other loads in the loading combination expressions.

4. Appropriate loads due to static displacements of the steel containment vessel and building settlement should be added to the loading combinations expressions for ASME Code, Section III, Class 2 and 3 systems.

5. ~~ASME Code, Section III, Class 1 and Class CS components are designed to the Level D service limits for this loading combination where DF represents a loss of coolant accident pipe rupture.~~
DBPB is design basis pipe break and includes LOCA and non-LOCA events

6. In combining earthquake loads and consequent plant transients, the timing and causal relationships that exist between DU, DE, DF, SSE, RVOS, RVOT, RVC, FV, and ~~SRV~~ XL are considered for determination of the appropriate load combinations.

7. The pressurizer safety valve discharge is a Level C service condition.

8. Either of these two loading combinations maybe used.

9. For components that behave as anchors to the piping system, such as equipment nozzles, E and ES are combined by absolute sum. For other components, such as valves, E and ES are combined by SRSS method.

10. For components that behave as anchors to the piping system, such as equipment nozzles, SSE and SSES are combined by absolute sum. For other components, such as valves, SSE and SSES are combined by SRSS method.

Table 3.9-6

~~Minimum Design Loading Combinations for ASME Class 1 Piping~~

~~Condition~~	~~Design Loading Combinations for Primary Stress~~	~~Other Loadings[6]~~
~~Design~~	~~P + DW~~	~~—~~
~~Level A Service~~	~~—~~	~~P, RVC, RVOS, RVOT, SRV, FV, TN~~
~~Level B Service~~	~~PMAX[1] + DW + RVC~~ ~~PMAX + DW + RVOS~~ ~~PMAX + DW + SRV[2]~~ ~~PMAX + DW + DU~~ ~~PMAX + DW + FV~~ ~~PMAX + DW + RVOT~~	~~P, RVC, RVOS, RVOT, SRV, FV~~ ~~DU, TU, E, ES[3]~~
~~Level C Service~~	~~PMAX + DW + DE~~	~~TE[3]~~
~~Level D Service[4]~~	~~PMAX + DW + SRSS[5] (SSE + RVC)~~ ~~PMAX + DW + DF~~ ~~PMAX + DW + SRSS (SSE + SRV)~~ ~~PMAX + DW + SSE + RVOS~~ ~~PMAX + DW + SSE~~ ~~PMAX + DW + SRSS (SSE + RVOT)~~ ~~PMAX + DW + SSRS (SSE + DF)[7]~~	~~TF[3]~~ ~~SSES[3]~~

~~Notes:~~

1. ~~The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100% power.~~
2. ~~Pressurizer safety valve discharge is classified as a level C event.~~
3. ~~See Table 3.9-9 for stress criteria~~
4. ~~The timing and causal relationships between safe shutdown earthquake and RVOS, RVOT, RVC, and SRV is considered.~~
5. ~~SRSS - square root of the sum of the squares~~
6. ~~Other loadings are used in analyses that include secondary stresses which include analysis of cyclic fatigue.~~
7. ~~This load combination is for primary loop piping only.~~

AP600

~~Table 3.9-7~~

~~Minimum Design Loading Combinations for ASME Class 2 and 3 Piping~~

~~Condition~~	~~Design Loading Combinations for Primary Stress~~	~~Other Loadings[6]~~
~~Design~~	~~P + DW~~	
~~Level A Service or Level B Service~~	~~PMAX[1] + DW + RVC~~	~~PMAX + DW + TNU[2]~~
	~~PMAX + DW + RVOS~~	
	~~PMAX + DW + SRV~~	
	~~PMAX + DW + DU~~	
	~~PMAX + DW + FV~~	
	~~PMAX + DW + RVOT~~	
~~Level C Service~~	~~PMAX + DW + DE~~	~~TE[3]~~
~~Level D Service[4]~~	~~PMAX + DW + SRSS[5](SSE + RVC)~~	~~TF[3]~~
	~~PMAX + DW + DF~~	~~SSES[3]~~
	~~PMAX + DW + SRSS(SSE + SRV)~~	
	~~PMAX + DW + SSE~~	
	~~PMAX + DW + SSE + RVOS~~	
	~~PMAX + DW + SRSS(SSE + RVOT)~~	

~~Notes:~~

1. ~~The values of PMAX in the load combinations may be different for different levels of service conditions as provided in the design transients. For earthquake loadings PMAX is equal to normal operating pressure at 100% power.~~
2. ~~Appropriate loads due to static displacements of the steel containment vessel and building settlement should be added to the loading combinations expressions for Class 2 and 3 systems.~~
3. ~~See Table 3.9-9 for stress criteria~~
4. ~~In combining earthquake loads and consequential plant transients, the timing of the loads is appropriately considered.~~
5. ~~SRSS - square root of the sum of the squares~~
6. ~~Other loadings are used in analyses which include secondary stresses.~~

Table 3.9-8

Minimum Design Loading Combinations for Supports for ASME Class 1, 2, 3 Piping and Components(2)

Condition	Design Loading Combinations
Design	DW
Level A Service	DW + TH
Level B Service	DW + TH + ~~SRSS(RVC + SWE)~~ RVC DW + TH + ~~SRSS(RVOT + SWE)~~ RVOT(5) DW + TH + ~~SRSS(DU + SWE)~~ DU ~~DW + TH + SRSS(SRV + SWE)(6)~~ DW + TH + RVOS(5) DW + FV
Level C Service	DW + TH + DE ~~SRSS(1) (DE + SWE)~~
Level D Service(54)	DW + TH + ~~SRSS (~~SSE + SSES + SWE~~)~~ (6) DW + TH + SRSS(1) (~~SSE + SSES +~~ RVC + (SSE + SSES + SWE))(6) DW + TH + SRSS (~~SSE + SSES +~~ RVOT + (SSE + SSES + SWE))(6) DW + TH + RVOS + ~~SRSS (~~SSE + SSES + SWE~~)~~(6) DW + TH + ~~SRSS (~~DF ~~+ SWE)(4)~~ ~~DW + TH + SRSS (SSE + SSES + SRV)~~ DW + TH + SRSS (~~SSE + SSES + DF)(3)~~ DBPB + (SSE + SSES + SWE))(3)(6)
Hydrostatic Test	HTDW

Notes:

1. SRSS - square root of the sum of the squares
2. Appropriate loads due to static displacement of the steel containment vessel and building settlement should be added to the loading combinations expressions for Class 2 and 3 systems.
3. ~~Class 1 component supports are designed for this condition where DF represents a limiting pipe rupture.~~ DBPB is design basis pipe break and includes LOCA and non-LOCA events
4. ~~For piping supports, an acceptable alternative is to permit support failure and evaluate the consequences on piping system integrity and operability.~~

54. In combining earthquake loads and consequential plant transients, the timing of the loads is appropriately considered.

65. The pressurizer safety valve discharge is a Level C Service condition.

6. Combine SSE, SSES, and SWE by absolute sum method. SWE is self weight excitation, the effect of the acceleration of the support mass caused by building filtered loads such as SSE)

Table 3.9-6

Load Combinations and Stress Limits for ASME Class 1 Piping

Condition	Loads(13)	Equation (NB3650)	Stress Limit
Design	P + DW + DML	9	1.5 S_m
Level B	PMAX(1) + DW + DU	9	1.8 S_m, 1.5 S_y
	PMAX, TNU, E, ES, RVC, RVOS(2), DU, FV, RVOT(2)(9)(20)	10 11, 14	3.0 S_m CUF = 1.0
	TNU	12	3.0 S_m
	PMAX + DW + DU PMAX + DW + RVOS	13 13	3.0 S_m 3.0 S_m
	DW + TNU + RVOS(16)	15	3125 CA_b S_y/36(14)
	DW + TNU + DU(16)	16	6250 CA_b S_y/36(14)
Level C	PMAX + DW + DENR	9	2.25 S_m, 1.8 S_y
	PMAX + DW + DER	9	3.15 S_m (4),(3),(18)
	TE + SCVE	Note (3)	Note (3)
	PMAX + DW + TF + SCVE + DE(16)	17	FLC(15)

Table 3.9-6 (Continued)

Load Combinations and Stress Limits for ASME Class 1 Piping

Condition	Loads(13)	Equation (NB3650)	Stress Limit
Level D(8)	PMAX + DW + DFNR	9	$3.0\ S_m$, $2.0\ S_y$
	PMAX + DW + SRSS(9) (SSE + NRDL)(6)	9	$4.5\ S_m$ (4),(3),(19),(7)
	PMAX + DW + RVOS + SSE(6)	9	$3.0\ S_m$, $2.0\ S_y$
	PMAX + DW + SRSS(9) (SSE(12) + DBPBNR)	9	$4.5\ S_m$ (4),(3),(19),(7)
	PMAX(5) + DW + DFR	9	$4.5\ S_m$ (4),(3),(19)
	PMAX(5) + DW + SRSS (SSE(12) + DBPBR)	9	$4.5\ S_m$ (4),(3),(19)
	PMAX(5) + DW + SRSS (SSE(12) + RDL)(6)	9	$4.5\ S_m$ (3),(4),(19)
	SSES	$C_2 D_o M_{RAM}/2I$ (10)	$6.0\ S_m$
	SSES	F_{AM}/A_M (11)	$1.0\ S_m$
	TF + SCVF	Note (3)	Note (3)
	TNU + SSES	$C_2 D_0 (M1 + M2)/2I$ (17)	$6.0\ S_m$

Table 3.9-6 (Continued)

Load Combinations and Stress Limits for ASME Class 1 Piping

Condition	Loads[13]	Equation (NB3650)	Stress Limit
Level D[8]	PMAX + DW + TF + SCVF + DF[16]	17	FLC[15]
	PMAX + DW + TF + SCVF + RVOS + SSE[6][16]	17	FLC
	PMAX + DW + TF + SCVF + SRSS (SSE + DL)[6][16]	17	FLC
	PMAX + DW + TF + SCVF + SRSS (SSE + DBPB)[16]	17	FLC

Notes to Table 3.9-6

(1) The values of PMAX in the load combinations may be different for different levels of service conditions. For earthquake loading, PMAX is equal to normal operating pressure at 100% power.

(2) Pressurizer safety valve discharge is classified as a Level C event.

(3) See Table 3.9-11 for functional capability requirements.

(4) Sustained stress due to deadweight is limited by $B_2DoM_{DW}/2I \leq 0.5\ S_m$, where B_2, Do, I, and S_m are per ASME III; if sustained stress limit is not met, replace 4.5 S_m with 3.0 S_m, 2.0 S_y, or replace 3.15 S_m with 2.25 S_m, 1.8 S_y, as applicable. Materials are selected from Table 2A, ASME II, Part D, P-Nos. 1 through 9.

(5) PMAX shall not exceed 1.1 times the Design Pressure.

(6) The timing and causal relationships between SSE and RVOS, NRDL, and RDL are considered to determine if a load combination is required.

(7) Sustained stress due to algebraic sum of deadweight and DBPBNR, or deadweight and NRDL is limited by $B_2DoM_{DW}/2I \leq 1.0\ S_m$; if sustained stress limit is not met, replace 4.5 S_m with 3.0 S_m, 2.0 S_y

(8) Alternatively, the limits of Appendix F of ASME III may be used.

(9) Square root sum of the squares (SRSS) combination is used for ES, E, and other transient loads.

(10) C_2, Do, I based on ASME III. M_{RAM} is range of resultant moment for SSES.

(11) F_{AM} is amplitude of axial force for SSES; A_M is nominal pipe metal area.

(12) SSE stresses are based on linear elastic response spectra analysis with ±15% broadening, 5% damping, and 0.3g peak ground acceleration when the 4.5 S_m limit is used.

(13) See Table 3.9-16 for description of loads.

(14) C, A_b, S_y based on ASME III.

(15) $FLC = [11250\ A_b - (\pi/16)^2\ D_f^2\ P_{fd}]\ C\ S_y/36$, where A_b, D_f, P_{fd}, C, S_y are based on ASME III.

(16) The timing and causal relationships among TNU, RVOS, RVC,, DL, DU, DE, DF, FV, RVOT, TE, TF, SCVE, and SCVF are considered to determine appropriate load combinations.

Notes to Table 3.9-6 (Continued)

(17) Where: M1 is range of moments for TNU, M2 is one half the range of SSES moments.
M1 + M2 is larger of M1 plus one half the range of SSES, or full range of SSES

(18) Alternatively, inelastic dynamic analysis may be performed with 70% of the limits in Note 19.

(19) Alternatively, inelastic dynamic analysis, including incremental ratcheting, may be performed with the following limits:

(a) effective ratchet strain averaged through the wall thickness due to simultaneous effects of pressure, gravity, thermal expansion ranges, anchor motion ranges, and reversing dynamic loading ranges shall not exceed 5%.

(b) effective local peak cyclic single-amplitude strain, e_{AN}, due to the application of the applied loading ranges considered in (a) above shall not exceed:

$$e_{AN} \leq S_{a10} / (EN^{1/2})$$

where:

S_{a10} = ASME Code, Appendix I, value from the applicable fatigue curve at 10 cycles

E = Young's Modulus

N = Number of cycles of reversing dynamic, not less than 10

(20) The earthquake loads are assumed to occur at normal 100 percent power operation for the purposes of determining the total moment ranges.

Table 3.9-7

Load Combinations and Stress Limits for ASME Class 2, 3 Piping

Condition	Loads(3)	Equation (NC/ND3650)	Stress Limit
Design	P + DW + DML	8	$1.5\ S_h$
Level A/B	PMAX(1) + DW + DU	9	$1.8\ S_h$, $1.5\ S_y$
	PMAX + DW + TNU + SCVNU(4)	11	$S_h + S_A$
	Building Settlement	10a	$3.0\ S_C$
	DW + TNU + RVOS(4)	12	$3125\ CA_b\ S_y/36000$(5)
	DW + TNU + DU(4)	13	$6250\ CA_b\ S_y/36000$(5)
Level C	Same as Class 1 - Table 3.9-6, with S_m replaced by S_h		
Level D	TNU + SSES	$i\ (M1 + M2)/Z$(2)	$3.0\ S_h$
	Same as Class 1 - Table 3.9-6, with S_m replaced by S_h		

Notes for Table 3.9-7

(1) The values of PMAX in the load combinations may be different for different levels of service conditions. For earthquake loading PMAX is equal to normal operating pressure at 100% power.

(2) Where: M1 is range of moments for TNU, M2 is one half the range of SSES moments,
M1 + M2 is larger of M1 plus one half the range of SSES, or full range of SSES

(3) See Table 3.9-16 for description of loads.

(4) The timing and causal relationships among TNU, RVOS, DU, and SCVNU are considered to determine appropriate load combinations.

(5) C, A_b, S_y based on ASME III.

~~Table 3.9-11~~

~~Interim Stress Criteria for ASME Class 1, 2, and 3 Piping Loads (Secondary Stress Producing Loads)~~

~~Loading~~	~~Stress Criteria~~
~~TE, TP~~	~~Equation 10a, ASME III, NC 3653.2~~[1] ~~(Allowable is 3.0 S_c)~~
~~ES,SSES~~	~~When Equation 10 cannot be satisfied, including ES loading, the following is met for Class 1 piping:~~ ~~S(sam) = C2*DO/2/I*(M1 + M2) less than or equal to 6.0 Sm~~ ~~Where:~~ ~~M1 = range of thermal moments per Equation 12 (TN,TU)~~ ~~M1 + M2 = larger of M1 plus one half the range of SSE seismic anchor motion moments (SSES) , or full range of SSE seismic anchor motion moments~~ ~~For Class 2 and Class 3 piping:~~ ~~P*DO/t*0.25 + 0.75*i*MA/Z + i/Z* (MC + M2) less than or equal to SA + 4.0 Sh~~ ~~where:~~ ~~MA, MC, SA, Sh, P, DO, t, i, and Z per Equation 11~~ ~~MC +M2 = larger of MC plus one half the range of SSE seismic anchor motion moments (SSES) , or full range of SSE seismic anchor motion moments~~

~~Notes:~~

1. ~~Applicable to Level C or Level D plant events for which the piping system must maintain an adequate fluid flow path.~~

Table 3.9-16

Loadings for ASME Class 1, 2, and 3 Piping

Load	Description
P	Internal Design Pressure
DW	Deadweight
DML	Design Mechanical Loads (other than DW) This includes RVOS loads that are Service Level A or B.
PMAX	Peak Pressure
DU	Dynamic Transient event associated with Level B (upset) service condition including RVC, RVOT, and FV.
RVC	Relief/safety Valve - Closed System - Transient
RVOS	Relief/safety Valve - Open System - Sustained Load
FV	Fast Valve Closure
RVOT	Relief/safety Valve - Open System - Transient
TNU	Service Level A and B (Normal and Upset, plant condition thermal loads; including thermal stratification and thermal cycling
TF	Plant Faulted Condition Thermal Loads Service Level D
E	Earthquake smaller than SSE - inertia portion - equal to 1/3 times SSE
SSE	Safe shutdown Earthquake - inertia portion
ES	Earthquake smaller than SSE - anchor motion - equal to 1/3 times SSES
TE	Plant Emergency Condition Thermal Loads Service Level C
SSES	Safe shutdown earthquake - anchor motion

Table 3.9-16 (Continued)

Loadings for ASME Class 1, 2, and 3 Piping

Load	Description
DE	Dynamic Transient Event associated with Level C (Emergency) service condition including RVC, RVOT, and FV.
DENR	Dynamic Transient Event associated with Level C (Emergency) service condition - Non reversing dynamic load or reversing dynamic load in combination with nonreversing dynamic load
DER	Dynamic Transient Event associated with Level C (Emergency) Service condition - Reversing dynamic load
DF	Dynamic Transient Event associated with Level D (Faulted) service condition including RVC, RVOT, and FV.
DFNR	Dynamic Transient Event associated with Level D (Faulted) service condition - Nonreversing dynamic load or reversing dynamic load in combination with nonreversing dynamic load
DFR	Dynamic Transient Event associated with Level D (Faulted) service condition - Reversing dynamic load
DL	Dynamic Load
NRDL	Nonreversing Dynamic Loads are those loads which do not cycle about a mean value and include the initial thrust force due to sudden opening or closure of valves and waterhammer resulting from entrapped water in two phase flow systems. Examples are RVOT, and RVC.
DBPB	Design Basis Pipe Break
DBPBNR	Design Basis Pipe Break - Nonreversing Dynamic load or reversing dynamic loads in combination with nonreversing dynamic load, includes jet loads and thrust loads at the break location
DBPBR	Design Basis Pipe Break - Reversing dynamic load, includes transient motions at terminal ends.

