Westinghouse Electric Corporation

Energy Systems

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NTD-NRC-94-4184
DCP/NRC0118
Docket No.: STN-52-003

June 30, 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 March 16, 1994, April 19, 1994, April 29, 1994, May 2, 1994, May 11, 1994, May 12, 1994, May 23, 1994 and May 26, 1994. In addition, revisions of responses previously submitted are provided. This completes the responses for the letter dated March 16, 1994.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A. Attachment B is a complete listing of the questions associated with the March 16, 1994 letter and the corresponding letters that provide our response.

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-94-4184
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED JUNE 30,1994

RAI No.	Issue
100.013	Industry codes and standards
210.024R01	Inservice testing
210.033	Acceptable code verification methods
210.042	SSAR sections 3.6.2.3.2, 3.9.3.4, 3.10.1.3
210.052	SSAR section 3.9.1.1
210.060	Elimination of OBE, SSAR scetion 3.7
210.062	Loading combinations for level D condition
210.065	SSAR section 3.9.3.1.7
210.070	SSAR section 3.9.5.2.4
210.073	ASME Class 1,2,3 components procurement specs
210.077	SSAR section 3.6.2
210.080	SSAR section 3.9.3
210.095	SSAR section 3.9.2.5
210.097	SSAR section 3.9.5, basis of deflection allowables
210.100	Flow-induced vibrations of reactor internals
210.102	Preoperational vibration test program
210.103	Stress limits for core support structures
210.104	Exceptions to positions C.1 & C.2 of RG 1.20
210.105	Section 4 of Section 3.9.2 of SRP
210.107	SSAR section 3.9.3.4
210.109	List of ASME Code cases used in design
210.110	Inservice testing of pumps and valves
220.049R01	Exclusion of Cat II structures for foundation anal
220.058	Seismic shear/moments from out-of-phase vibration
220.083	Design information per SRP format
220.089	Critical sections for detailed structural design

RAI No.		Issue
410.117	:	CCS functions
410.118	:	Description of CCS components
410.121	:	CCS conformance to RG 1.29 position C.2
410.122	:	Impact of environmental & dynamic effects on CCS
410.123	:	CCS compliance wuth GDC 44, 45 & 46
410.124	:	CCS compliance with SRP 9.2.2 (IEEE 279)
410.125	:	CCS test and inspections
410.129	:	TB CCWS applicable codes & standards
410.131	:	TB CCWS potential for water hammer
410.132	:	TB CCWS pump NPSH
410.134	:	Analysis of TB CCWS cracks
410.141	:	Turbine missile protection
410.148	:	Condenser air removal sytem quality standards
410.156	:	Oil/hydrocarbon content of CAS air
410.172	:	Effects of dust & dirt on diesel generator
410.173	:	Diesel generator trips
410.189	:	Failure of electrically operated valves
410.237	:	Reference C of Figure 1.7-2
435.074	:	Offsite power recovery probability
440.055	:	Failure of temporary RCS boundaries
440.059	:	Design features for mid-loop
440.062	:	Design features to reduce passive sysems challenge
440.063	:	Analysis of system performance during shutdown
440.065	:	Outage and maintenance activities
440.069	:	Compliance with GL 87-12
440.071	:	Operating experience for shutdown events

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AP600 RAI RESPONSES
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RAI No.		Issue
440.076	:	Effects of water relief on pressurizer safety vlvs
440.081	:	TS LCO 3.4.13, LTOP system
440.085	:	Emergency core makeup & boration
440.087	:	Boration capability to restore shutdown margin
440.088	:	PXS nonsafety-related design requirements
440.092	:	PXS/PCS safe shutdown capability
440.100	:	PXS valve failure states
440.103	:	PRHR operation beyond 72 hours
440.104	:	CMT check valve arrangement
440.112	:	Extended operation by PRHR heat exchangers
440.113	:	Operator actions during mid-loop operation
440.123	:	Commercial service record of canned motor pumps
440.124	:	RCP performance characteristics
440.130	:	RNS functions during shutdown operations
440.134	:	RNS isolation valves
440.137	:	Conformance to RG 1.68
440.138	:	RNS interlocks
440.140	:	GDC compliance for RNS
440.141	:	RV head vent system
440.143	:	Size of RCS vent line
440.144	:	Venting from steam generator U-tubes
460.017	:	Applicability of IE Bulletin 80-65
460.024	:	Dilution flow for liquid waste discharge
460.026	:	Addition of N-16 activity to SSAR Table 11.1-8
480.078	:	Max cont. P,T for severe accident conditions
490.001	:	Similarity to 17x17 VANTAGE-5H