Table 3.9-16 (Continued)

Loadings for ASME Class 1, 2, and 3 Piping

Load	Description
RDL	Reversing Dynamic Loads are those loads which cycle about a mean value and include building filtered loads, earthquake, and reflected waves in a piping system due to flow transients from sudden opening or closure of valves. A reversing load shall be treated as nonreversing when either of the following are met: (a) frequency ratio of the dynamic dominant load driving frequency to the lowest piping system natural frequency is less than 0.5 (b) number of reversing dynamic load cycles, exclusive of earthquake, exceeds 20 Examples are RVOT, and RVC.
SCVNU	Static displacement of steel containment vessel - normal and upset conditions
SCVE	Static displacement of steel containment vessel - emergency condition
SCVF	Static displacement of steel containment vessel - faulted condition

Table 3.9-11

Piping Functional Capability - ASME Class 1, 2, and 3(2)

Wall thickness:	$D_o/t \le 50$, where Do, t are per ASME III
External Pressure:	$P_{external} \le P_{internal}$
Steady-State Stress:	$B_2 D_o M/2I \le 0.25\ S_y$, where B_2, D_o, I, S_y per ASME III M = Steady state resultant moment
PMAX + DW + DER	Equation 9 $\le 3.0\ S_m$, $2.0\ S_y$(1)
TE + SCVE	Equation 10a (NC3653.2) $\le 3.0\ S_c$
PMAX + DW + DFR	Equation 9 $\le 3.0\ S_m$, $2.0\ S_y$(1)
PMAX + DW + SRSS (SSE + DBPBR)(3)	Equation 9 $\le 3.0\ S_m$, $2.0\ S_y$(1)
PMAX + DW + SRSS (SSE + DBPBNR)(3)	Equation 9 $\le 3.0\ S_m$, $2.0\ S_y$(1)
PMAX + DW + SRSS (SSE + RDL)(3)	Equation 9 $\le 3.0\ S_m$, $2.0\ S_y$(1)
PMAX + DW + SRSS (SSE + NRDL)(3)	Equation 9 $\le 3.0\ S_m$, $2.0\ S_y$(1)
TF + SCVF	Equation 10a (NC 3653.2) $\le 3.0\ S_c$

Notes for Table 3.9-11

(1) In addition, the response stress for frequencies $\leq$ 2 cps must not be greater than 1.0 S_y.

(2) Applicable to Level C or Level D plant events for which the piping system must maintain an adequate fluid flow path.

(3) SRSS - square root of the sum of the squares

Revise the first paragraph of Subsection 3.7.1.3 as follows:

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The safe shutdown earthquake damping values used in the dynamic analysis are presented in Table 3.7.1-1. The damping values are based on Regulatory Guide 1.61, ASCE Standard 4-86 (Reference 3), and ~~ASME Code Case N-411 (Reference 4)~~ **5 percent damping for piping**, except for the damping values of the primary coolant loop piping, which is based on Reference 22, and conduits, cable trays and their related supports.

Revise Subsection 3.7.3.15 as follows:

3.7.3.15 Analysis Procedure for Damping

Damping values used in the seismic analyses of subsystems are presented in Subsection 3.7.1.3. For subsystems that are composed of different material types, the composite modal damping approach with either the weighted mass or stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems. Piping systems **analyzed by the response spectra method**, including coupled equipment, and valves, can be evaluated with ~~Code Case N-411~~ **5 percent** damping. **When piping systems and non-simple module steel frames (subsection 3.7.3.8.3) are in a single coupled model, composite damping, as described in subsection 3.7.1.3 is used.**

Revise the first paragraph of Subsection 3.7.3.2 as follows:

Seismic Category I structures, systems, and components are evaluated for one occurrence of the safe shutdown earthquake (SSE). In addition, subsystems sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. Using analysis methods, these effects are considered by inclusion of seismic events with an amplitude not less than one-third of the SSE amplitude. The number of cycles is calculated based on IEEE-344-1987 (Reference 21) to provide the equivalent fatigue damage of two SSE events with 10 high-stress cycles per event. Typically, there are five seismic events with an amplitude equal to one-third of the SSE response. Each event has 63 high-stress cycles. **For ASME Class 1 piping, the fatigue evaluation is performed based on five seismic events with an amplitude equal to one-third of the SSE response. Each event has 63 high-stress cycles.**

Revise the description of independent support motion in Subsection 3.7.3.9 as follows:

Independent Support Motion - Method B - When there are more than one supporting structure, the independent support motion (ISM) method for seismic response spectra may be used.

Each support point is considered to be in a random- phase relationship to the other supports. The displacement response in the modal coordinate, equation (1), then becomes:

$$q_i = [\sum_{j=1}^{N} (P_{ij}\ d_{ij})^2]^{1/2} \qquad (2)$$

A support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor (or portions of a floor) of a structure. ~~The SSE damping values for piping systems that are analyzed with the ISM method are 3 percent for piping larger than 12-inch diameter and 2 percent for smaller piping.~~

Revise Table 3.7.1-1 as follows:

Table 3.7.1-1

Safe Shutdown Earthquake Damping Values

Welded aluminum structures (%)	4
Welded and friction-bolted steel structures and equipment (%)	4
Bearing bolted structures and equipment (%)	7
Prestressed concrete structures (%)	5
Reinforced concrete structures (%)	7
~~Primary coolant loop (%)~~	~~4 or ASME Code Case N411~~
Piping systems(for response spectra analysis)	~~ASME Code Case N411~~ 5
Piping systems (alternative for time history analysis)	
Less than or equal to 12-inch diameter (%)	2
Greater than 12-inch diameter (%)	3
Primary coolant loop (%)	4
Fuel assemblies (%)	20
Control rod drive mechanisms (%)	5
Cable trays & related supports (%) (see Figure 3.7.1-13)	20
Conduits & related supports (%)	7
HVAC ductwork (%)	7
Cabinets and panels for electrical equipment (%)	5
Equipment such as welded instrument racks and tanks (%)	3

Question 220.17

The bearing stress of 33.6 ksf due to the dead load, live load, and safe shutdown earthquake described in Section 3.8.5.5.1 of the SSAR should be included in Table 2.0-1 as the minimum dynamic soil bearing capacity. Modify the table or provide justification for not doing so.

Response:

The bearing stress of 33.6 ksf due to the dead load, live load, and safe shutdown earthquake described in Section 3.8.5.5.1 of the SSAR is the result of the analysis performed for design of the basemat. The analysis used a conservative approach to bound the range of potential sites. This is described in Subsection 3.8.5.4, where it is stated:

> "Safe shutdown earthquake loads for the soft rock case, in combination with the properties of soft-to-medium soft soil, are used in the analysis since the soft rock case produces higher applied seismic forces to the structure than the soft to medium soft soil case. Hence, the approach is conservative."

The evaluation of the soils and design of soil improvement, if required, is part of the Combined Licence application and is site-specific. It is conservative to use the reactions calculated from the analyses of the base mat. However, this is unduly conservative for certain sites. It is sufficient to demonstrate that the bearing reactions for the site-specific soils and SSE are acceptable based on site-specific analyses. As a result, the maximum bearing reaction is not included as the minimum dynamic soil bearing capacity in Table 2.0-1.

The AP600 interfaces for standard design, Table 1.8-1, Item number 2.13, states that the bearing capacity of foundation materials is a site-specific item to be qualified by the Combined License applicant. As stated in the second paragraph of Section 2.5, the Combined License applicant must demonstrate that the proposed site "... can support the foundation mat of the AP600 under all specified site conditions. There is no potential for liquefaction at the plant site due to a safe shutdown earthquake." Furthermore, the last paragraph of Section 2.5 provides that "Bearing loads during seismic conditions for the generic plant are the base reactions from the seismic analyses described in Subsection 3.7.2. The Combined License applicant may either use these loads to demonstrate soil bearing acceptability or may perform site-specific seismic analyses to develop bearing loads applicable to his site and seismic conditions." Finally, the last sentence of Section 3.8.5.5.1 repeats the requirement of "the Combined License applicant will address the interface capability of the soil to support the applied foundation loads."

The soil bearing parameter in Table 2.0-1 will be modified to show that the Combined License applicant must address the dynamic capability of the soil to support the applied foundation loads. In addition , the last paragraph of Section 2.5 will be revised to reference/accentuate the combined static and dynamic foundation stress.

SSAR Revision: (Revision 1)

Note - The revision shown is to Revision 1 of the SSAR. The original response to RAI 220.17 added the information on bearing strength.

The site interface for soils in Table 2.0-1 will be revised as follows:

Soil	
Bearing Strength	Soils must support the AP600 under ~~all~~ specified conditions, including loads induced by the safe shutdown earthquake. The ~~average static bearing reaction of the AP600 is about 8000 pounds/square foot; the~~ maximum static bearing reaction at a corner is 12000 ~~11000~~ pounds/square foot
Shear Wave Velocity	Greater than or equal to 1000 ft/sec
Liquefaction Potential	None

Note - The following revision was part of the original response and incorporated into SSAR Revision 1

The last paragraph of Section 2.5 will be revised to read:

The average static bearing reaction of the AP600 is about 8000 pounds/square foot; the maximum bearing reaction at a corner is 12000 pounds/square foot. Bearing loads during seismic conditions for the generic plant are the base reactions from the seismic analyses described in Subsection 3.7.2. **The maximum bearing stress due to the dead load, live load, and safe shutdown earthquake is presented in Subsection 3.8.5.5.1 for the worst combination of site and soil conditions.** The Combined License applicant may either use these loads to demonstrate soil bearing acceptability or may perform site-specific seismic analyses to develop bearing loads applicable to ~~his~~ the site and seismic conditions.

Question 220.27

Provide the potential sources of a missile or sources of high pressure resulting from high-energy line break between the steel containment and the operating floor and refueling cavity walls, between the secondary shield walls and the steel containment, and between the steel containment and the shield building (Section 3.8.2 of the SSAR).

Response: (Revision 1)

Potential sources of missiles inside the containment are discussed in SSAR Subsection 3.5.1.2. Criteria are defined to determine if certain rotating equipment or high energy systems could result in credible missiles. When the equipment is procured and detail design information is available, the equipment will be reviewed against the criteria defined in SSAR Subsection 3.5.1.2. If missiles are determined to be credible, an evaluation will be performed to confirm that such missiles do not jeopardize safe shutdown.

High energy piping is identified in SSAR Appendix 3E. These figures show the containment boundary. Subcompartments are designed for the pressure and temperature effects calculated for the postulated pipe breaks as described in SSAR Subsection 3.6.1.2.1. Table 220.27-1 lists the high energy piping (greater than 1 inch nominal diameter) inside each compartment within the containment, showing the nominal size of each line **along with the terminal end break locations. High stress intermediate break locations, if any, are also evaluated for subcompartment pressure and temperature effects. See revision to Subsection 3.6.1.2.1 in response to RAI 210.76 for break location definition**. The subcompartments are identified using the room numbers and room names given on SSAR non-proprietary Figures 1.2-4 **through** ~~thru~~ 1.2-10 **as supplemented by Table 220.27-2**. There is no high energy piping that can pressurize the annulus between the containment vessel and the shield building. Guard pipes are provided for the mainsteam, feedwater and steam generator blowdown containment penetrations passing through the annulus as shown in SSAR Figures 3.8.2-4. The CVS makeup piping is classified as high energy due to its design pressure but does not cause pressurization because it is at ambient temperature.

SSAR Revision: NONE

TABLE 220.27-1
AP600 SUBCOMPARTMENTS AND POSTULATED PIPE RUPTURES

~~COMPARTMENT~~	~~LINES QUALIFIED TO LBB ASME Class 1 and 2~~	~~LINES NOT QUALIFIED TO LBB~~
~~ROOM NUMBERS 11201, 11301, 11401, 11501 STEAM GENERATOR COMPARTMENT 1~~	~~31" Hot Leg (RCS)~~ ~~22" Cold Leg (RCS)~~ ~~18" Surge Line (RCS)~~ ~~12" Fourth stage ADS (RCS)~~ ~~10" Passive RHR (RCS/PXS)~~ ~~16" Feed Water (SGS)~~ ~~4" Pressurizer spray (RCS)~~ ~~4" SG Blowdown (SGS)~~	~~3" Purification (CVS)~~ ~~2" SG Blowdown (SGS)~~
~~ROOM NUMBERS 11202, 11302, 11402, 11502 STEAM GENERATOR COMPARTMENT 2~~	~~31" Hot Leg (RCS)~~ ~~22" Cold Leg (RCS)~~ ~~12" Fourth stage ADS (RCS)~~ ~~20", 12" Normal RHR (RCS/RNS)~~ ~~16" Feed Water (SGS)~~ ~~8" CL to CMT (PXS)~~ ~~4" SG Blowdown (SGS)~~	~~2" SG Blowdown (SGS)~~
~~ROOM NUMBER 11205 REACTOR VESSEL NOZZLE AREA~~	~~31" Hot Leg (RCS)~~ ~~22" Cold Leg (RCS)~~ ~~8" Direct Vessel Injection (RCS)~~	~~None~~
~~ROOM NUMBER 11206 PXS VALVE AND ACCUMULATOR ROOM A~~	~~8" Direct Vessel Injection (RCS/PXS)~~ ~~8" line from CMT (PXS)~~ ~~6" line from IRWST (PXS)~~	~~2" CMT (PXS)~~ ~~2" Accumulator (PXS)~~
~~ROOM NUMBER 11207 PXS VALVE AND ACCUMULATOR ROOM B~~	~~8" Direct Vessel Injection (RCS/PXS)~~ ~~8" line from CMT (PXS)~~ ~~6" line from IRWST (PXS)~~	~~2" CMT (PXS)~~ ~~2" Accumulator (PXS)~~

~~ROOM NUMBER 11208~~ ~~RNS VALVE ROOM~~	~~12", 10" Normal RHR (RNS)~~	~~None~~
~~ROOM NUMBER 11209~~ ~~CVS ROOM~~	~~None~~	~~3" Purification (CVS)~~ ~~2" (CVS)~~
~~ROOM NUMBER 11303~~ ~~PIPE AND VALVE ROOM~~ ~~(BELOW PRESSURIZER)~~	~~18" RCS Surge Line (RCS)~~ ~~10" Passive RHR (PXS)~~ ~~4" Pressurizer spray (RCS)~~ ~~4" SG Blowdown (SGS)~~	~~None~~
~~ROOM NUMBER 11403~~ ~~LOWER PRESSURIZER~~ ~~COMPARTMENT~~	~~10" Passive RHR (PXS)~~ ~~6" Pressurizer spray line (RCS)~~	~~None~~
~~ROOM NUMBER 11503~~ ~~UPPER PRESSURIZER~~ ~~COMPARTMENT~~	~~14" ADS (RCS)~~ ~~10" Passive RHR (PXS)~~ ~~6" Pressurizer spray (RCS)~~	~~None~~
~~ROOM NUMBER 11300~~ ~~MAINTENANCE FLOOR~~ ~~(LOWER COMPARTMENT)~~	~~32" Main Steam (SGS)~~ ~~16" Feed Water (SGS)~~ ~~10" Passive RHR (PXS)~~ ~~4" SG Blowdown (SGS)~~	~~3" Purification (CVS)~~ ~~2" (CVS)~~
~~ROOM NUMBER 11500~~ ~~OPERATING DECK~~ ~~(UPPER COMPARTMENT)~~	~~32" ID Main Steam (SGS)~~ ~~16" ID Feed Water (SGS)~~ ~~10" PRHR (PXS)~~ ~~14", 8", 4" ADS (RCS)~~ ~~6" Pressurizer Safety (RCS)~~	~~None~~

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Steam generator compartment 1	11201	22" Cold Leg (RCS)	RC pump nozzles (2)		
		12" Fourth Stage ADS (RCS)	Hot leg nozzle		
		4" Pressurizer spray (RCS)	Cold leg nozzles (2)		
	11301	31" Hot Leg (RCS)	SG nozzle	3" Purification (CVS)	3" by 10" PRHR branch
		18" Surge Line (RCS)	Hot leg nozzle		
		12" & 10" Fourth stage ADS (RCS)	Valves: V014A/C		
		10" PRHR return (RCS)	SG nozzle		
		4" Pzr Spray(RCS)	None		
	11401	16" Feedwater (SGS)	None	4"SG Blowdown (SGS)	2 1/2" SG nozzle
		10" PRHR Supply(PXS)	None	2" SG Drain (SGS)	2 1/2" SG nozzle, 4"x2" Tee
		4" Pzr Spray(RCS)	None		
	11501	16" Feedwater (SGS)	None		
	11601	16" Feedwater (SGS)	SG nozzle		
	11701			2" SG Wet Layup (SGS)	SG nozzle

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Steam generator compartment 2	11202	22" Cold Leg (RCS)	RC pump nozzles (2)		
		12" Fourth stage ADS (RCS)	Hot leg nozzle		
		20", 12" Normal RHR (RCS/RNS)	Hot leg nozzle, 20"x12" Reducer		
	11302	31" Hot Leg (RCS)	SG nozzle		
		12" & 10" Fourth stage ADS (RCS)	Valves: V014B/D		
		8" Cold Leg to CMT (RCS)	Cold leg nozzles (2)		
	11402	16" Feedwater (SGS)	None	4" SG Blowdown (SGS)	2 1/2" SG nozzle
		8"Cold Leg to CMT (RCS)	None	2" SG drain (SGS)	2 1/2" SG nozzle, 4"x2" Tee
	11502	16" Feedwater (SGS)	None		
	11602	16" Feedwater (SGS)	SG nozzle		
	11702			2" SG Wet Layup (SGS)	SG nozzle
Reactor vessel nozzle area	11205	31" Hot Leg (RCS)	Reactor vessel nozzles (2)		
		22" Cold Leg (RCS)	Reactor vessel nozzles (4)		
		8" Direct Vessel Injection (RCS)	Reactor vessel nozzles (2)		

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
PXS Valve and accumulator room A	11206	8" Accumulator Injection (PXS)	Accumulator nozzle	2" CMT drain PXS)	2" by 8" Branch & Valve- V012A
		8" Direct Vessel Injection (RCS)	None		
		8" CMT Injection (PXS)	CMT nozzle	2" Accumulator drain (PXS)	2" by 8" Branch & Valve - V026 A
		6" line from Normal RHR (RNS)	Valve: V017A		
		6" line from IRWST (PXS)	Valves: V125A & V123A		
PXS Valve Room B	11207 -PXS	6" line from Normal RHR (RNS)	Valve: V017B	2" CMT drain (PXS)	2" by 8" Branch & Valve - V012 B
		8" Direct Vessel Injection (RCS)	None		
		6" line from IRWST (PXS)	Valves: V125B & V123B		
Accum. Room B	11207 ACCUM	8" Accumulator Injection (PXS)	Accumulator nozzle	2" Accumulator drain (PXS)	2" by 8" Branch & Valve - V026 B
		8" CMT Injection (PXS)	CMT nozzle		
SG Cmpt Vertical Access Tunnel	11204	8" Direct Vessel Injection (RCS)	None	3" Line from Regen HX to SG 01 (CVS)	Anchor at wall
				3" Purification from Cold Leg to Regen. HX (CVS)	Anchor at wall
RNS valve room	11208	12"/10" Normal RHR (RNS)	Valves: V001A/B		

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Pressurizer Compartment	11303	18" Surge line (RCS) (in 11303 & 11403)	Pressurizer nozzle	3" Purification (CVS)	None
		10" Passive RHR return (PXS)	PRHR HX nozzle	2" Aux Spray (CVS)	None
				4" SG Blowdown (SGS)	None
	11403	18" Surge line (RCS)	Pzr nozzle	3" Purification (CVS)	None
		10" Passive RHR supply (PXS)	PRHR HX nozzle	2" Aux Spray (CVS)	None
	11403 Valve Room	4" Pzr spray line (RCS)	None	2" Aux spray (CVS)	4"x2" Tee
				3" Purification (CVS)	None
	11503	14" ADS (RCS)	Pzr nozzles (2)		
		4" Pzr spray line (RCS)	Pzr nozzle		
		10" PRHR supply (PXS)	None		
	11603	14", 8" & 4" ADS (RCS)	Valves: V011A, V012A & V013A, 14"x8" Reducer (2), 14"x4" Reducer		
		6" Pzr safety (RCS)	Valves: V005A/B, 14"x6" Tee (2)		
	11703	14", 8" & 4" ADS (RCS)	Valves: V011B, V012B & V013B, 14"x8" Reducer (2), 14"x4" Reducer		

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Mainte-nance Floor (Lower Compart-ment)	11300	32" Main Steam (SGS)	None	3" make-up (CVS)	Anchor at Con-tainment pene-tration
		16" Feedwater (SGS)	None		
		10" Passive RHR (PXS)	PRHR HX nozzles (2)	2" SG Wet Layup (SGS)	Valves: V083A/B
		8" CMT Piping	CMT nozzles (2)	2" line to degasifier (CVS)	Valve: V045
				4" SG Blowdown (SGS)	Anchors (2) at Containment penetration
Operating Deck (Upper Compart-ment)	11500	32" Main Steam (SGS)	SG nozzles (2)		
		10"PRHR Supply (PXS)	None		

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
CVS room	11209			3" Purification from anchor to Regen HX (CVS)	Regen HX nozzle
				3" return, aux spray, RNS pump suction (CVS)	Regen HX nozzle
					Valves: V080 & V079
				3" letdown from Regen HX to Letdown HX (CVS)	Regen HX & Letdown HX nozzles.
				3" letdown from Letdown HX to demins (CVS)	Letdown HX nozzle & demin nozzles (5) Valves: V017-A/B & V013A/B
				2" spent resin lines (CVS)	Mixed bed demin nozzles (4), cation bed demin nozzles(2)
					Valves: V015A, V015B, V032
				3" & 2" piping from demins to filters (CVS)	Demin nozzles (2), cation bed demin nozzles (3), filter nozzles (2)
					Valves: V033, V031, V022A & V022B

Compartment		Lines Qualified to LBB		Lines Not Qualified to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
CVS room (continued)				3" & 2" piping from filters to Regen HX (CVS)	Regen HX nozzle, filter nozzles (2) Valve: V044
CVS room pipe tunnel	11209 Pipe Tunnel			3" Makeup(CVS)	None
				3" Purification from Anchor to Regen HX	Anchor
				3" Return from Regen HX to anchor (CVS)	Anchor
Reactor coolant drain tank room	11104	None	None	None	None
Lower Reactor vessel cavity	11105	None	None	None	None

TABLE 220.27-2

AP600 SUBCOMPARTMENTS INSIDE CONTAINMENT

ROOM #	DESCRIPTION	BOTTOM ELEV.	TOP ELEV.
11104	RCDT Room	66'-6"	81'-0"
11105	LOWER RV CAVITY	66'-6"	98'
11205	UPPER RV CAVITY	98'	107'-2"
11201	SG CMPT 1	83'	104'-7"
11202	SG CMPT 2	83'	104'-7"
11204	SG CMP VERT ACCESS TUNNEL	83'	107'-2"
11206	PXS VALVE Room A	87'6"	105'-2"
11300	MAINTENANCE FLOOR	107'-2"	135'-3"
11301	SG CMPT 1	104'-7"	113'-9"
11302	SG CMPT 2	104'-7"	113'-9"
11401	SG CMPT 1	113'-9"	135'-3"
11402	SG CMPT 2	113'-9"	135'-3"
11501	SG CMPT 1	135'-3"	149'-7"
11502	SG CMPT 2	135'-3"	149'-7"
11601	SG CMPT 1	149'-7"	162'-1"
11602	SG CMPT 2	149'-7"	162'-1"
11701	SG CMPT 1	162"-1"	--
11702	SG CMPT 2	162'-1"	--
11500	OPERATING DECK	135'-3"	256'-2 3/8"
11303	PZR LOWER CMPT	107'-2"	118'-6"
11403	PZR CMPT	118'-6"	135'-3"
11403-VAL RM	PZR CMPT VALVE Room	118'-6"	135'-3"
11503	PZR UPPER CMPT	135'-3"	160'-6"
11603	PSADS LOWER TIER	160'-6"	169'-0"
11703	PSADS UPPER TIER	169'-0"	--
11207-ACCUM	ACCUM Room B	87'-6"	105'-2"
11207-PXS	PXS VALVE Room B	96'-0"	105'-2"
11208	RNS VALVE Room	94'	105'-2"
11209	CVS Room	80'-6"	105'-2"
11209 PIPE	CVS Room PIPE TUNNEL	100'-0"	105'-2"

Question 220.83

For each seismic Category I structure, provide in the SSAR, a summary of design information using the format for design loads/results as indicated in Appendix C to Section 3.8.4 of the SRP.

Response: (Revision 1)

As stated in the Standard Review Plan, the primary objective of the Design Report is to provide the NRC reviewer with design and construction information more specific than that contained in the Safety Analysis Report, which can assist in the planning and conduct of a structural audit. The information contained in the Design Report is a level of detail beyond that of the SSAR.

Westinghouse will compile design summary reports using the format for loads and information indicated in Appendix C to Section 3.8.4 of the Standard Review Plan. The design summary reports will be available to the NRC for audit. The summary reports will be completed by July, 1995 and will incorporate those changes in criteria agreed to with the staff prior to issue of the draft Safety Evaluation Report as a result of the review of the SSAR and RAI responses. Table 220.83-1 shows the preliminary outline for these reports. These design reports will be updated during construction to incorporate as-procured and as-constructed information to satisfy the ITAAC for construction of the building structures (ITAAC Table 4.2-1, item 1).

SSAR Revision:

Add the following paragraph at the end of Subsections 3.8.3.4 and 3.8.4.4.1:

The structural design of the nuclear island is summarized in the following design summary reports:

- Nuclear island basemat and stability
- Auxiliary building
- Containment internal structures
- Shield building

The format and content of each of the design summary reports follows the guidelines of Appendix C to Section 3.8.4 of the Standard Review Plan. These design reports are updated during construction to incorporate as-procured and as-constructed information to satisfy the ITAAC for construction of the building structures (ITAAC Table 4.2-1, item 1).

Table 220.83-1
Outline of Design Summary Reports

Nuclear Island Basemat

- Objective
- Description of Nuclear Island Basemat
- Governing Codes and Regulations
- Structural Material Requirements
- Structural Loads and Load Combinations
- Structural Analysis and Design of Basemat
- Overturning and Sliding
- Summary of Results
- Conclusions
- List of Engineering Drawings and Design Calculations
- References

Auxiliary Building

- Objective
- Description of Auxiliary Building
- Governing Codes and Regulations
- Structural Material Requirements
- Structural Loads and Load Combinations
- Structural Analysis and Design of Shear Walls
- Structural Analysis and Design of Structural Steel Framing
- Structural Analysis and Design of Floor and Roof Slabs
- Structural Analysis and Design of Structural Modules
- Summary of Results
- Conclusions
- List of Engineering Drawings and Design Calculations
- References

Containment Internal Structures

- Objective
- Description of Containment Internal Structures
- Governing Codes and Regulations
- Structural Material Requirements
- Structural Loads and Load Combinations
- Structural Analysis and Design of Basemat
- Structural Analysis and Design of Structural Modules
- Structural Analysis and Design of IRWST
- Structural Analysis and Design of Floor Slabs
- Structural Analysis and Design of Structural Steel Columns
- Summary of Results
- Conclusions
- List of Engineering Drawings and Design Calculations
- References

Shield Building Roof

- Objective
- Description of Shield Building Roof
- Governing Codes and Regulations
- Structural Material Requirements
- Structural Loads and Load Combinations
- Structural Analysis and Design of Roof (including containment air baffle, PCS tank, precast panels and roof to cylinder connection)
- Summary of Results
- Conclusions
- List of Engineering Drawings and Design Calculations
- References

Question 220.92

Section 3.8.2.4.2.2 of the SSAR discusses the theoretical elastic buckling pressure capacity of the AP600 using an approach similar to that permitted in ASME Code Case N284. The theoretical elastic buckling pressure was calculated to be 536 psig using BOSOR-4. A reduction factor (defined as the product of the capacity reduction factor and the plastic reduction factor) was established as 0.385 based on the lower bound curve of test results of 20 ellipsoidal and 28 torispherical test specimens. This resulted in a predicted elastic buckling capacity of 206 psig.

For the stiffened and unstiffened spherical shells, Figure 1512-1 in Code Case N284 provides the capacity reduction factors for both (a) equal biaxial compression and (b) uniaxial compression. These capacity reduction factors are determined from lower bound values of test data. The ellipsoidal head of the AP600 containment will experience tension in the meridional direction and compression in the hoop direction under internal pressure. Under this condition, the staff believes that it would be appropriate to use the uniaxial compression capacity reduction factors in Code Case N284 for the AP600. With the uniaxial compression and an unstiffened top head arc length, l_i, of 1231.4 inches ($\pi/4$ x a x b), the capacity reduction factor is 0.21. The plasticity reduction factor is assumed by the staff to be 1.0 because of elastic buckling. Therefore, the staff believes that the reduction factor should be 0.21 (0.21 x 1). Explain why the reduction factor of 0.21 based on Code Case N284 is not applicable to the AP600.

Response:

The purpose of the BOSOR-4 analysis described in the SSAR is to provide an alternate approach for comparison with the results of the BOSOR-5 non-linear analyses which serve as the basis for the estimate of containment pressure capacity. The results of these BOSOR-4 and BOSOR-5 analyses estimated the buckling capacity in the knuckle region to be 206 psig and 174 psig respectively, after applying the capacity reduction factors.

The stresses due to internal pressure for the AP600 containment vessel are shown in SSAR Figure 3.8.2-5. There is a small zone of circumferential compression in the knuckle region, extending over a height of about 15 feet. The circumferential stresses in other regions and the meridional stresses are tensile. This stress state is typical of ellipsoidal and torispherical heads under internal pressure. Buckling in this local region is a function of the local meridional and circumferential radii, the distribution of hoop compression along the meridional length, and the length over which hoop compression is present. This can be considered as a reduced arc length of the shell. As shown in Figure 1512-1 in Code Case N284, there is a significant increase in the capacity reduction factor as the arc length of the shell is reduced. The capacity reduction factor in Code Case N-284 depends upon the parameter (M) which is determined as the smaller of M_ϕ and M_θ, which are calculated using the unsupported lengths and local radii in the meridional and circumferential directions respectively. It is not clear why the staff selected the arc length of 1231.4 inches.

In the BOSOR-4 analyses reported in the SSAR, the capacity reduction factors were determined from lower bound values of test data, thus eliminating the need to estimate the effective parameters of the equivalent sphere. This analysis of test results of 20 ellipsoidal and 28 torispherical test specimens representative of the AP600 head, which resulted in the capacity reduction factor of 0.385, is considered to be an appropriate method to estimate the capacity reduction factor for the AP600. It provided reasonable agreement with the BOSOR-5 analyses which are the basis

for the estimate of containment pressure capacity. The capacity reduction factor of 0.21 suggested in the RAI appears to be based on a uniform sphere under uniaxial compression which is not applicable to the AP600 top head.

SSAR Revision: NONE

Question 230.9

ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," which has not been endorsed by the staff, should be submitted and docketed for the staff review for the AP600 standard design (Section 3.7.1.3).

Response: (Revision 1)

ASCE Standard 4-86 is a publicly available standard published by the American Society of Civil Engineers in 1986. It differs from other codes and standards only in that the NRC Staff has not completed their review and established a position. ~~The NRC staff is requested to complete its review so that this document can be applied to advanced plants.~~The global reference to the standard is deleted from the SSAR as shown below.

SSAR Revision: (Revision 1)

Revise Subsection 3.7.2.1 as follows:

Seismic analyses of the nuclear island are performed in conformance with ~~the guidelines provided in ASCE Standard 4-86 (Reference 3) and~~ the criteria within SRP 3.7.2.

Question 230.15

Section 3.7.2.1.2 of the SSAR states that "Certain subsystems...are analyzed using the time histories obtained from a series of soil-specific analyses." What are these soil-specific analyses? Provide details of these analyses.

Response: (Revision 2)

The soil-specific analyses correspond to the design soil profiles presented in subsection 3.7.1.4. The methods of analysis are presented in Subsection 3.7.3~~, SSE time history analysis will be performed for the reactor coolant loop piping by December 1993~~. The SSE time history analysis for the reactor coolant loop piping is described in SSAR Appendix 3C.

SSAR Revision:

Revise the last paragraph of Subsection 3.7.2.1.2 as follows:

Seismic responses of the nuclear island structures for the various design soil profiles are enveloped and the resulting response spectra are used in the design and analysis for most of the seismic subsystems. Certain subsystems, as described in Subsection 3.7.3.6, are analyzed using the time histories obtained from ~~a series of~~ soil-specific analyses for the design soil profiles presented in Subsection 3.7.1.4.

Question 230.35

The following request for additional information pertains to Section 3.7.2.1.1 of the SSAR:

a. Provide the detailed comparison of the results obtained from the 2D SSI analyses and the 3D response spectrum analyses for the hard-rock site condition.

b. As described in Section 3.7.2.1.1, the structural member forces and moments are obtained from the response spectrum analysis of the finite element model for the hard-rock site, and from the SSI analysis of the stick model for the soil sites. Provide a comparison of responses from the response spectrum analyses of a stick model and a finite element model at rock site.

c. From the staff's review of Section 3.7.2.1.1 and Table 2A.17, the staff determined that the hard-rock site condition (R1) is not the governing case for the steel containment shell. Describe how the steel containment shell was analyzed for the rock site condition.

d. Provide the rationale for excluding the SB roof in the finite element model, as shown in Figure 3.7.2-1.

e. From the staff's review of Tables 3.7.2-1 through 3.7.2-4 of the SSAR, the staff determined that the AP600 nuclear island structures (except the steel containment shell) are very rigid. Some predominant frequencies are much higher than 33 Hz. Provide justification for the statement "since the shear wave velocity for the hard rock site is in excess of 8000 feet per second, the soil- structure interaction effect is negligible." This statement has also been made in Sections 3.7.2.1.2 and 3.7.2.4.