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RAI No.		Issue
490.002	:	Use of option in fuel management
490.003	:	Approval of calculational procedures
490.005	:	VANTAGE-5H vibration problems
491.007	:	Gray rods
492.002	:	Bypass flow
952.074	:	SPES-2 leakage power

ATTACHMENT B
CROSS REFERENCE OF WESTINGHOUSE RAI RESPONSE TRANSMITTALS TO NRC LETTERS OF MARCH 16, 1994

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
220.051	Seismic & structural design consistency	03/16/94	05/17/94
220.052	Computer validation package for INETEC programs	03/16/94	05/17/94
220.053	Rationale for 6 ft thick foundation	03/16/94	05/11/94
220.054	Rationale for 3 ft thick outer walls	03/16/94	05/17/94
220.055	Unevenly distributed construction loads	03/16/94	05/17/94
220.056	Finite element analysis model for basemat	03/[illegible]	05/19/94
220.057	Basis for use of uniform Winkler spring	03/16/94	05/19/94
220.058	Seismic shear/moments from out-of-phase vibration	03/16/94	06/30/94
220.059	Radius and thickness of dome	03/16/94	04/28/94
220.060	Wind loads in Level A&C combinations	03/16/94	05/11/94
220.061	Watertight/airtight seal in SSAR 3.8.2.1.1	03/16/94	06/02/94
220.062	Descriptions for polar crane system	03/16/94	05/11/94
220.063	Air baffle structural design	03/16/94	05/17/94
220.064	Pre-operational SIT of containment in SSAR	03/16/94	05/19/94
220.065	List of missile sources	03/16/94	05/19/94
220.066	Basis for using "1.,0.4,0.4" method	03/16/94	05/17/94
220.067	Operating pressure in load combinations	03/16/94	05/11/94
220.068	Containment shell yield stress properties	03/16/94	04/28/94
220.069	Computer codes used for nonaxisymmetrical loads	03/16/94	05/19/94
220.070	Discussion of structural modules	03/16/94	05/17/94
220.071	Structural modules	03/16/94	05/19/94
220.072	Aging degradation of structural modules	03/16/94	05/17/94
220.073	Design of module joints and connections	03/16/94	05/17/94
220.074	Procedures for concrete placement in modules	03/16/94	05/17/94
220.075	Experience with concrete filled steel structures	03/16/94	05/17/94
220.076	Structural elements in finite element model	03/16/94	05/17/94
220.077	Modular construction design information in SAR	03/16/94	05/19/94
220.078	Modular construction design information	03/16/94	06/16/94
220.079	IRWST design information in SAR	03/16/94	05/11/94
220.080	Computer code for cont. internal structure anal.	03/16/94	05/17/94
220.081	Concrete cracking in seismic analysis	03/16/94	05/17/94
220.082	Note 3 of SSAR Tables 3.8.4-1 and 3.8.4-2	03/16/94	05/17/94
220.083	Design information per SRP format	03/16/94	06/30/94
220.084	Staff positions from 1/20/94 meeting	03/16/94	05/17/94
220.085	Internal friction angle for backfill soil	03/16/94	05/17/94
220.086	Applying seismic loads to finite element model	03/16/94	05/19/94
220.087	Energy component for embedment effect	03/16/94	05/11/94
220.088	Use of coated rebar	03/16/94	05/11/94
220.089	Critical sections for detailed structural design	03/16/94	06/30/94
220.090	Type and characteristics of water seals	03/16/94	05/11/94
230.050	Auditable trail for final seismic calculations	03/16/94	05/11/94
230.051	COL commitment for reconciliation analysis	03/16/94	04/28/94
230.052	Evaluation of foundation mat uplift potential	03/16/94	05/17/94
230.053	Inclusion of other site conditions	03/16/94	05/11/94
230.054	Zone 3 requirements of UBC for analysis	03/16/94	06/27/94
230.055	Use of SASSI	03/16/94	05/11/94
230.056	Structure to structure interaction	03/16/94	05/19/94
230.057	Integrity of cont. shell-shield bldg connection	03/16/94	04/28/94
230.058	High frequency modes of structures	03/16/94	05/17/94
230.059	Comparison between SRSS and 1,.4,.4 method	03/16/94	05/19/94
230.060	Accidental torsion in overall seismic member force	03/16/94	05/17/94
230.061	Effects of high-frequency structural modes	03/16/94	05/17/94
230.062	Validity of fixed base seismic analysis	03/16/94	05/11/94
230.063	Soil column properties for horiz.& vert. models	03/16/94	05/17/94
230.064	Adequacy of M-O method	03/16/94	05/19/94
230.065	Lateral earth pressures on NI structure walls	03/16/94	05/19/94
230.066	Adequacy of Zone 2A requirements of UBC	03/16/94	06/27/94
230.067	Lower bound shear wave for soft soil site	03/16/94	05/17/94
230.068	Use of concentric or dual zone systems	03/16/94	06/27/94
230.069	Shallow soil site conditions	03/16/94	05/11/94
230.070	Procedure for developing seismic response envelope	03/16/94	05/11/94
230.071	Use of seismic responses for soft rock site	03/16/94	05/19/94
230.072	Statistical independence of acceleration time hist	03/16/94	05/17/94
230.073	Clarification of SSAR Section 3.2.1.1.2	03/16/94	06/27/94
230.074	Ordinates and units for ground motion plots	03/16/94	05/17/94