Response: (Revision 1)

a. Maximum member forces for the hard rock (R1) case of the 2D SSI analysis are given in Table 2A-17. Maximum member forces for the hard rock analyses of the 3D stick model using the computer program BSAP are given in Table 3.7.2-11 (sheet 1). Floor response spectra for the R1 case of the 2D analyses are given in Figures 2A-29, 2A-30 and 2A-31. Floor response spectra for the BSAP hard rock analyses of the 3D stick model are given in Figures 3.7.2-29, 3.7.2-30 and 3.7.2-31. The 3D stick model was developed from the finite element model and the frequencies and modal participation of the 3D stick model and finite element model are consistent. The 2D SSI analysis was performed to establish the design soil profiles for the AP600 plant. The 3D response spectrum analysis reported in Revision 1 of SSAR Subsection 3.7.2 for the hard rock site condition was performed to obtain in-plane member forces in the individual elements of the finite element model. There were slight differences in the plant configuration considered in the 2D SSI model and the 3D models. More detailed comparison of results from the two analyses is not meaningful.

b. The response spectrum analysis described in Subsection 3.7.2.1.1 was performed only for the hard rock site and used the three-dimensional finite element models. It generated forces and moments in the various elements such as individual walls and slabs. The member forces and moments obtained in the time history three-dimensional analyses of the lumped-mass stick models (described in the last paragraph of Subsection 3.7.2.1.1)

are typically the total shear force, axial force, and moment at a given elevation in the structure. A direct comparison is not available.

c. Table 3.7.2-12 shows the maximum member forces in the containment vessel stick model for the three design soil conditions (hard rock, soft rock and soft-to-medium stiff soil). These results show that the hard rock case gives the maximum forces. Table 3.7.2-6 shows the maximum absolute accelerations for the same soil conditions. The hard rock case results in the highest accelerations of the vessel, except in the node representing the polar crane where the acceleration in the east-west direction is 6% higher for the soft rock case than for the hard rock case. This is considered in design of the crane girder which uses the crane wheel loads from the polar crane design analyses. These design analyses will be reconciled by the Combined License applicant once the final design of the crane is established.

The steel containment vessel is analyzed using the shell of revolution model for the equivalent static accelerations from the SSI seismic analyses reported in SSAR Table 3.7.2-6.

The analyses of Appendix 2A are intended to select the appropriate soils cases for the 3D analyses reported in SSAR Section 3.7.2. They are not used to define the governing case for the containment vessel design. Table 2A.17 shows the seismic member forces for the containment vessel for these parametric soils analyses. This data is for a configuration in which the containment vessel was supported up to elevation 82'-6". As reported in Table 2A.15 this model had a fundamental frequency of 2.14 Hz in the east-west direction. Based on review of these results a design change was incorporated to raise concrete around the vessel to elevation 100'. This increased the fundamental frequency of the containment vessel to 7.61 Hz (see SSAR Figure 3.7.2-10). This model is included in the analyses of SSAR Section 3.7.2. The analyses of Appendix 2A are appropriate for the selection of soil conditions because the mass of the containment vessel is small compared with that of the rest of the nuclear island.

d. A lumped-mass stick model of the shield building roof structure was constructed and coupled with the finite element model and the stick model of the coupled auxiliary and shield buildings. The stick model of the shield building roof structure was included in all seismic analyses performed. The lumped-mass stick model of the shield building roof was not shown in Figure 3.7.2-1 to maintain visual clarity of the finite element model.

e. For the hard rock site, a fixed-base analysis was performed based on the acceptance criteria specified in Revision 2 of SRP 3.7.2, "For structures supported on rock or rock-like material, a fixed base assumption is acceptable. Such materials are defined by a shear wave velocity of 3500 feet per second or greater at a shear strain of 10^{-3} percent or smaller ...etc." Furthermore, as noted in Section 3.7.2.2, the total cumulative mass of the nuclear island participating in the seismic response, up to the frequency limit of 34 Hz, constitute 90, 90 and 83 percent of the total mass, excluding the building mass within the embedded portion. The predominant frequencies of the coupled auxiliary/shield buildings and the steel containment vessel are below 34 Hz. The relatively rigid containment internal structures, coupled to the other flexible structures on a common basemat, are expected to have negligible effect in the overall soil-structure interaction responses of the nuclear island. Therefore, for the hard rock site, only a fixed-base analysis is required.

SSAR Revision:

Revise the first paragraph of Subsection 3.7.2.3.1 as follows:

The finite element models of the coupled shield and auxiliary buildings and the reinforced concrete portions of the containment internal structures are based on the gross concrete section with the modulus based on the specified compressive strength of concrete of contributing structural walls and slabs. The properties of the concrete-filled structural modules are computed using the combined gross concrete section and the transformed steel face plates of the structural modules. Furthermore, the weight density of concrete plus the uniformly distributed miscellaneous dead-weights are considered by adjusting the material mass density of the structural elements. Major equipment weights are included as concentrated lumped masses at the equipment locations. Figures 3.7.2-1 and 3.7.2-2 show, respectively, the finite element models of the coupled shield and auxiliary buildings and the containment internal structures. A lumped-mass stick model of the shield building roof structure is coupled with the finite element model and the stick model of the coupled auxiliary and shield buildings. The stick model of the shield building roof structure is included in the seismic analyses. The lumped-mass stick model of the shield building roof is not shown in Figure 3.7.2-1 to maintain visual clarity of the finite element model.

Question 230.58

Provide the following information pertaining to the high frequency modes of the structures:

a. Provide justification to demonstrate that the time steps used in the time-history seismic analyses are small enough to account for the high-frequency modes that have significant mass participation factors.

b. Make "missing mass" corrections to the seismic analyses (horizontal as well as vertical) where significant high-frequency modes were left out. Note that the seismic forces computed without such "missing mass" correction (if applicable) would result in underprediction (example: a foundation mat design where seismic forces were used in the equivalent static analysis).

Response: (Revision 2)

Seismic analyses for the hard rock soil case are performed using mode superposition time history analysis which might be affected by the effect of high frequency modes of response. Seismic analysis for the soft rock and the soft-to-medium soil cases are performed using the complex frequency response analysis method which considers all masses of the model, and therefore additional consideration is not required.

The synthetic time histories were based on time steps of 0.01 seconds. As shown in SSAR Figures 3.7.1-6 to 3.7.1-8, the spectra for these time histories match the design spectra and satisfy the requirement in Standard Review Plan 3.7.1, f~~. F~~or time history analysis of structures having significant modes up to 33 Hertz. For structural response analyses an integration time step of 0.005 seconds was used. ~~, the input time histories are interpolated to give time steps of 0.005 seconds.~~

The containment internal structure is a relatively stiff structure with significant response of high frequency modes as shown in Table 3.7.2-3 of the SSAR. Mode superposition time history analysis for the containment internal structures includes high-frequency modes to bring the cumulative participated mass up to an acceptable level, and the time step size of 0.005 second becomes unacceptable. Therefore, as presented in the third paragraph of Subsection 3.7.2.2 of the SSAR, the member forces for the 3-dimensional lumped-mass stick model of the containment internal structures for the hard rock soil case are calculated by the response spectrum analysis including the high frequency responses using the procedure given in Appendix A to SRP 3.7.2, Revision 2. Hence the "missing mass" correction is considered in the seismic forces computed from the containment internal structures.

For the basemat design, seismic forces and accelerations for the soft rock and soft-to-medium soil properties are obtained from the SASSI analyses of the 3D lumped mass stick models. In SASSI analysis "missing mass" is not a concern and corrections are not required. Seismic forces and accelerations for the hard rock soil profile are obtained as described in the previous paragraphs and the "missing mass" correction has been considered.

SSAR Revision: NONE

Question 230.59

Provide, in the SSAR, a comparison between the SRSS method and the 1.0, 0.4, 0.4 method, or the bases for use of only the 1.0, 0.4, 0.4 method for the combination of seismic loads. Also Q220.67.

Response: (Revision 1)

Note: The reference to Q220.67 should be to Q220.66

In the AP600 two methods are permitted for the combination of the effects due to three spatial components of an earthquake using response spectrum methods. The SRSS method is identified in Regulatory Guide 1.92 as an acceptable method to combine maximum structural response values associated with each of the three components of earthquake motion. Co-directional structural responses of interest (eg., stress, deflection, strain, seismic anchor motion) are calculated for each of the three components of earthquake motion. The term "co-directional response" indicates that it is a unidirectional response with contributions from each of the three directions of seismic input. The co-directional responses due to the three directions of seismic input are combined by the SRSS method in order to obtain the estimated maximum response. This is appropriate when the design methods are based on allowable stresses or deflections for a single direction of response. **When there is more than one response parameter to be used in the design calculation** ~~Certain formulations~~ (eg., principal stresses), **the SRSS method** may become overly conservative ~~when using the SRSS method~~ since stresses in two directions are each taken at their estimated maximum response. For these cases the 40% method is considered appropriate.

The 1.0, 0.4, 0.4 method, referred herein as the 40% method, is appropriate for nuclear plant applications. An example of two references that allow its use are given below:

NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," Newmark and Hall, May 1978, Prepared for U.S. Nuclear Regulatory Commission, p. 30.

> "It is conservative, simpler, and much more readily defined and calculated to take the combined effects as 100 percent of the effects due to motion in one particular direction and 40 percent of the effects corresponding to the two directions of motion at right angles to the principal motion considered. It is this combination that is recommended for general use, especially in nuclear power plant design."

ASCE Standard, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures," ASCE 4-86, American Society of Civil Engineers, September 1986, Section 3.2.7.1.2, pp. 24 and 25.

> "Alternatively, the responses may be combined directly, using the assumption that, when the maximum response from one component occurs, the responses from the other two components are 40% of the maximum. In this method, all possible combinations of the three components, ..., including variations in sign (plus or minus), shall be considered. .... ."

To further support the use of the 40% seismic criteria method, comparisons between the SRSS method and the 40%

method are given below. Combinations of maximum co-directional component responses and principal stresses are considered. In order to compare the results from the SRSS and 40% methods, the results obtained from these two methods are compared to those obtained from a time history analysis. A comparison of the two methods is also provided in the response to RAI 220.29 for the containment vessel.

1. Combination of maximum co-directional component responses

   A representative set of co-directional responses are assumed having different relationships between these responses. Figure 230.59-1 shows these relationships for each of one hundred cases. The co-directional response for the X shock, Y shock, and Z shock have all been normalized by the maximum response. There are various cases that include two of the components of equal magnitude, three of equal magnitude, and many cases in between with one component being dominant. Figure 230.59-2 shows the formulation and results of SRSS and 40% combination methods. In only one case does the 40% combination method yield results that are lower (only 1%, 1.414 versus 1.4) than the SRSS method. This is when two of the components are equal, and the other is zero.

2. Principal Stresses

   Principal stresses in a plate were studied along with the maximum shear stress and stress intensity. The sum of σ_Y and τ is also included in the study since it is representative of design for tangential shear. It was assumed that there was a shear stress (τ) directly proportional to the X seismic input (with no contribution from the Y and Z seismic input), that one membrane stress was zero, and that the other membrane stress (σ_Y) was a combination of the co-directional responses due to the Y and Z seismic input (with no contribution from the X seismic input). This would be representative of the seismic response of a shear wall or the containment vessel. The magnitudes of the X, Y and Z responses were those shown in Figure 230.59-1.

   Two of the cases are shown as examples in Table 230.59-1. Case 1 has the response components X and Y equal with Z zero. Case 100 has the response components X, Y, and Z all equal. In the SRSS method the co-directional responses for the Y and Z seismic input are first combined by SRSS to give the maximum membrane stress, which is then used with the shear stress to calculate the principal stresses. The ratio between the 40% method and the SRSS results range from 0.75 to 0.91 for the various principal stress combinations, and are 0.70 to 0.75 for the combination of σ_Y and τ. The 40% method results are lower than the SRSS methods by as much as 30%. The reason for the difference is that the SRSS method does not reflect the statistical independence of the individual co-directional responses.

   The results, along with the associated SRSS and 40% method formulations, are shown in Figures 230.59-3. Two combinations were studied so as to reflect the effect of sign of the components on the results. One combination considered all of the co-directional responses X, Y, and Z as positive, while the other considered Y and Z as negative, and X positive. The results were similar with Sigma 1 (σ_1) and 2 (σ_2) reversing themselves. The results for the 40% method and the SRSS method are similar to those given in Table 230.59-1, recognizing that the SRSS method tends to reflect the absolute summation of responses in complex motions.

3. Time History Comparison Results

The 40% method and the SRSS method were compared against results using two sets of time histories. The first set of time histories were the seismic input time histories as described in SSAR Subsection 3.7.1, which are of equal magnitude (0.3g) and are statistically independent. In addition, arbitrary time histories were developed as shown in Figure 230.59-4. For these time histories no attempt was made to assure that each component resulted from statistically independent motions. These time histories were considered as the component responses (X, Y, and Z) for the same examples of co-directional component response and principal stresses discussed previously. As in the first section that discussed co-directional component response cases, the maximum co-directional responses associated with the X, Y, and Z components represent the stresses as used in the respective formulations as shown on Figure 230.59-2 and 230.59-3. The results are shown in Table 230.59-2. For the co-directional resultant response, the 40% method produced results equal to 89% of the time history method and the SRSS method gave results equal to 85% of the time history method. For the principal stresses, the results obtained using the SRSS combination method are the more conservative. The results obtained for the 40% combination method are close to the time history results with the smallest result being smaller by only 12 percent. Note that these examples were selected specifically to maximize the difference between the various methods and more practical cases would not show as much difference.

In conclusion, the 40% combination method provides realistic results that are not overly conservative. The 40% method is a valid method for combining multiple directional seismic responses. This method provides a margin for those design cases involving combinations of multi-directional responses that is consistent with the margin obtained by use of the SRSS combination for a co-directional response.

SSAR Revision: NONE

Table 230.59-1 Principal Stress Example

Stress Component	Seismic Response Due to X, Y, Z Input			SRSS	40% Method				Ratio 40% to SRSS
Case 1	X	Y	Z		1, .4, .4	.4, 1, .4	.4, .4, 1	Max	
σ_Y	0	1	0	1.0	0.40	1.0	0.40	1.0	1.00
τ	1	0	0	1.0	1.0	0.40	0.40	1.0	1.00
τ_{max}				1.118	1.020	0.640	0.447	1.020	0.91
σ_1				1.618	1.220	1.140	0.647	1.220	
σ_2				-0.618	-0.820	-0.140	-0.227	-0820	
Max.Abs σ_1, σ_2				1.618	1.220	1.140	0.647	1.220	0.75
SI				2.236	2.040	1.280	0.874	2.040	0.91
$\sigma_Y + \tau$				2.0	1.40	1.40	0.80	1.40	0.70
Case 100	X	Y	Z		1, .4, .4	.4, 1, .4	.4, .4, 1	Max	
σ_Y	0	1	1	1.414	0.80	1.40	1.40	1.40	0.99
τ	1	0	0	1.000	1.0	0.40	0.40	1.0	1.0
τ_{max}				1.225	1.077	0.806	0.806	1.077	0.88
σ_1				1.932	1.477	1.506	1.506	1.506	
σ_2				-0.518	-0.677	-0.106	-0.106	-0.677	
Max.Abs σ_1, σ_2				1.932	1.477	1.506	1.506	1.506	0.78
SI				2.449	2.154	1.612	1.612	2.154	0.88
$\sigma_Y + \tau$				2.414	1.80	1.80	1.80	1.80	0.75

Table 230.59-2 - Time History Comparisons

	AP600 Time Histories			Arbitrary Time Histories Figure 230.59-4		
Stress State	Time History	SRSS	40%	Time History	SRSS	40%
Co-directional Resultant Response	0.61	0.52	0.54	2.31	2.72	3.01
Principal Stresses						
Max Shear Stress	0.31	0.37	0.32	1.98	2.17	2.02
Max Principal Stress	0.49	0.58	0.45	2.89	3.11	2.55
Stress Intensity	0.63	0.73	0.65	3.96	4.33	4.04
Sigma Y + Shear Stress	0.61	0.72	0.54	2.64	3.84	3.01

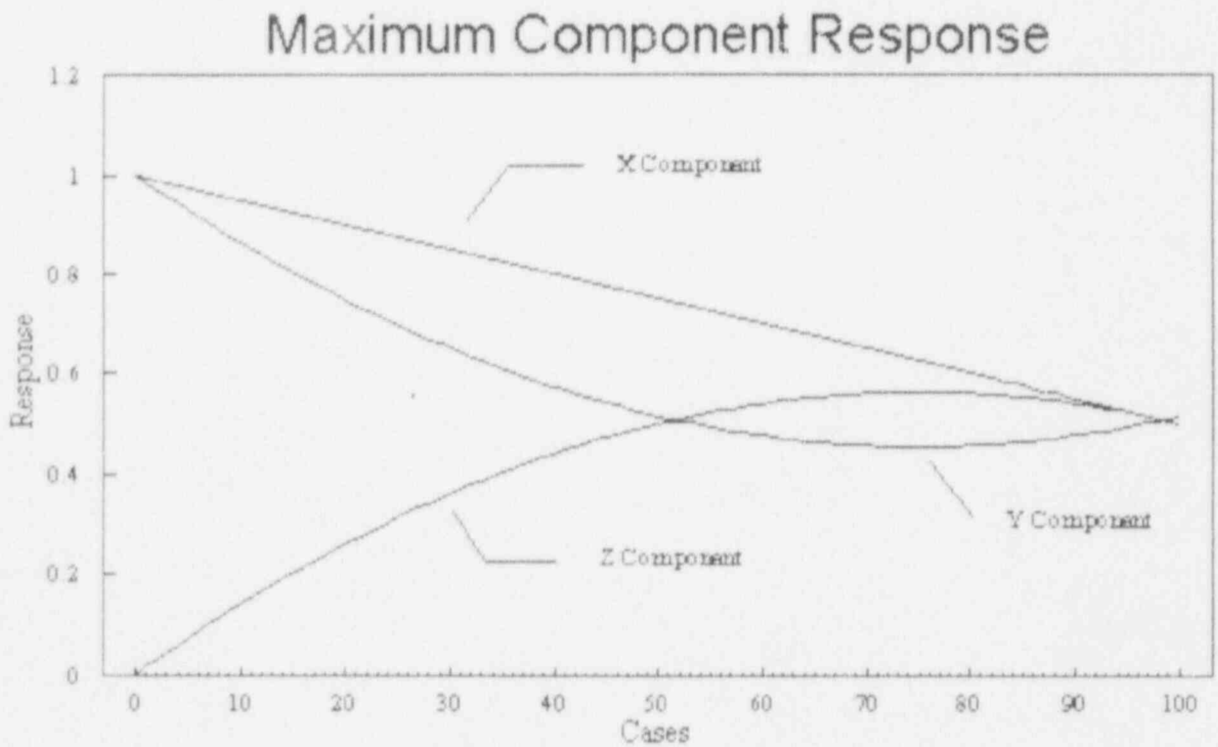
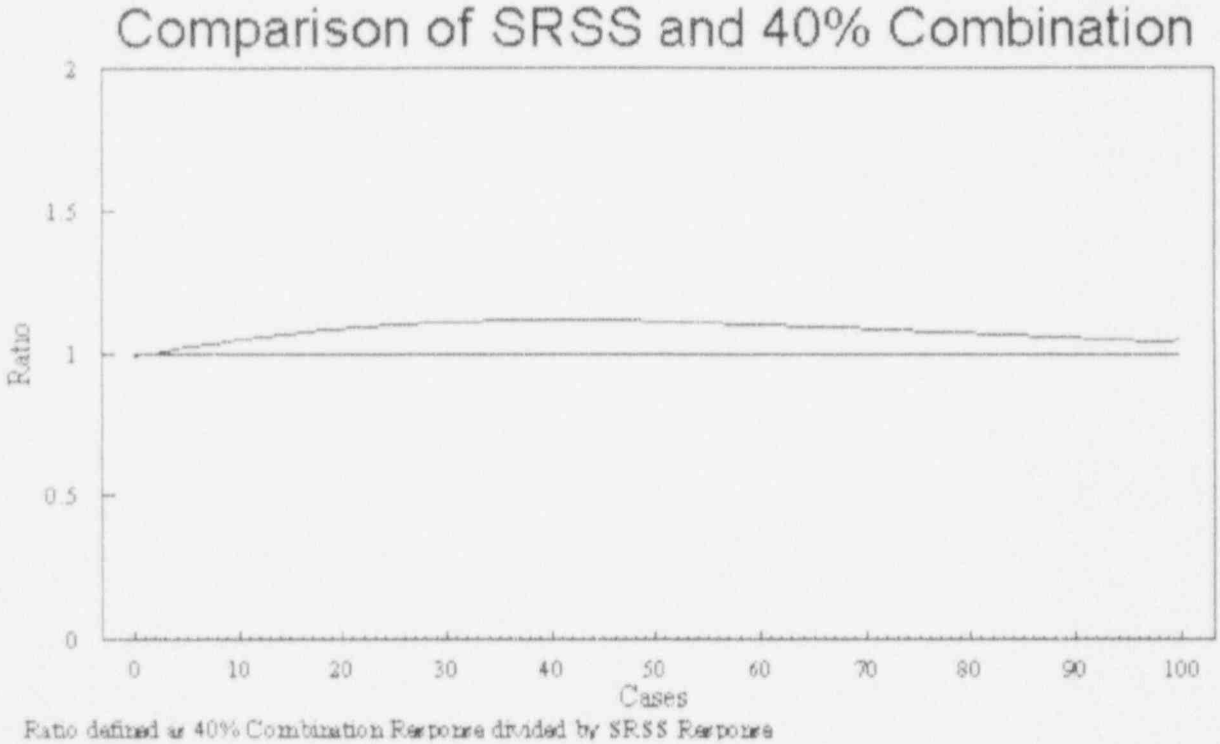


Figure 230.59-1 - Relationships between Maximum Component Responses for X, Y, Z

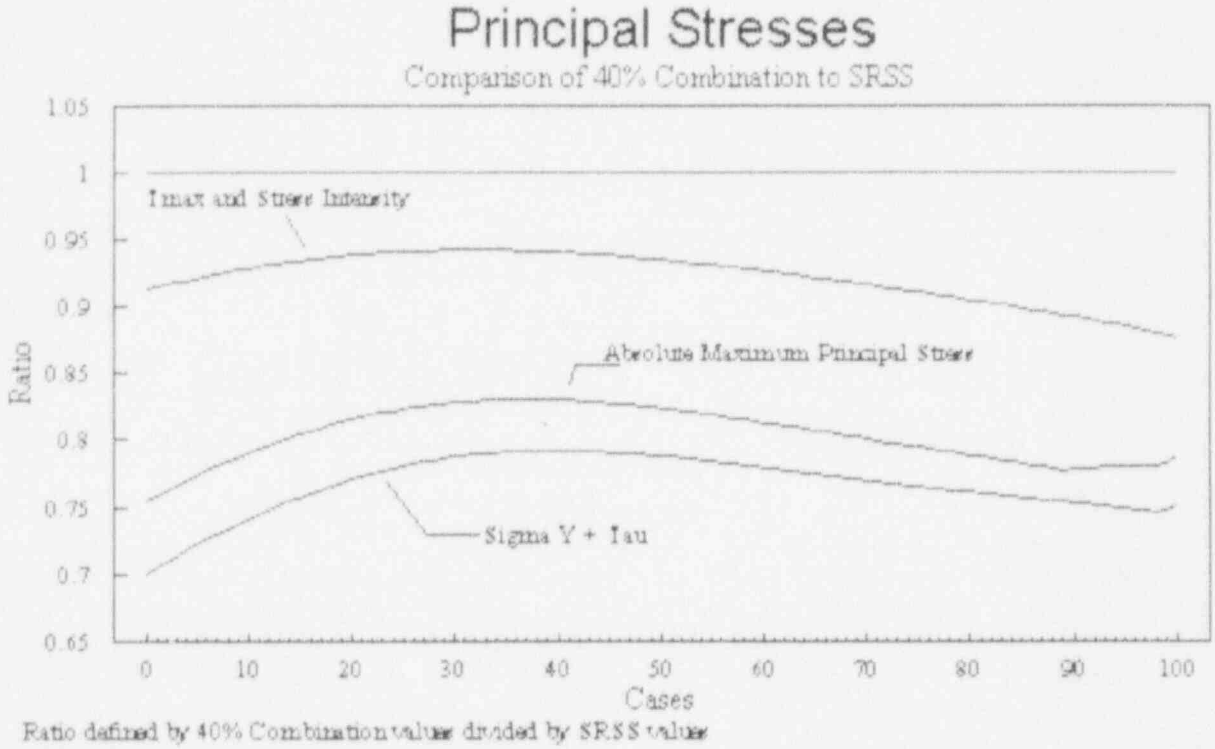


Formulations

$$SRSS_{Response} = (X^2 + Y^2 + Z^2)^{1/2}$$

$$40\%_{Response} = \text{Max}\ [\ (X + 0.4(Y + Z));\ (Y + 0.4(X + Z));\ (Z + 0.4(X + Y))]$$

Figure 230.59-2 - Resultant Response Comparisons



Formulations

General

$\sigma_X = 0$; $\sigma_Y = f(Y,Z)$; $\tau = g(X)$

$\tau_{max} = [(\sigma_Y^2 + 4\tau^2)^{1/2}]/2$

$\sigma_1 = (\sigma_Y/2) + \tau_{max}$

$\sigma_2 = (\sigma_Y/2) - \tau_{max}$

Stress Intensity = Max Absolute Value of $[(\sigma_1 - \sigma_2), \sigma_1, \sigma_2]$

SRSS

$\sigma_Y = (Y^2 + Z^2)^{1/2}$, note that σ_Y retains the sign of Y and Z; $\tau = X$

τ_{max}, σ_1, σ_2 = As Shown Above

40% Combination

$\sigma_Y = \beta Y + \gamma Z$

$\tau = \alpha X$

τ_{max}, $(\sigma_1)_{max}$, $(\sigma_2)_{max}$, Stress Intensity defined as the absolute max value (note sign retained) from results for three sets of (α, β, γ). Where the three sets of (α, β, γ) = [(1., .4, .4); (.4, 1., .4); (.4, .4, 1.)]

Figure 230.59-3 - Principal Stress Comparison

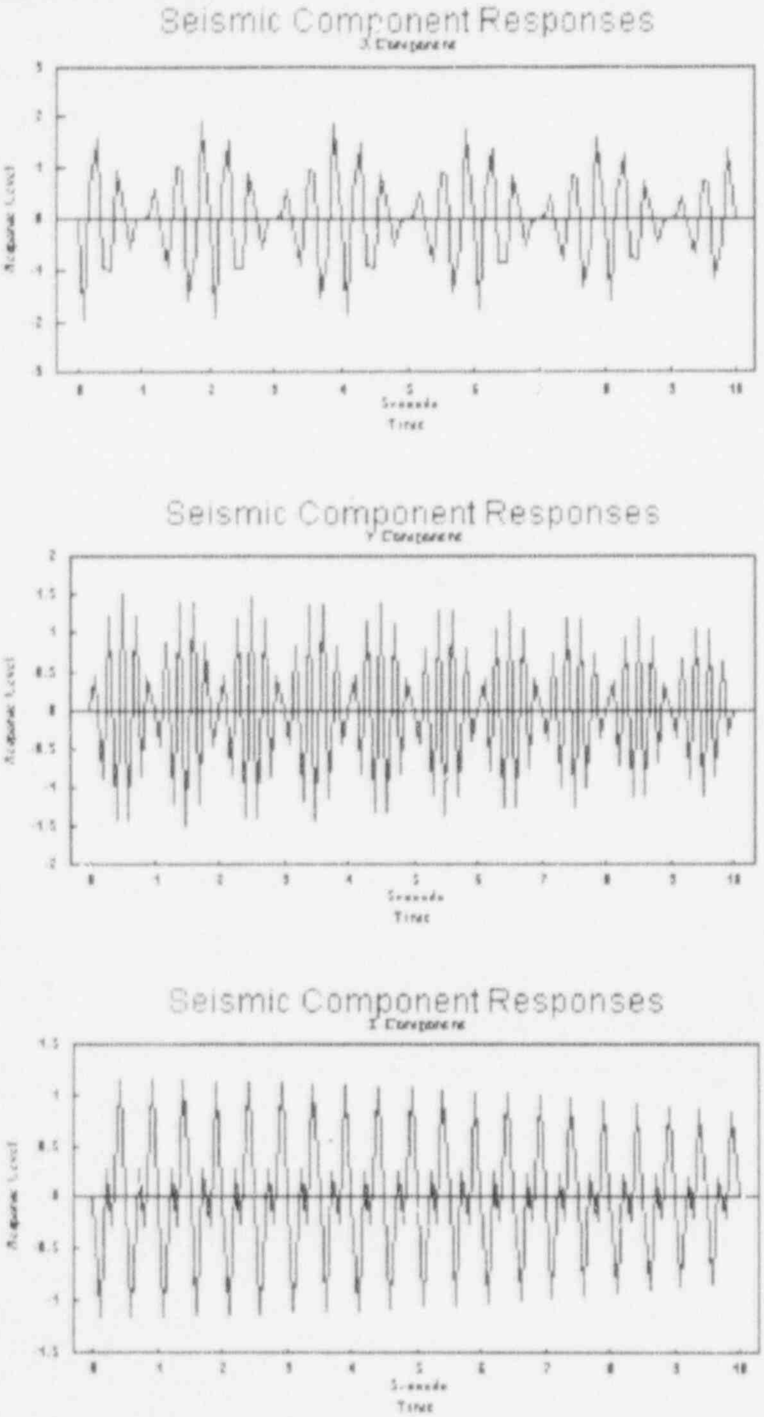


Figure 230.59-4 - Time History Response Characteristics for X, Y, and Z Components

Question 230.82

The first paragraph of Section 3.7.2.1.1 of the SSAR states that response spectrum analyses, using computer code SAP, are performed to obtain the seismic forces and moments for the structural design of the auxiliary building, the shield building, and the containment internal structures on the nuclear island (NI). However, in the third paragraph of the same section, it is stated that Table 2A.17 of the SSAR shows that the hard rock site governs the seismic response forces and moments for the AP600 seismic Category I structures (the auxiliary building, the shield building, and the containment internal structures). Based on the above,

a. describe which method of analysis was used to calculate the seismic forces and moments for the design of the containment vessel for each of the three design site conditions,

b. clarify the inconsistency between Table 2A.17 and Section 3.7.2.1.1, in which a statement is made that the seismic loads for the hard rock site do not always govern, and

c. provide the basis for making the statement in the last paragraph of Section 3.7.2.1.1 that, in such cases, the seismic forces used for the design of NI structures are obtained by multiplying the results from the hard rock response spectrum analysis at each elevation by the ratio of the soil case to the hard rock case member forces at that elevation.

Response: (Revision 1)

a. The steel containment vessel is analyzed using a shell of revolution model for the equivalent static accelerations from the SSI seismic analyses reported in SSAR Table 3.7.2-6.

b. The inconsistency between the two subsections has been corrected. The reference to Table 2A.17 in the third paragraph of Subsection 3.7.2.1.1 is deleted as shown below.

c. As stated in the responses to RAIs 230.50 and 230.90, the response spectrum analyses of the 3D finite element models are used to obtain the in-plane forces for the design of the floors and walls of the nuclear island structures. The finite element models are also the basis for the 3D lumped mass stick models. The member forces in the 3D lumped mass stick models represent the total forces at a given elevation. Hence the finite element member in-plane forces can be adjusted for the other soil conditions by multiplying the results from the hard rock response spectrum analysis at each elevation by the ratio of the soil case to the hard rock case member forces at that elevation. ~~This assumption is implicit in the development of the 3D stick models from the finite element models.~~

SSAR Revision:

Revise the third paragraph of Subsection 3.7.2.1.1 to read as follows:

Response spectrum analyses are performed only for the hard rock site. ~~, based on the comparison of seismic member forces obtained from the two-dimensional soil-structure interaction analyses presented in Appendix 2A. Table 2A.17 of Appendix 2A shows that the hard rock site condition governs the seismic response forces and moments for the seismic Category I building structures of AP600 (the auxiliary building, the shield building, and the containment internal structures). In addition, s~~ Since the shear wave velocity for the hard rock site is in excess of 8000 feet per second, the soil-structure interaction effect is negligible. Therefore, the response spectrum analyses are performed using the fixed-base, three-dimensional, finite element models.

Question 410.138

Section 10.2.1.2 of the SSAR states that the main turbine system is designed in accordance with applicable interface requirements and system design requirements of the Westinghouse NSSS. What are the "applicable interface requirements" and "system design requirements" of the Westinghouse NSSS? Where are those requirements defined? Are there any specific SSAR sections that can be cross-referenced? Does the term "Westinghouse NSSS" mean the AP600 reactor and reactor control design?

Response:

The statement in the SSAR was intended to indicate that the AP600 turbine is designed specifically for use with the AP600 plant. The design requirements are those indicated in SSAR Chapters 4, 5, 7, 9, and 10.

SSAR Revision:

Revise the first two items in Subsection 10.2.1.2 as follows:

- The turbine-generator is intended for baseload operation and also has load follow capability consistent with the capabilities of the AP600. ~~Westinghouse nuclear steam supply system (NSSS)~~.

- The main turbine system (MTS) is designed for electric power production consistent with the capability of the AP600 reactor coolant system. ~~in accordance with applicable interface requirements and system design requirements of the Westinghouse NSSS~~.

Question 410.203

Include COL information which requires the COL applicant to provide an updated flood analysis incorporating as-built information in the SSAR.

Response:

Flood and ground water elevations are site interfaces for the Combined License applicant. The site interfaces parameters are given in SSAR Section 2.4. For cases where a site characteristic exceeds the envelope parameter, it is the responsibility of the Combined License applicant referencing the AP600 to demonstrate that the site characteristic does not exceed the capability of the design. Thus, it is not necessary or appropriate to include in the design certification of the AP600, requirements and commitments for applicants with sites that do not meet the site characteristics for the standard design.

The AP600 is a standard plant and therefore there is no need to update the flood analysis.