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
230.075	Use of 7% damping	03/16/94	05/17/94
230.076	Use of damping ratio for cable tray systems	03/16/94	05/19/94
230.077	Limits of modal damping	03/16/94	05/17/94
230.078	Time discretization of ground motion time hist.	03/16/94	05/17/94
230.079	Use of Seed-Idriss 1970 curves in SSI analyses	03/16/94	05/11/94
230.080	Adequacy of using only 3 soil site conditions	03/16/94	05/11/94
230.081	Use of computer code SAP or BSAP	03/16/94	05/11/94
230.082	Method of anlaysis used to calculate seismic force	03/16/94	05/17/94
230.083	Seismic response forces for soil site conditions	03/16/94	05/17/94
230.084	Additional information in Section 3.7.2.1.1	03/16/94	05/11/94
230.085	Additional information in section 3.7.2.3	03/16/94	05/17/94
230.086	Effects of energy feedback	03/16/94	05/19/94
230.087	Method of ground motion combination	03/16/94	05/19/94
230.088	SSAR Section 3.7.2.2 inconsistency	03/16/94	05/17/94
230.089	SSAR Section 3.7.2.3.1 explanation needed	03/16/94	05/17/94
230.090	Dynamic model usage	03/16/94	05/17/94
230.091	Section 3.7.2.1.2 clarification	03/16/94	05/17/94
230.092	Stiffness properties	03/16/94	05/17/94
230.093	Exclusion of accidental torsion	03/16/94	05/17/94
230.094	Decoupling subsystems from primary structural sys	03/16/94	05/17/94
230.095	Seismic instrumentation	03/16/94	05/17/94
231.015	Geography & demography limits for a site	03/16/94	05/11/94
231.016	Flood level to plant grade design features	03/16/94	05/11/94
231.017	Site qualification flow chart	03/16/94	05/17/94
231.018	SSAR Table 2.0-1 clarification	03/16/94	05/17/94
231.019	SSAR Table 2.0-1 modification	03/16/94	05/17/94
231.020	Basemat location for COL applicant comparison	03/16/94	05/17/94
231.021	SRP 3.7.2 guidance	03/16/94	05/19/94
231.022	SSI studies for the rock model	03/16/94	05/11/94
231.023	Acceptable ranges of backfill properties	03/16/94	05/19/94
231.024	SSAR Section 2A.2	03/16/94	05/17/94
231.025	Dynamic shear properties of rock material	03/16/94	05/17/94
231.026	Properties in SSAR Table 2A-6	03/16/94	05/19/94
231.027	Statements in SSAR Section 2A.4 justification	03/16/94	05/17/94
231.028	Dynamic properties of soil columns	03/16/94	05/17/94
231.029	Effects of assumed Poisson ratio	03/16/94	05/19/94
231.030	Impact of non-vertically incident ground motion	03/16/94	05/11/94
231.031	SSAR Appendix 2A.2	03/16/94	05/11/94
231.032	Geosciences investigation performed by COL App.	03/16/94	05/17/94

Records printed: 104

Question 100.13

Provide a listing of all industry codes and standards used in AP600 design in the SSAR.

Response:

A listing of the industry codes and standards referenced in the AP600 SSAR is provided below.

	Code or Standard	Title
ANSI		
1.	ANSI N16.1-75	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
2.	ANSI/ANS-59.4-1979	Generic Requirements for Light Water Nuclear Power Plant Fire Protection
3.	ANSI N16.9-75	Validation of Calculational Methods for Nuclear Criticality Safety
4.	ANSI N210-76	Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations
5.	ANSI N14.6-1986	Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More
6.	ANSI C2-1990	National Electric Safety Code
7.	ANSI C37.010-1972	Application Guide for ac High Voltage Circuit Breakers
8.	ANSI C37.90-1989	IEEE Standard for Relays and Relay Systems Associated with Electric Power Apparatus
9.	ANSI C57.12.00-1973	General Requirements for Distribution, Power, Regulating Transformers, and Shunt Reactors
10.	ANSI 58.6-1983	Criteria for Remote Shutdown for Light Water Reactors
11.	ANSI N278.1-1975	Self-Operated and Power-Operated Safety-Relief Valves Functional Specification Standard
12.	ANSI B16.41-1983	Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants
13.	ANSI B16.34-1981	Valves - Flanged and Buttwelding End
14.	ANSI N18.2a-75	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
15.	ANSI N18.2-1973	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
16.	ANSI N57.2-1983	Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations

	Code or Standard	Title
17.	ANSI N57.3-1983	Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants
18.	ANSI N4.5-1980	Criteria for Accident Monitoring Functions in Light-Water Cooled Reactors
19.	ANSI 5.1-1979	Decay Heat Power in Light Water Reactors
20.	ANSI 56.5-1979	PWR and BWR Containment Spray System Design Criteria
21.	ANSI B96.1-81	Welded Aluminum-Alloy Storage Tanks
22.	ANSI/ANS-58.2	Design Bases for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture
23.	ANSI/ANS-56.11-1988	Design Criteria for Protection against the Effects of Compartment Flooding in Light Water Reactor Plants
24.	ANSI/AMCA 211-85	Certified Ratings Program Air Performance
25.	ANSI/AMCA 210-85	Laboratory Method of Testing Fans for Rating Purposes
26.	ANSI/AMCA 300-85	Reverberant Room Method of Testing Fans for Rating Purposes
27.	ANSI/ARI 410-91	Forced-Circulation Air Cooling and Air Heating Coils
28.	ANSI/HF 100-1988	American National Standard for Human Factors Engineering of Visual Display Terminal Workstations
ANS		
1.	ANS 57.2-1983	Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants
2.	ANS 57.1-1980	Design Requirements for Light Water Reactor Fuel Handling Systems
3.	ANSI/ANS-56.8-1987	Containment System Leakage Testing Requirements
4.	ANSI/ANS-5.1-1979	Decay Heat Power in Light Water Reactors
5.	ANSI/ANS-51.1-1983	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
6.	ANSI/ANS-58.8-1984	Time Response Design Criteria for Nuclear Safety Related Operator Actions
7.	ANSI/ANS-55.6	Liquid Radioactive Waste Processing Systems for Light Water Reactor Plants
8.	ANSI/ANS-18.1	Radioactive Source Term for Normal Operation of Light Water Reactors