SSAR Revision: NONE

Question 410.209

The March 18, 1993, response to Q410.54 states that protection of safety-related SSCs from failure of non-safety-related SSCs is accomplished by separation, as discussed in Section 3.7.3.13.1 of the SSAR. Section 3.7.3.13 clarifies the approach used to protect safety-related SSCs from the failure of nonseismic SSCs. However, it is still unclear whether protection of safety-related SSCs from nonseismic SSCs is ever achieved through the use of enclosure of safety-related SSCs in compartments. Clarify this issue.

Response:

As described in SSAR Subsection 3.7.3.13, the preferred method for protection of safety-related SSCs is by locating the equipment in rooms (compartments) which are separated from non-seismic components. This separation is indicated on the General Arrangement drawings given in SSAR Section 1.2.

As described in SSAR Subsection 3.7.3.13, each area of the plant containing safety related components is reviewed for potential interactions due to non-seismic structures, systems and components. In performing this review there may be cases where local enclosures provide protection to safety-related equipment. An example of such a situation would be a case where a safety related electrical cable tray is surrounded by a local fire barrier. These cases are evaluated for failures of non-seismic components as an impact analysis following the criteria of SSAR Subsection 3.7.3.13.2.

SSAR Revision: NONE

Question 435.81

The last bullet in Section C17.6.1.2 of the PRA states that "The unavailability data for the dc batteries in the AP600 probabilistic risk assessment data base is assumed to be due to component failure and associated unscheduled corrective maintenance to repair the damaged equipment due to a fault." Table C17-5, Component Test Assumptions, states that "A conservative PRA approach assuming a complete battery test is performed every three months." Clarify these assumptions and how they fit what may be the practice in the AP600 plants of conducting battery service discharge tests at power every two years (refueling cycle interval) on each battery, using the spare battery and charger as a replacement. Describe how battery unavailability due to surveillance testing and maintenance is accounted for in the PRA with regard to both the normal batteries and the spare battery.

Response:

The assumption from Table C17-5 regarding a complete battery test being performed every three months was not intended to imply that such a test is planned every 3 months, but was used to generate the hours (t/2) to use as a multiplier with the hourly failure rate for the batteries failing to provide output on demand. Considering that battery status (cell voltage, battery voltage and battery current) is continuously monitored, a three month interval is overly conservative. The battery test interval will be based on the recommendation of the battery vendor, but is not expected to be any more frequent than the two-year refueling cycle interval.

The assumption from Section C17.6.1.2 that the unavailability data for the dc batteries in the AP600 probabilistic risk assessment data base is assumed to be due to component failure and associated unscheduled corrective maintenance to repair the damaged equipment due to a fault reflects how battery test and maintenance is modeled in the fault trees. When quantified, the fault tree event, which models the unavailability of a battery bank, captures three basic events: a fuse opening spuriously, the battery bank failing to provide output on demand, and the unavailability of the battery bank due to test and unscheduled maintenance. Scheduled maintenance for the battery bank is not explicitly modeled in the fault tree.

As one global test and maintenance unavailability number, 5.0×10^{-4}, was used to quantify the fault tree, the model did not distinguish unavailabilities due to testing from unavailabilities due to maintenance. The data used to quantify test and maintenance unavailability was taken from the Advanced Light Water Reactor Utility Requirements Document, Volume III, Section A2.2, page A.A-19, Revisions 5 & 6. 5.0×10^{-4}, suggested by the URD as a maintenance unavailability for any major component of a standby safety system, represents an unavailability of 4.38 hours/year, per battery bank train.

Neither of these assumptions were meant to explicitly address a spare battery bank or battery testing at power. A spare battery, capable of replacing a battery bank on any one of the four dc buses during a maintenance activity, is not explicitly credited in the fault tree model. While acknowledging that the existence of a spare battery bank improves availability, an unavailability of ~ 4 hours/year appears to be a reasonable unavailability estimate given the time required to accomplish an unplanned swap to the spare battery.

PRA Revision: NONE
SSAR Revision: NONE

Question 440.68

Section 2.3.3.7 of Chapter 1 of the EPRI ALWR Requirements Document states that all operating conditions (including shutdown operations) should be taken into account in the probabilistic risk analysis (PRA). Appendix A to Chapter 1 of the EPRI Requirements Document specifies guidance on the scope and contents of a PRA for shutdown conditions. Address compliance of the AP600 design to this guidance regarding an evaluation to identify system vulnerability for shutdown or mid-loop operation.

Response:

The ALWR Utility Requirements Document does not constitute a regulatory requirement for the AP600. The scope of the AP600 PRA is consistent with the ALWR Utility Requirements Document.

The shutdown portion of the AP600 PRA addresses plant operations, accident scenarios, and system vulnerabilities during hot shutdown through startup conditions. The evaluation searches for system vulnerability for shutdown or midloop operation.

The shutdown evaluation examines system and operator performances during various postulated initiating events including loss of decay heat removal (normal residual heat removal, component cooling water, or service water), loss of offsite power, loss of coolant accident (normal residual heat removal pipe break or inadvertent drains), and reactivity accidents (boron dilution or rod withdrawal).

The PRA shutdown model reflects the success criteria of available systems during shutdown or midloop operation to determine dominant contributors to the AP600 core damage frequency and fission product release.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.111

Section 6.3.3.3.2 of the SSAR defines a loss of coolant accident as a rupture of the RCS piping that results in a decrease in the RCS inventory that exceeds the flow capability of the normal makeup system. However, because the AP600 normal makeup system is a non-safety-related system, credit for its makeup capability should not be taken to compensate for the loss of coolant. Appendix K to 10 CFR Part 50 requires consideration of a spectrum of possible pipe breaks. Either confirm that the small break LOCA analysis is extended to break sizes within normal makeup capability, i.e., less than 0.375-inch diameter hole, or provide justification (other than makeup capability of non-safety systems) for not evaluating this small break size.

Response:

The effect of postulated break size in the SSAR LOCA analysis is such that the smaller the break size the greater the minimum reactor coolant system (RCS) inventory condition. This is illustrated in the table below:

SSAR Break Equivalent Diameter, inches	Minimum Mass Inveatory, lbm
DEDVI (6.8 inches)	90000 (core uncovers)
Inadvertent ADS	100000 (no core uncovery)
Two-inch cold leg	105000 (no core uncovery)
One-inch cold leg	115000 (no core uncovery)

Higher minimum mass inventories provide greater margin to possible core uncovery conditions. Extending the SSAR break spectrum to even smaller sizes will produce less limiting results than the one inch break exhibits.

Small break LOCA cases performed subsequent to the SSAR in support of design changes (References 440.111-1, 440.111-2) further demonstrate that the smaller size LOCA breaks are non-limiting for the AP600, with its passive safety systems.

The one-inch cold leg break has been reanalyzed in support of the design change to remove the pressurizer/core makeup tank pressure balance lines. The minimum RCS inventory computed for this case is reported in Reference 440.111-2) as 107,000 lbs. The double-ended pressure balance line break reported in Reference 440.111-2 involves a 7.001 inch equivalent diameter break and exhibits a minimum inventory of 104,000 lbs. Therefore, over a range of primary reactor coolant break sizes (the reanalyzed double-ended pressure balance line, DEDVI and one-inch cold leg break cases) involving a factor of fifty change in the postulated break area, predicted minimum primary mass inventories vary only slightly. Adequate minimum RCS inventories are maintained to ensure that no core uncovery occurs, and the one inch break exhibits the largest minimum mass inventory among the three cases. The observed behavior shows that there is no need to extend the AP600 LOCA break spectrum to break sizes smaller than one inch equivalent diameter.

References:

440.111-1 AP600 Design Change Description Report, February 15, 1994, Westinghouse letter NTD-NRC-94-4064.

440.111-2 AP600 Design Change Description Report, June 30, 1994, Westinghouse letter NTD-NRC-94-4175

SSAR Revision: None
PRA Revision: None

Question 440.165

GSI 125.11.7 addresses the need for plant owners to assess the benefit of automatic isolation of the emergency feedwater (EFW) system after a secondary line break against the potential disadvantages of automatic isolation of the EFW where the secondary heat sink may be lost if the EFW is lost and the main steam isolation valve is closed. From the regulatory analysis, the staff determined that, for a new plant, the design need not include automatic isolation of the EFW system following a steamline or feedwater line break provided that the results of the analyses of the secondary side line break and the containment analysis meet the applicable design criteria. For the AP600 design, the startup feedwater (SFW) control valves (SFCV) serve the dual purpose of controlling SFW flow rate and providing isolation of the SFW. The SFW isolation valve (SFIV) is used to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater rupture.

a. Clarify whether the isolation of the SFW in the event of a secondary line break is automatic or manual.

b. Provide the evaluation of the automatic isolation of the SFW with respect to the concern of GSI.125.11.7.

c. Confirm that automatic isolation of the SFW is not assumed in the analyses of a feedwater line break (Section 15.1.2 of the SSAR) and a steam system piping failure (Section 15.1.5 of the SSAR), and the mass and energy release analysis for postulated secondary system pipe rupture inside containment (Section 6.2.1.4 of the SSAR). In addition, NRC IE Bulletin 80-04 states that the analyses of a steamline break and containment overpressure event should include an assumption of continued addition of startup feedwater. Confirm that this assumption is made in these analyses.

Response:

a. Isolation of the startup feedwater system is automatically initiated. SSAR Subsections 7.3.1.1.3.4 and 7.3.1.1.4.4 discuss the "Engineered Safety Features" isolation logic for the startup feedwater system. The isolation logic is illustrated on Figure 7.2-1 sheets 2, 10, and 11.

b. Table 1.9-2 of the SSAR "Listing of Unresolved Safety Issues and Generic Safety Issues" defines a screening criteria on the applicability of the GSI and USI issues to the AP600 design. On generic issue 125.II.7, the status screening criteria provided is "c" indicating the issue is resolved with no new requirements imposed.

   The AP600 utilizes a non-safety related startup feedwater system as the first line of defense to remove the core decay heat after a reactor trip or during a postulated non-LOCA event. The startup feedwater pumps automatically start following anticipated transients resulting in a low steam generator level. Startup feedwater will continue to be delivered to the steam generators unless excessive SG levels or excessive primary system heat removal conditions develop. However, operation of the non-safety related startup feedwater system is not credited or required to mitigate licensing design basis accidents. The safety-related passive core cooling system (PXS) provides emergency core decay heat removal during transients, accidents, or whenever the normal nonsafety-related heat removal paths are unavailable. Concern over the loss of the ultimate heat sink as a result

of isolating feedwater is not pertinent to the AP600 given the availability and capability of the passive residual heat removal feature of the PXS. The safety-related passive core cooling system design basis and criteria are described in SSAR Section 6.3.

c. Section 15.2.8 of the SSAR is a feedwater system pipe break. This event is a reactor coolant system heatup transient and therefor heat removal via startup feedwater is not credited in accident mitigation.

Section 15.1.2 is a feedwater system malfunction that results in a step increase in feedwater flow to one steam generator from 0 to 115% of the nominal full load value. All feedwater flow is terminated by a steam generator high level trip.

Section 15.1.5 evaluates non-containment aspects of a steam system piping failure. Startup feedwater is isolated via closure of the redundant safety-related isolation and control valves on a low T_{cold} signal.

Section 6.2.1.4 evaluates the containment aspects of a steam system piping failure. Within the first minute following a steam line break, the startup feedwater system may be initiated on any one of several signals. The addition of startup feedwater to the steam generators increases the secondary mass available for release to the containment, as well as the heat transferred to the secondary fluid. The effects on the steam generator mass are maximized by assuming full startup feedwater flow to the faulted steam generator starting at time zero from the safeguard system(s) signal or low steam generator level reactor trip and continuing until automatically terminated on a low T cold signal. Startup feedwater flow termination is accomplished by closure of the redundant safety-related isolation and control valves. No additional startup feedwater is assumed following closure of the redundant safety-related isolation and control valve.

SSAR Revision: None

PRA Revision: None

Question 480.58

Note 6 of Table 6.2.3-1 of the SSAR states that airlock seal testing will be done at reduced pressure. A test pressure lower than Pa would be contrary to Appendix J criteria. Provide additional basis to support this position. This exception should be added to Table 6.2.5-1.

Response:

The personnel hatches (airlocks) are designed to be tested by internal pressurization. The doors of the personnel hatches have testable seals as shown in SSAR Figure 3.8.2-3. Mechanical and electrical penetrations on the personnel hatches are also equipped with testable seals. Appendix J testing will be performed at a test pressure of P_a. Table 6.2.3-1 will be revised to reflect the correction.

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

Question 480.61

Figure 9.2.7-1 of the SSAR indicates that the chilled water return penetration has 10-inch isolation valves. Table 6.2.3-1 of the SSAR indicates that this penetration uses 6-inch valves. Clarify the valve size.

Response:

Figure 9.2.7-1 is correct in that the chilled water return line utilizes 10 inch isolation valves. The revision to Table 6.2.3-1 of the SSAR is attached reflecting the correct chilled water return line size and additional changes in response to RAIs 480.52, 480.53, 480.55, 480.57, 480.58, 480.59 and 480.60.

SSAR Revision:

Table 6.2.3-1 attached

Question 480.70

This question pertains to Westinghouse's statement of conformance to paragraph 6.2.1.5 of the Standard Review Plan, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," BTP CSB 6-1, that is identified on page 6-10 of Revision 1 to WCAP-13054, "AP600 Compliance with SRP Acceptance Criteria."

Provide the heat transfer coefficients used to address this criteria.

Response:

The large break LOCA ECCS performance analysis presented in the AP600 SSAR utilizes an assumed containment pressure of 14.7 psia. The WGOTHIC analyses of the AP600 presented in Reference 480.70-1 demonstrate that this assumed pressure is a highly conservative minimum value. Thus, no heat transfer coefficients to the structures inside containment are calculated in defining the AP600 minimum containment pressure in the SSAR.

Future AP600 large break LOCA ECCS performance analysis will utilize a suitably conservative containment pressure boundary condition.

Reference

480.70-1 Westinghouse letter NTD-NRC-94-4174, N. J. Liparulo to R. W. Borchardt, dated June 30, 1994

SSAR Revision: NONE

PRA Revision: NONE

Question 720.272

Q720.175 requested Westinghouse to show how maintenance unavailabilities were included in the shutdown PRA. The October 20, 1993 response states that the PRA credits both safety- and non-safety-related systems, as specified in the Technical Specifications. However, the Technical Specifications will not prevent a licensee from entering into the LCOs. The response referenced Appendix C of the PRA for specific maintenance unavailabilities. The staff could not find maintenance unavailabilities for dc power in Table C17-6. Table C13-8 of Appendix C states that the PRA modeled the NRHR pump as being maintained once every five years. The staff believes that this value is unrealistic and that Westinghouse should include reasonable maintenance unavailability estimates in the PRA based on operating experience. If unreasonable maintenance estimates are used, then the actual shutdown core damage risk incurred by a COL holder will be higher than estimated in the PRA because the system availabilities have been significantly underestimated. Address this concern.

Response:

The assumptions on maintenance unavailability for dc power (batteries) are addressed in RAI 435.81 for at-power operation. The response provided for RAI 435.81 applies also to shutdown conditions.

The information provided in Table C13-8 is inaccurate. Typically, the normal residual heat removal pumps will be maintained during at-power operation. It is conservatively assumed in the PRA that the normal residual heat removal pumps are scheduled to be maintained every 6 months. It is also assumed in the PRA that the normal residual heat removal valves are scheduled to be maintained once every refueling cycle, as shown in Table C13-6. This correction will be reflected in revision 2 of the PRA.

SSAR Revision: NONE

PRA Revision:

The documentation of revision 2 of the PRA will correct the error in Table C13-8, as shown in the response above. PRA Revision 2 will be completed by December 31, 1994.

Question 720.273

Q720.175 requested Westinghouse to include loss of the NRHR and loss of the NRHR support systems as shutdown initiators. The October 20, 1993 response indicates that the loss of offsite power was the only way that the NRHR was postulated to fail, causing a loss of decay heat removal during normal and reduced inventory conditions. Include the loss of NRHR and the loss of NRHR support systems as potential shutdown initiators. When considering these initiators, include system maintenance and the extended mission times for which they must operate. These initiators should be included in the loss of decay heat removal event trees.

Response:

The current AP600 shutdown event trees include loss of normal residual heat removal and its support systems as well as loss of offsite power. These initators are combined in one event tree. Westinghouse agrees that this event tree should be reconstructed as suggested. An event tree will be developed for the loss of offsite power initiator, and a separate event tree will be developed for the loss of normal residual heat removal as an initiator. These initiators will be reflected in trees for conditions when the reactor coolant system is filled and pressurized, and when the reactor coolant system is drained to mid-loop and depressurized.

For the normal residual heat removal and its support systems, there will be no planned maintenance during shutdown. Scheduled maintenance will be done during at-power operation. Therefore, both trains of normal residual heat removal will be available when entering shutdown conditions. If one train of normal residual heat removal is lost during nondrained, cold shutdown conditions, the plant will be kept in the nondrained, cold shutdown condition and normal residual heat removal capability will be restored. If one train of normal residual heat removal is lost during drained conditions, the plant must be taken from drained condition to the depressurized but filled condition and normal residual heat removal capability restored.

Based on the operational requirements of the normal residual heat removal system discussed above, Westinghouse does not believe that the loss of normal residual heat removal initiator should include maintenance and extended mission times of the normal residual heat removal and its support systems.