	Code or Standard	Title
ASME		
1.	ASME Boiler and Pressure Vessel Code	Section II, Material Specifications Section III, Nuclear Specifications Section V, Nondestructive Examination Section XIII, Pressure Vessels Section IX, Welding and Brazing Qualification Section XI, Rules for inservice inspection of Nuclear Power Plant Components
2.	ASME NOG-1-1989	Rules for Construction of Overhead and Gantry Cranes
3.	ASME/ANSI B30.9-1990	Slings
4.	ASME/ANSI-B31.1-1989	Code for Power Piping
5.	ASME/ANSI AG-1-85	Code on Nuclear Air and Gas Treatment
6.	ASME 19.11-1970	Power Test Code
7.	ANSI/ASME B30.2-1983	Overhead and Gantry Cranes
8.	ANSI/ASME OM-1987	Operation and Maintenance of Nuclear Power Plants
IEEE		
1.	IEEE 232-1983	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
2.	IEEE 279-1971	IEEE Criteria for Protection Systems for Nuclear Power Generating Stations
3.	IEEE 308-1980	IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
4.	IEEE 317-1983	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
5.	IEEE 323-1983	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
6.	IEEE/ANSI 334-82	Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations
7.	IEEE 338-1987	IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems
8.	IEEE 344-1987	IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
9.	IEEE 379-1988	IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

	Code or Standard	Title
10.	IEEE 382-1985	IEEE Standard for Qualification for Actuators for Power Operated Valve Assemblies with Safety Related Functions for Nuclear Power Plants
11.	IEEE 383-1974	IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations
12.	IEEE 384-1974	IEEE Separation of Class 1E Equipment and Circuits
13.	IEEE 384-1981	IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits
14.	IEEE 422-1986	Guide for the Design and Installation of Cable Systems in Power Generating Stations
15.	IEEE 450-1987	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations
16.	IEEE 484-1987	IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations
17.	IEEE 485-1983	IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations
18.	IEEE 603-1991	IEEE Criteria for Safety Systems for Nuclear Power Generator Stations
19.	IEEE 627-1980	IEEE Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations
20.	IEEE 741-1990	IEEE Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations
21.	IEEE 796-1983	IEEE Microcomputer System Bus
22.	IEEE 946-1985	IEEE Recommended Practice for the Design of Safety-Related dc Auxiliary Power Systems for Nuclear Power Generating Stations
23.	IEEE C37-89	IEEE Standards on Circuit Breakers, Switch Gear, Relays, Substations, Fuses, etc.
ASTM		
1.	ASTM B8	Standard Specification for Concentric-Lay-Stranded Copper Conductors, Hard, Medium-Hard, or Soft, 1971
2.	ASTM B-33	Standard Specification for Tinned Annealed Copper Wire for Electrical Purposes 1971

	Code or Standard	Title
3.	ASTM D-4082	Test Method for Effects of Radiation on Coatings Used in Light Water Nuclear Power Plants
4.	ASTM D-4256	Test Method for Determination of the Decontaminability of Coatings Used in Light Water Nuclear Power Plants
5.	ASTM D-3911	Test Method for Evaluating Coatings used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions
6.	ASTM D-4227	Practice for Qualification of Journeyman Painters for Application of Coatings to Concrete Surfaces of Safety Related Areas in Nuclear Facilities
7.	ASTM D-4228	Practice for Qualification of Journeyman Painters for Application of Coatings to Steel Surfaces of Safety Related Areas in Nuclear Facilities
8.	ASTM D-4537	Guide for Establishing Procedures to Qualify and Certify Inspection Personnel for Coating Work in Nuclear Facilities
9.	ASTM A-609-78	Standard Specification for Longitudinal Beam Ultrasonic Inspection of Carbon and Low-alloy Steel Castings
10.	ASTM-E-165-80	Practice for Liquid Penetrant Inspection Method
11.	ASTM C 94-1990	Specifications for Ready-Mixed Concrete
12.	ASTM C 150-1989	Specification for Portland Cement
13.	ASTM C 33-1990	Specification for Concrete Aggregates
14.	ASTM C 131-1989	Resistance to Abrasion of Small Size Course Aggregate by Use of the Los Angeles Machine
15.	ASTM C 535-1989	Test Method for Resistance to Degradation of Large-Size Course Aggregate by Abrasion and Impact in the Los Angeles Machine
16.	ASTM D 512-1989	Chloride Ion in Industrial Water
17.	ASTM E-185-82	Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels
18.	ASTM-E-142-86	Methods for Controlling Quality of Radiographic Testing
19.	ASTM-A-580-90	Specification for Stainless and Heat-resisting Steel Wire
20.	ASTM Standard D 2487	Standard Test Method for Classification of Soils for Engineering Purposes
21.	ASTM D 1888-1978	Particulate and Dissolved Matter in Industrial Water
22.	ASTM C 618-1989	Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete

	Code or Standard	Title
23.	ASTM C 311-1990	Sampling and Testing Fly Ash or Natural Pozzolans for Use as Mineral Admixture in Portland Cement Concrete
24.	ASTM C 260-1986	Air Entraining Admixtures for Concrete

Miscellaneous

1.	Uniform Plumbing Code	Section 318; 1991
2.	ICEA P-54-440-1986	Ampacities of Cables in Open-Top Cable Trays
3.	ICEA S-19-81	Rubber-Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy, Rev. 5, 1976
4.	ICEA S-66-524	Cross-Linked-Thermosetting-Polyethylene-Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy
5.	ICEA S-68-516	Ethylene-Propylene-Rubber-Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy 1977
6.	NEC	National Electrical Code 1990
7.	ACI 304R-1989	Guide for Measuring, Mixing, Transporting, and Placing Concrete
8.	API 610-81	Centrifugal Pumps for General Refinery Services
9.	API-650-80	Welded Steel Tanks for Oil Storage, 1984
10.	AWWA D100-84	Welded Steel Tanks for Water Storage
11.	API-620-82	Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks, 1985
12.	NEMA MG-1-78	Motors and Generators, 1984
13.	SMACNA	HVAC Duct Construction Standards, 1985
14.	NFPA	National Fire Protection Association Codes and Standards
15.	IES	Lighting Handbook
16.	ASHRAE 52-76	Methods of Testing Air-Cleaning Devices Used in General Ventilation for Removing Particulate Matter
17.	UL-900	Test Performance of Air-Filter Units
18.	UL-586	High-Efficiency, Particular, Air-Filter Units
19.	UL-1096	Electric Central Air Heating Equipment
20.	ASHRAE 33-78	Methods of Testing for Rating Forced Circulation Air Cooling and Air Heating Coils
21.	ARI 620-80	Self-Contained Humidifiers