SSAR Revision: NONE

PRA Revision:

Event trees CSD and CSLD will be reconstructed to address the loss of offsite power and loss of NRHR initiators in separate event trees. Assumptions relative to planned maintenance will be documented in Revision 2 of the PRA. Revision 2 will be completed by December 31, 1994.

Question 720.274

In Q720.178, the staff requested that a quantitative basis for excluding overdraining events be included in the shutdown PRA during reduced inventory conditions. The October 20, 1993 response referenced Appendix F.4.3 of the PRA. The staff believes that this response is insufficient to address the staff's concerns, and that Westinghouse should develop an event tree that includes overdraining of the reactor vessel during reduced inventory conditions. This event tree should include operator recovery. This event tree should also consider the adequacy of core cooling given that the hot leg is nearly or completely drained, and that the NRHR pumps continue to run. Westinghouse should consider that the hot leg level instrumentation provides input to the non-safety-related plant control system and provides input to the diverse actuation system, which will not be in Technical Specifications and could be out of service for maintenance.

Response:

Revision 0 of the PRA includes the quantitative evaluation of overdraining events due to loss of coolant accident from normal residual heat removal system pipe failure or human error creating a diversion path to the IRWST.

Revision 2 of the PRA will also evaluate the scenario where the hot leg is drained and the operator does not stop the normal residual heat removal system pumps. The basic model for this scenario will cover the following: a) draining down the reactor coolant system; b) operator fails to stop the normal residual heat removal system pumps; c) normal residual heat removal system pump seal LOCA occurs; and d) operator fails to isolate the normal residual heat removal system.

Revision 2 of the PRA will address the availability of the hot leg level instrumentation.

SSAR Revision: NONE

PRA Revision:

Revision 2 of the Shutdown PRA will evaluate core cooling adequacy based on the following sequence of events:

a) Drain down the reactor coolant system
b) Operator fails to stop the normal residual heat removal system pumps
c) Normal residual heat removal system pump seal LOCA occurs
d) Operator fails to isolate the normal residual heat removal system.

Revision 2 of the PRA will address the availability of the hot leg level instrumentation.

Revision 2 will be completed by December 31, 1994.

Question 720.278

The September 21, 1993, response to Q720.65 indicated that the time window was 1 minute and the operator response time took 30 seconds for the operator to manually trip the reactor following an ATWS event. The failure rate, HEP=1.36E-2, was indicated for five multiple actions that are to be taken in less than 1 minute. The crew's stress level was modelled as "moderate" instead of "high," which conflicts with HRA procedures in the PRA Guidebook (WCAP-12699). If the crew realize that they have only one minute to take these actions, the crew's stress level would arguably be "high" instead of "moderate." If the crew is distracted or interrupted by events in the control room, the margin (residual time) could be reduced from 30 seconds to 15 seconds. Re-calculate the HEP for this operator action taking these concerns into account.

Response:

The modeling of operator action ATW-MAN03 consists of the following 3 subtasks:

a) Respond to 3 alarms

b) Verify neutron flux increasing

c) Scram the reactor.

Subtasks (b) and (c) can each fail as a result of error of omission or error of commission. For example, the modeling of subtask (b) is shown as failure of item 2 or 3 in the response to RAI 720.65; subtask (c) is shown as failure of item 4 or 5 of the model shown in the response to RAI 720.65 . Therefore, 5 possible ways of failing the action to trip the reactor are shown in the model.

Westinghouse agrees that the stress level used in the evaluation of this action should be re-calculated with a high stress level.

In performing HRA for several Individual Plant Examinations (IPEs), training and operating personnel from seven plants were interviewed to determine the performance shaping factors during each accident sequence. It was determined that tripping the reactor is one of the most highly trained scenario for the operators, and it is classified as a skilled-based action, whereby the operator response is viewed as being second nature. Despite the 1-minute time window, this task is viewed by the operators as the most unlikely action to fail. The HEP evaluated for this action in the IPEs is on the order of 2.0E-03. A similar level of training will be provided to the AP600 control room operators to repond to this action.

Re-calculating this operator action for the AP600 with a high stress level will change the HEP from 1.56E-02 to 3.9E-02. This value is somewhat conservative, because redundant cues and scram controls, assumed to be available to the operators, are not considered in the AP600 model.

PRA Revision:

Re-calculation of the HEP for operator action ATW-MAN03 will be reflected in Revision 2 of the AP600 PRA. Revision 2 will be completed by December 31, 1994.

AP600

Table 6.2.3-1 (Sheet 1 of 4)

Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration							Isolation Device										Test		
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg	Closed Sys IRC	Qty	Size	Type	Operator	Location	Position N-S-A	Signal	Actuation Mode: Primary	Actuation Mode: Secondary	Closure Times	Type[1] & Note	Medium	Direction
[illegible]	P01	[illegible]	Breathing air in	In	2"	56	No	1 1	2" 2"	Globe Check	Manual	ORC IRC	C-O-C C-O-C	None None	Manual Self	None None	N/A N/A	C	Air	Forward
	P02	C02	Service air in	In	6"	56	No	1 1	6" 6"	Globe Check	Air	ORC IRC	O-O-C O-O-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
CCS	P03	C01	IRC loads in	In	8"	56	No	1 1	8" 8"	Gate Gate	Motor Motor	ORC IRC	O-O-C O-O-C	S S	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	P04	C02	IRC loads out	Out	8"	56	No	1 1 1	8" 8" 3/4"	Gate Gate Check	Motor Motor	ORC IRC IRC	O-O-C O-O-C C-C-C	S S None	Automatic Automatic Self	Remote Manual Remote Manual None	std. std.	C	Air	Forward
CVS	P05	C01	Spent resin flush out	Out	2"	55	No	1 1 1	2" 2" 3/4"	Ball Ball Relief	Manual Manual –	ORC IRC IRC	C-C-C C-C-C C-C-C	None None None	Manual Manual Self	None None None	N/A N/A N/A	C	Air	Forward
	P06	C02	Letdown	Out	2"	55	No	1 1 1	3/4" 2" 2"	Relief Globe Globe	Manual – Air Air	IRC ORC IRC	C-C-C O-O-C O-O-C	None T T	Self Automatic Automatic	None Remote Manual Remote Manual	N/A std. std.	C	Air	Forward
	P07	C03	Charging	In	3"	55	No	1 1	3" 3"	Globe Globe	Motor Motor	ORC IRC	O-O-C O-O-C	TLP TLP	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	P08	C04	H_2 injection to RCS	In	3/4"	55	No	1 1	3/4" 3/4"	Globe Check	Air –	ORC IRC	O-C-C O-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
	P09	C10	Water to CMT and accumulators	In	2"	55	No	1 1	2" 2"	Globe Check	Air –	ORC IRC	C-C-C C-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
DWS	P10	C01	Demin. water sys	In	8 2"	56	No	1 1	2" 2"	Globe Check	Manual –	ORC IRC	C-O-C C-O-C	‡None None	Manual Self	None None	N/A N/A	C	Air	Forward
FHS	P11		Fuel transfer	N/A	36"	56	No	1	36"	Blind Flange	N/A	IRC	C-O-C	None	N/A	N/A	N/A N/A	B	Air	Forward
FPS	P12	C01	Fire protection standpipe sys.	In	4"	56	No	1 1	4" 4"	Gate Check	Manual –	ORC IRC	C-O-C C-C-C	None None	Manual Self	None None	N/A N/A	C	Air	Forward

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Table 6.2.3-1 (Sheet 2 of 4)

Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration: Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Isolation Device: Qty	Size	Type	Operator	Location	Position N-S-A	Signal	Actuation Mode: Primary	Actuation Mode: Secondary	Closure Times	Test: Type[1] & Note	Medium	Direction
PCS	P13	CD1	Cont. pressure	N/A	3/4"	RG1.141 RG1.11	Yes	1 1	3/4"	Globe Bellows	Manual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
	P14	CD2		N/A	3/4"	RG1.141 RG1.11	Yes	1 1	3/4"	Globe Bellows	Manual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
	P15	CD3		N/A	3/4"	RG1.141 RG1.11	Yes	1 1	3/4"	Globe Bellows	Manual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
	P16	CD4		N/A	3/4"	RG1.141 RG1.11	Yes	1 1	3/4"	Globe Bellows	Manual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
PSS	P17	CD1	RCS/PSX/CVS samples out	Out	3/8"	55	No	1 2	3/8" 3/8"	Globe Globe	Solenoid Solenoid	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
		CD2	Cont. air samples out	Out	3/8"	56	No	1 2	3/8" 3/8"	Globe Globe	Solenoid Solenoid	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C4	Air	Forward
		CD3	RCS/Cont. air sample return	In	1"	56	No	1 1	1" 1"	Globe Check	Solenoid –	ORC IRC	C-C-C C-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
			Spare	N/A	3/8"	56	No	1 1	3/8" 3/8"	Cap Cap	N/A N/A	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A N/A	A	Air	Forward
PXS	P18	CD1	N_2 to accumulators	In	1"	55	No	1 1	1" 1"	Globe Check	Air –	ORC IRC	C-O-C [illegible]	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
RNS	P19	CD1	RCS to RHR pump	Out	10"	55	No	2 1 1 1 1 1	10" 10" 10" 3" 3" 3/8"	Gate Gate Gate Relief Gate Globe	Motor Motor Motor Self Manual Manual	IRC IRC ORC IRC IRC IRC	C-O-C O-O-C C-O-C C-C-C C-C-C C-C-C	None HR None None None None	Remote Manual Remote Manual Remote Manual Self Manual Manual	None None Manual None None None	std. std. std. N/A N/A N/A	C4	H_2O	Forward
	P20	CD2	RHR pump to RCS	In	8"	55	No	1 1	8" 8"	Globe Check	Motor –	ORC IRC	C-O-C C-O-C	None None	Remote Manual Self	None None	std. N/A	C4	H_2O	Forward
SFS	P21	CD1	IRWST/Ref. cav. SFP pump discharge	In	4"	56	No	1 1	4" 4"	Gate Check	Motor –	ORC IRC	C-O-C C-O-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
	P22	CD2	IRWST/Ref. cav. purif. out	Out	4"	56	No	1 1 1	6" 6" 3/4"	Gate Gate Check	Motor Motor –	ORC IRC IRC	C-O-C C-O-C C-C-C	T T None	Automatic Automatic Self	Remote Manual Remote Manual None	std. std. N/A	C	Air	Forward

Table 6.2.3-1 (Sheet 3 of 4)

Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration							Isolation Device										Test		
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Qty	Size	Type	Operator	Location	Position N-S-A	Signal	Activation Mode: Primary	Activation Mode: Secondary	Closure Times	Type[1] & Note	Medium	Direction
SGS	P23	CD1A	Main steamline 01	Out	32"	57	Yes	1 1 3 1 1	32" 6" 8" 2" 3"	Gate Gate Safety Globe Globe	Pneumatic Motor — Air Air	ORC ORC ORC ORC ORC	O-C-C O-O-C C-C-C C-C-C C-C-C	MS LSL None MS MS	Automatic Automatic Manual Automatic Automatic	Remote Manual Remote Manual None Remote Manual Remote Manual	5 sec 5 sec N/A std. std.	A,2	N_2	Forward
	P24	CD1B	Main steamline 02	Out	32"	57	Yes	1 1 3 1 1	32" 6" 8" 2" 3"	Gate Gate Safety Globe Globe	Pneumatic Motor — Air Air	ORC ORC ORC ORC ORC	O-C-C O-O-C C-C-C C-C-C C-C-C	MS LSL None MS MS	Automatic Automatic Manual Automatic Automatic	Remote Manual Remote Manual None Remote Manual Remote Manual	5 sec 5 sec N/A std. std.	A,2	N_2	Forward
	P25	CD2A	Main and startup feedwater 01	In	16"	57	Yes	1 1	16" 4"	Gate Globe	Pneumatic Air	ORC ORC	O-C-C C-O-C	MF LTC	Automatic Automatic	Remote Manual Remote Manual	5 sec std.	A,2	H_2O	Forward
	P26	CD2B	Main and startup feedwater 02	In	16"	57	Yes	1 1	16" 4"	Gate Globe	Pneumatic Air	ORC ORC	O-C-C C-O-C	MF LTC	Automatic Automatic	Remote Manual Remote Manual	5 sec std.	A,2	H_2O	Forward
	P27	CD3A	SG blowdown 01	Out	4"	57	Yes	1	4"	Globe	Air	ORC	O-O-C	PRHR	Automatic	Remote Manual	std.	A,2	H_2O	Forward
	P28	CD3B	SG blowdown 02	Out	4"	57	Yes	1	4"	Globe	Air	ORC	O-O-C	PRHR	Automatic	Remote Manual	std.	A,2	H_2O	Forward
	P29	CD4	SG blowdown recirculation	In	3"	57	Yes	1	3"	Globe	Manual	ORC	C-O-C	None	Manual	None	N/A	A,2	H_2O	Forward
VFS	P30	CD1A	Cont. air filter supply A	In	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C,4	Air	Forward
	P31	CD1B	Cont. air filter supply B	In	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C	Air	Forward
	P32	CD2A	Cont. air filter exhaust A	Out	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C	Air	Forward
	P33	CD2B	Cont. air filter exhaust B	Out	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C	Air	Forward
VWS	P34	CD1	Fan Coolers out	Out	10"	56	No	1 1	& 10" & 10"	Butterfly Butterfly	Pneumatic Pneumatic	ORC IRC	O-O-C O-O-C	T,CP T,CP	Automatic Automatic	Remote Manual Remote Manual	std. std.	C,3,4	Air	Forward
	P35	CD2	Fan coolers in	In	10"	56	No	1 1	10" 10"	Butterfly Butterfly	Pneumatic Pneumatic	ORC IRC	O-O-C O-O-C	T,CP T,CP	Automatic Automatic	Remote Manual Remote Manual	std. std.	C,3,4	Air	Forward

Table 6.2.3-1 (Sheet 4 of 4)

Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration							Isolation Device										Test		
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Qty	Size	Type	Operator	Location	Position N-S-A	Signal	Actuation Mode: Primary	Actuation Mode: Secondary	Closure Times	Type[1] & Note	Medium	Direction
WLS	P36	CD1	Reactor coolant drain tank out	Out	2"	56	No	1 1	2" 2"	Globe Globe	Air Air	ORC IRC	O-O-C O-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	P37	CD2	Reactor coolant drain tank gas	Both	3/4"	56	No	1 1	3/4" 3/4"	Globe Globe	Air Air	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	P38	CD3	Normal cont. sump	Out	1 1/2"	56	No	1 1	1 1/2" 1 1/2"	Globe Globe	Air Air	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
SPARE	P39			N/A	12"	56	No	1 1	12" 12"	Flange Flange	N/A N/A	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A	B	Air	Forward
SPARE	P40			N/A	12"	56	No	1 1	12"	Flange Flange	N/A N/A	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A	B	Air	Forward
SPARE	P41			N/A	12"	56	No	1 1	12"	Flange Flange	N/A N/A	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A	B	Air	Forward
SPARE	P42			N/A	12"	56	No	1 1	12"	Flange Flange	N/A N/A	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A	B	Air	Forward
SPARE	P43			N/A	12"	56	No	1 1	12"	Flange Flange	N/A N/A	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A	B	Air	Forward
CNS	H01	N/A	Main equipment hatch	N/A	264"	56	No	1		Double Sealed Hatch		IRC	C-C-C	None	Manual	None	N/A	B	Air	Forward
	H02	N/A	Maintenance hatch	N/A	192"	56	No	1		Double Sealed Hatch		IRC	C-C-C	None	Manual	None	N/A	B	Air	Forward
	H03	N/A	Personnel hatch	N/A	118"	56	No	1		Double Sealed Hatch		IRC	C-C-C	None	Manual	None	N/A	B,6	Air	Forward
	H04	N/A	Personnel hatch	N/A	118"	56	No	1		Double Sealed Hatch		IRC	C-C-C	None	Manual	None	N/A	B,6	Air	Forward

Table 6.2.3-1

Containment Mechanical Penetrations and Isolation Valves

Explanation of Heading and Acronyms for Table 6.2.3-1

System: Fluid system penetrating containment

Containment Penetration: These fields refer to the penetration itself
Penetration Sleeve I.D.: Actual penetration identification number
P&ID Sequence No.: Penetration identification number used on the P&IDs
Line: Fluid system line
Flow: Direction of flow in or out of containment
Size.: Line size
GDC or RG: Applicable general design criteria or Regulatory Guide
Closed Sys IRC: Closed system inside containment as defined in SSAR Section 6.2.3.1.1

Isolation Device: These fields refer to the isolation devices for a given penetration
Qty.: Number of subject devices per penetration
Size: Device size
Type: Device body type
Operator: Operator type (for valves)
Location: Device location inside or outside containment
Position N-S-A: Device position for N (normal operation)
S (shutdown)
A (post-accident)

Signal: Device closure signal
MS: Main steamline isolation
LSL: Low steamline pressure
MF: Main feedwater isolation
LTCA: Low Tcoldavg
PRHR: Passive residual heat removal actuation
T: Containment isolation
RCP: Reactor Coolant Pump auto trip signal
CP: High containment pressure
TLP: Containment isolation coincident with low header pressure
S: Safety Injection Signal

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 automatically closes 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)

Closure Time: Required valve closure stroke time
STD: Industry standard for valve type
N/A: Not Applicable

Test: These fields refer to the penetration testing requirements

Type: Required test type
A: Integrated Leak Rate Test
B: Local Leak Rate Test -- penetration
C: Local Leak Rate Test -- fluid systems

Note: See notes below
Medium: Test fluid on valve seat
Direction: Pressurization direction
Forward: high pressure on containment side
Reverse: high pressure on outboard side

Notes:

1. Containment leak rate tests are designated Type A, B, or C according to 10CFR50 Appendix J.

2. The secondary side of the steam generator, including main steam, feedwater, startup feedwater, blowdown and sampling piping from the steam generators to the containment penetration, is considered an extension of the containment. These systems are not part of the reactor coolant pressure boundary and do not open directly to the containment atmosphere during post-accident conditions. During type A tests, the secondary side of the steam generators is ~~drained and~~ vented to the atmosphere outside containment to ensure that full test differential pressure is applied to this boundary.