	Code or Standard	Title
22.	AMCA 500-83	Testing Methods for Louvers, Dampers, and Shutters
23.	UL-555	Fire Dampers
24.	SMACNA, 1975	High-Pressure Duct Construction Standards
25.	SMACNA, 1985	HVAC Duct Construction Standards
26.	SMACNA, 1985	Duct Leakage Test Manual
27.	SMACNA, 1983	HVAC Systems - Testing, Adjusting, and Balancing
28.	UL-555S, 1983	Leakage Rated Dampers for Use in Smoke Control System
29.	ISA-S18.1	Instrument Society of America, Annunciator Sequences and Specifications, 1979
30.	ASHRAE 55-81	Thermal Environmental Conditions for Human Occupancy
31.	IEC-964	International Electrotechnical Commission, Design for Control Rooms of Nuclear Power Plants
32.	ICRP Publication 2	Report of Committee II on Permissible Dose for Internal Radiation
33.	ICRP Publication 30	Limits for Intakes of Radionuclides by Workers
34.	EPA-520/1-88-020	Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion
35.	AWWA 504-80	Rubber Seated Butterfly Valves

SSAR Revision: NONE

Question 210.24

Section 3.9.6 of the SSAR states that an inservice testing (IST) program for pumps and valves will be submitted by the Combined License (COL) applicant. However, there is no mention of a submittal of an IST program by Westinghouse for the AP600 design certification application.

Provide an IST program to demonstrate that adequate design and access provisions will be incorporated to permit the effective performance of IST. The staff will review this IST program to ensure that the Westinghouse's commitments regarding the ability to test pumps and valves can be met.

Response (Revision 1):

The AP600 inservice testing (IST) program plan to support design certification is contained in the SSAR Revision below. This program plan describes the valves included in the IST plan and the type of testing to be performed on each. As described in SSAR Subsection 3.9.6, the AP600 IST program plan addresses the requirements in Section XI of the ASME Code for safety-related components. As discussed in SSAR Subsection 3.9.6 and in RAI 210.25 the AP600 has no safety-related pumps and therefore there are no safety-related pumps that require IST. The AP600 IST program plan does address inservice testing for safety-related valves.

The AP600 IST program plan does not include testing of nonsafety-related pumps or valves. This is based on the AP600 implementation of the regulatory treatment of nonsafety-related systems (RTNSS) process (refer to WCAP-13856, 9/93). Note that safety-related valves in nonsafety-related systems (containment isolation valves) are included in the IST program plan when they have safety-related missions.

The information included in the AP600 IST program plan contains an appropriate level of detail to support the AP600 design certification application and to provide guidance to the Combined License applicant in the development of the detailed IST program. This information demonstrates that adequate design provisions have been incorporated into the AP600 to permit effective performance of the IST program.

SSAR Revision:

Revise Subsection 3.9.6 as follows:

3.9.6 In-service Testing of Pumps and Valves

In-service testing of ASME Code, Section III, Class 1, 2, and 3 pumps and valves is performed in accordance with Section XI of the ASME Code and applicable addenda, as required by 10 CFR 50.55a(g), except where specific relief has been granted by the NRC in accordance with 10 CFR 50.55a(g). The Code includes requirements for system pressure tests and functional tests for active components.

The requirements for system pressure tests are defined in ASME Code, Section XI, IWA-5000. These tests verify the pressure boundary integrity in conjunction with in-service inspection.

Testing requirements for components constructed to the ASME Code are in several parts of the ASME document ASME OM (Reference 2). This document is periodically updated. The edition and addenda to be used is administratively controlled by the Combined License applicant.

The specific ASME Code requirements for functional testing of pumps are found in Section XI, IWP and ASME/ANSI OM, Part 6. The specific ASME Code requirements for functional testing of valves are found in Section XI, IWV and ASME/ANSI OM, Part 10. The functional tests are required for those pumps and valves that are required for safety.

The AP600 in-service test plan does not include testing of pumps and valves in nonsafety-related systems unless they perform safety-related missions, such as containment isolation. This is based on the AP600 implementation of the regulatory treatment of nonsafety-related systems (RTNSS) process (Reference 14).

A preservice test program which identifies the required functional testing for each unit is to be submitted to the NRC by the Combined License applicant prior to performing the tests. The in-service test program, which identifies requirements for functional testing, will be submitted to the NRC by the Combined License applicant. These programs will comply with applicable provisions of 10 CFR 50.55a(g) and NRC guidelines. The preservice test program provides details of components subject to testing, as well as the method and extent of preservice testing. The in-service test program details the components subject to testing and method, extent, and frequency of testing.