3. The central chilled water system remains water-filled and operational during the Type A test in order to maintain stable containment atmospheric conditions.

4. The containment isolation valves for this penetration are open during the Type A test to facilitate testing. Their leak rates are measured separately.

5. The inboard butterfly valve is tested in the reverse direction.

6. ~~Testing of the double seals on the personnel hatch doors is performed at a reduced pressure of [LATER] psig. Testing at a lower pressure is conservative since the test pressure tends to open the doors, whereas containment pressure would hold the doors closed.~~

7. The inboard globe valve is tested in the reverse direction. The test is conservative since the test pressure tends to unseat the valve disc, whereas containment pressure would tend to seat the disc.

Table 6.2.3-1 (

Containment Mechanical Pene

System	Containment Penetration											
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Qty	Size	Type	Operator
CAS	P01	C01	Breathing air in	In	2"	56	No	1 1	2" 2"	Globe Check	Manual
	P02	C02	Service air in	In	6"	56	No	1 1	6" 6"	Globe Check	Ai
CCS	P03	C01	IRC loads in	In	8"	56	No	1 1	8" 8"	Gate Gate	Motor Motor
	P04	C02	IRC loads out	Out	8"	56	No	1 1 1	8" 8" 3/4"	Gate Gate Check	Motor Motor
CVS	P05	C01	Spent resin flush out	Out	2"	55	No	1 1 1	2" 2" 3/4"	Ball Ball Relief	Manual Manual --
	P06	C02	Letdown	Out	2"	55	No	1 1 1	3/4" 2" 2"	Relief Globe Globe	~~Manual~~ -- Air Air
	P07	C03	Charging	In	3"	55	No	1 1	3" 3"	Globe Globe	Motor Motor
	P08	C04	H_2 injection to RCS	In	3/4"	55	No	1 1	3/4" 3/4"	Globe Check	Air --
	P09	C10	Water to CMT and accumulators	In	2"	55	No	1 1	2" 2"	Globe Check	Air --
DWS	P10	C01	Demin. water sys	In	~~8~~ 2"	56	No	1 1	2" 2"	Globe Check	Manual --
FHS	P11		Fuel transfer	N/A	36"	56	No	1	36"	Blind Flange	N/A
FPS	P12	C01	Fire protection standpipe sys.	In	4"	56	No	1 1	4" 4"	Gate Check	Manual --

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trations and Isolation Valves

Isolation Device						Test		
Location	Position N-S-A	Signal	Actuation Mode: Primary	Actuation Mode: Secondary	Closure Times	Type[1] & Note	Medium	Direction
ORC IRC	C-O-C C-O-C	None None	Manual Self	None None	N/A N/A	C	Air	Forward
ORC IRC	O-O-C O-O-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
ORC IRC	O-O-C O-O-C	S S	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
ORC IRC IRC	O-O-C O-O-C C-C-C	S S None	Automatic Automatic Self	Remote Manual Remote Manual None	std. std.	C	Air	Forward
ORC IRC IRC	C-C-C C-C-C C-C-C	None None None	Manual Manual Self	None None None	N/A N/A N/A	C	Air	Forward
IRC ORC IRC	C-C-C O-O-C O-O-C	None T T	Self Automatic Automatic	None Remote Manual Remote Manual	N/A std. std.	C	Air	Forward
ORC IRC	O-O-C O-O-C	TLP TLP	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
ORC IRC	O-C-C O-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
ORC IRC	C-C-C C-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
ORC IRC	C-O-C C-O-C	~~T~~None None	Manual Self	None None	N/A N/A	C	Air	Forward
IRC	C-O-C	None	N/A	N/A	N/A N/A	B	Air	Forward
ORC IRC	C-O-C C-C-C	None None	Manual Self	[illegible]	N/A N/A	C	Air	Forward

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Table 6.2.3-1 (S

Containment Mechanical Pene

System	Containment Penetration										
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Qty	Size	Type	Op
PCS	P13	C01	Cont. pressure	N/A	3/4"	RG1.141 ~~RG1.11~~	Yes	1 1	3/4"	Globe Bellows	Man
	P14	C02		N/A	3/4"	RG1.141 ~~RG1.11~~	Yes	1 1	3/4"	Globe Bellows	Man
	P15	C03		N/A	3/4"	RG1.141 ~~RG1.11~~	Yes	1 1	3/4"	Globe Bellows	Man
	P16	C04		N/A	3/4"	RG1.141 ~~RG1.11~~	Yes	1 1	3/4"	Globe Bellows	Man
PSS	P17	C01	RCS/PSX/CVS samples out	Out	3/8"	55	No	1 2	3/8" 3/8"	Globe Globe	Sole Sole
		C02	Cont. air samples out	Out	3/8"	56	No	1 2	3/8" 3/8"	Globe Globe	Sole Sole
		C03	RCS/Cont. air sample return	In	1"	56	No	1 1	1" 1"	Globe Check	Sole --
			Spare	N/A	3/8"	56	No	1 1	3/8" 3/8"	Cap Cap	N/A N/A
PXS	P18	C01	N_2 to accumulators	In	1"	55	No	1 1	1" 1"	Globe Check	Air --
RNS	P19	C01	RCS to RHR pump	Out	10"	55	No	2 1 1 1 1 1	10" 10" 10" 3" 3" 3/8"	Gate Gate Gate Relief Gate Globe	Moto Moto Moto Self Manu Manu
	P20	C02	RHR pump to RCS	In	8"	55	No	1 1	8" 8"	Globe Check	Moto --
SFS	P21	C01	IRWST/Ref. cav. SFP pump discharge	In	4"	56	No	1 1	4" 4"	Gate Check	Moto --
	P22	C02	IRWST/Ref. cav. purif. out	Out	4"	56	No	1 1 1	6" 6" 3/4"	Gate Gate Check	Moto Moto --

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trations and Isolation Valves

	Isolation Device						Test		
erator	Location	Position N-S-A	Signal	Actuation Mode		Closure Times	Type[1] & Note	Medium	Direction
				Primary	Secondary				
ual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
ual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
ual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
ual	ORC N/A	O-O-O C-C-C	None N/A	Manual None	None	N/A N/A	A	Air	Forward
noid noid	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
noid noid	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C,4	Air	Forward
oid	ORC IRC	C-C-C C-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
	ORC IRC	C-C-C C-C-C	N/A N/A	N/A N/A	N/A N/A	N/A N/A	A	Air	Forward
	ORC IRC	O-O-C C-C-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
    al al	IRC IRC ORC IRC IRC IRC	C-O-C O-O-C C-O-C C-C-C C-C-C C-C-C	None HR None None None None	Remote Manual Remote Manual Remote Manual Self Manual Manual	None None Manual None None None	std. std. std. N/A N/A N/A	C,4	H_20	Forward
	ORC IRC	C-O-C C-O-C	None None	Remote Manual Self	None None	std. N/A	C,4	H_20	Forward
	ORC IRC	C-O-C C-O-C	T None	Automatic Self	Remote Manual None	std. N/A	C	Air	Forward
	ORC IRC IRC	C-O-C C-O-C C-C-C	T T None	Automatic Automatic Self	Remote Manual Remote Manual Note	std. std. N/A	C	Air	Forward

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Table 6.2.3-1 (S

Containment Mechanical Pene

System	Containment Penetration										
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Qty	Size	Type	Op
SGS	P23	C01A	Main steamline 01	Out	32"	57	Yes	1 1 3 1 1	32" 6" 8" 2" 3"	Gate Gate Safety Globe Globe	Pneu Mot -- Air Air
	P24	C01B	Main steamline 02	Out	32"	57	Yes	1 1 3 1 1	32" 6" 8" 2" 3"	Gate Gate Safety Globe Globe	Pneu Mot -- Air Air
	P25	C02A	Main and startup feedwater 01	In	16"	57	Yes	1 1	16" 4"	Gate Globe	Pneu Air
	P26	C02B	Main and startup feedwater 02	In	16"	57	Yes	1 1	16" 4"	Gate Globe	Pneu Air
	P27	C03A	SG blowdown 01	Out	4"	57	Yes	1	4"	Globe	Air
	P28	C03B	SG blowdown 02	Out	4"	57	Yes	1	4"	Globe	Air
	P29	C04	SG blowdown recirculation	In	3"	57	Yes	1	3"	Globe	Manu
VFS	P30	C01A	Cont. air filter supply A	In	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air
	P31	C01B	Cont. air filter supply B	In	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air
	P32	C02A	Cont. air filter exhaust A	Out	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air
	P33	C02B	Cont. air filter exhaust B	Out	12"	56	No	1 1	12" 12"	Butterfly Butterfly	Air Air
VWS	P34	C01	Fan Coolers out	Out	10"	56	No	1 1	6 10" 6 10"	Butterfly Butterfly	Pneum Pneum
	P35	C02	Fan coolers in	In	10"	56	No	1 1	10" 10"	Butterfly Butterfly	Pneum Pneum

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trations and Isolation Valves

Isolation Device							Test		
erator	Location	Position N-S-A	Signal	Actuation Mode		Closure Times	Type[1] & Note	Medium	Direction
				Primary	Secondary				
matic or	ORC ORC ORC ORC ORC	O-C-C O-O-C C-C-C C-C-C C-C-C	MS LSL None MS MS	Automatic Automatic Manual Automatic Automatic	Remote Manual Remote Manual None Remote Manual Remote Manual	5 sec 5 sec N/A std. std.	A,2	N_2	Forward
matic or	ORC ORC ORC ORC ORC	O-C-C O-O-C C-C-C C-C-C C-C-C	MS LSL None MS MS	Automatic Automatic Manual Automatic Automatic	Remote Manual Remote Manual None Remote Manual Remote Manual	5 sec 5 sec N/A std. std.	A,2	N_2	Forward
matic	ORC	O-C-C	MF	Automatic	Remote Manual	5 sec	A,2	H_20	Forward
	ORC	C-O-C	LTC	Automatic	Remote Manual	std.			
natic	ORC	O-C-C	MF	Automatic	Remote Manual	5 sec	A,2	H_20	Forward
	ORC	C-O-C	LTC	Automatic	Remote Manual	std.			
	ORC	O-O-C	PRHR	Automatic	Remote Manual	std.	A,2	H_20	Forward
	ORC	O-O-C	PRHR	Automatic	Remote Manual	std.	A,2	H_20	Forward
al	ORC	C-O-C	None	Manual	None	N/A	A,2	H_20	Forward
	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C,4	Air	Forward
	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C	Air	Forward
	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C	Air	Forward
	ORC IRC	C-O-C C-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	5 sec. 5 sec.	C	Air	Forward
natic natic	ORC IRC	O-O-C O-O-C	T,CP T,CP	Automatic Automatic	Remote Manual Remote Manual	std. std.	C,3,4	Air	Forward
natic natic	ORC IRC	O-O-C O-O-C	T,CP T,CP	Automatic Automatic	Remote Manual Remote Manual	std. std.	C,3,4	Air	Forward

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Table 6.

Containment Mechanical Pene

Explanation of Heading and

System:	Fluid system penetrating containment
Containment Penetration:	These fields refer to the penetration itself
Penetration Sleeve I.D.:	Actual penetration identification number
P&ID Sequence No.:	Penetration identification number used on the P&IDs
Line:	Fluid system line
Flow:	Direction of flow in or out of containment
Size.:	Line size
GDC or RG:	Applicable general design criteria or Regulatory Guide
Closed Sys IRC:	Closed system inside containment as defined in SSAR Section 6.2.3.1.1
Isolation Device:	These fields refer to the isolation devices for a given penetration
Qty.:	Number of subject devices per penetration
Size:	Device size
Type:	Device body type
Operator:	Operator type (for valves)
Location:	Device location inside or outside containment
Position N-S-A:	Device position for N (normal operation) S (shutdown) A (post-accident)
Signal:	Device closure signal MS: Main steamline isolation LSL: Low steamline pressure MF: Main feedwater isolation LTC~~A~~: Low Tcold~~avg~~ PRHR: Passive residual heat removal actuation T: Containment isolation RCP: Reactor Coolant Pump auto trip signal CP: High containment pressure TLP: Containment isolation coincident with low header pressure S: Safety Injection Signal
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 automatically closes 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)
Closure Time:	Required valve closure stroke time STD: Industry standard for valve type N/A: Not Applicable

trations and Isolation Valves

Acronyms for Table 6.2.3-1

Test:	These fields refer to the penetration testing requirements
Type:	Required test type

A: Integrated Leak Rate Test
B: Local Leak Rate Test -- penetration
C: Local Leak Rate Test -- fluid systems

Note:	See notes below
Medium:	Test fluid on valve seat
Direction:	Pressurization direction

Forward: high pressure on containment side
Reverse: high pressure on outboard side

Notes:

1. Containment leak rate tests are designated Type A, B, or C according to 10CFR50 Appendix J.

2. The secondary side of the steam generator, including main steam, feedwater, startup feedwater, blowdown and sampling piping from the steam generators to the containment penetration, is considered an extension of the containment. These systems are not part of the reactor coolant pressure boundary and do not open directly to the containment atmosphere during post-accident conditions. During type A tests, the secondary side of the steam generators is ~~drained and~~ vented to the atmosphere outside containment to ensure that full test differential pressure is applied to this boundary.

3. The central chilled water system remains water-filled and operational during the Type A test in order to maintain stable containment atmospheric conditions.

4. The containment isolation valves for this penetration are open during the Type A test to facilitate testing. Their leak rates are measured separately.

5. The inboard butterfly valve is tested in the reverse direction.

6. ~~Testing of the double seals on the personnel hatch doors is performed at a reduced pressure of [LATER] psig. Testing at a lower pressure is conservative since the test pressure tends to open the doors, whereas containment pressure would hold the doors closed.~~

7. The inboard globe valve is tested in the reverse direction. The test is conservative since the test pressure tends to unseat the valve disc, whereas containment pressure would tend to seat the disc.

Table 6.2.3-1 (

Containment Mechanical Pene

System	Containment Penetration										
	Penetration Sleeve I.D.	P&ID Sequence No.	Line	Flow	Size	GDC or Reg Guide	Closed Sys IRC	Qty	Size	Type	C
WLS	P36	C01	Reactor coolant drain tank out	Out	2"	56	No	1 1	2" 2"	Globe Globe	Ai Ai
	P37	C02	Reactor coolant drain tank gas	Both	3/4"	56	No	1 1	3/4" 3/4"	Globe Globe	Ai Ai
	P38	C03	Normal cont. sump	Out	1 1/2"	56	No	1 1	1 1/2" 1 1/2"	Globe Globe	Ai Ai
SPARE	P39			N/A	12"	56	No	1 1	12" ~~12"~~	Flange ~~Flange~~	N/ ~~N/~~
SPARE	P40			N/A	12"	56	No	1 1	12"	Flange ~~Flange~~	N/ ~~N/~~
SPARE	P41			N/A	12"	56	No	1 1	12"	Flange ~~Flange~~	N/ ~~N/~~
SPARE	P42			N/A	12"	56	No	1 1	12"	Flange ~~Flange~~	N/ ~~N/~~
SPARE	P43			N/A	12"	56	No	1 1	12"	Flange ~~Flange~~	N/ ~~N/~~
CNS	H01	N/A	Main equipment hatch	N/A	264"	56	No	1		Double Sealed Hatch	
	H02	N/A	Maintenance hatch	N/A	192"	56	No	1		Double Sealed Hatch	
	H03	N/A	Personnel hatch	N/A	118"	56	No	1		Double Sealed Hatch	
	H04	N/A	Personnel hatch	N/A	118"	56	No	1		Double Sealed Hatch	

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trations and Isolation Valves

perator	Isolation Device						Test		
	Location	Position N-S-A	Signal	Actuation Mode: Primary	Actuation Mode: Secondary	Closure Times	Type[1] & Note	Medium	Direction
	ORC IRC	O-O-C O-O-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	ORC IRC	C-C-C C-C-C	T T	Automatic Automatic	Remote Manual Remote Manual	std. std.	C	Air	Forward
	ORC ~~IRC~~	C-C-C ~~C-C-C~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A	B	Air	Forward
	ORC ~~IRC~~	C-C-C ~~C-C-C~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A	B	Air	Forward
	ORC ~~IRC~~	C-C-C ~~C-C-C~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A	B	Air	Forward
	ORC ~~IRC~~	C-C-C ~~C-C-C~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A	B	Air	Forward
	ORC ~~IRC~~	C-C-C ~~C-C-C~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A ~~N/A~~	N/A	B	Air	Forward
	IRC	C-C-C	None	Manual	None	N/A	B	Air	Forward
	IRC	C-C-C	None	Manual	None	N/A	B	Air	Forward
	IRC	C-C-C	None	Manual	None	N/A	B,6	Air	Forward
	IRC	C-C-C	None	Manual	None	N/A	B,6	Air	Forward

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