3.9.6.1 In-service Testing of Pumps

There are no safety-related pumps in the AP600. The test program for the pumps is based on the applicable requirements of the applicable construction code or reliability considerations.

The test program for pumps is controlled administratively by the Combined License holder and does not have to be addressed in the technical specifications. The AP600 in-service test plan does not include testing of pumps in nonsafety-related systems unless they perform safety-related missions. Based on the AP600 implementation of the regulatory treatment of nonsafety-related systems (RTNSS) process (Reference 14), there are no nonsafety-related pumps that have IST.

3.9.6.2 In-service Testing of Valves

The safety-related ASME Code, Section III, Class 1, 2, and 3 valves are subject to operational readiness testing. Inservice testing of valves assesses operational readiness including actuating and position indicating systems. The valves which are subject to inservice testing include those valves that perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident. In addition, pressure relief devices used for protecting systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident, are subject to inservice testing.

The AP600 in-service test plan does not include testing of valves in nonsafety-related systems unless they perform safety-related missions. Based on the AP600 implementation of the regulatory treatment of nonsafety-related systems (RTNSS) process (Reference 14), there are no nonsafety-related valves that have IST.

The valve test program is based on the plan outlined in this subsection. Valves (including relief valves) subject to inservice testing in accordance with Section XI of the ASME Code are indicated in Table 3.9.6-1. This table includes the type of testing to be performed and the frequency at which the testing should be performed. The specific tests performed are based on the safety-related missions and functions for each valve. The test program is

to conform to the requirements of ASME OM Part 10, to the extent practical, and to comply with applicable portions of 10 CFR 50.55a(g). The guidance in NRC Generic Letters, AEOD reports, and industry and utility guidelines (including NRC Generic Letter 89-04) is also considered in developing the test program.

Relief from the requirements for testing, if required, and the alternative to the tests are justified and documented as part of the test program development.

3.9.6.2.1 Valve Functions Tested

The AP600 inservice testing program plan identifies the safety-related missions for safety-related valves for the AP600 systems. The following safety-related valve missions have been identified in Table 3.9.6-1.

- Maintain Closed
- Maintain Open
- Transfer Closed (active function)
- Transfer Open (active function)
- Throttle Flow (active function)

Based on the safety-related missions identified for each valve, the inservice tests to confirm the capability of the valve to perform these missions are identified. Active valves include valves that transfer open, transfer closed, and/or have throttling missions. Active valves, as defined in the ASME Code, include valves that change obturator (the part of the valve that blocks the flow stream) position to accomplish the safety-related function(s). Valve missions to maintain closed and maintain open are designated as passive and do not include valve exercise inservice testing.

If upon removal of the actuation power (electrical power, air or fluid for actuation) an active valve fails to the position associated with performing its safety-related function, it is identified as "active to fail" in the Table 3.9.6-1.

Valve functions are used in determining the type of inservice testing for the valve. These valve functions include:

- Active or Active to Fail for fulfillment of the safety-related mission(s),
- Reactor coolant system pressure boundary isolation function,
- Containment isolation function,
- Seat leakage (in the closed position), is limited to a specific maximum amount when important for fulfillment of the safety-related mission(s),
- Actuators that fail to a specific position (open/closed) upon loss of actuating power for fulfillment of the safety-related mission(s),
- Safety-related remote position indication.

The ASME inservice testing categories are assigned based on the safety-related valve functions and the valve characteristics. The following criteria are used in assigning the ASME inservice testing categories to the AP600 valves.

Category A - Safety-related valves with safety-related seat leakage requirements.
Category B - Safety-related valves requiring inservice testing, but without safety-related seat leakage requirements.
Category C - Safety-related, self-actuated valves (such as check valves and pressure relief valves).
Category D - Safety-related, explosively-actuated valves.

3.9.6.2.2 Valve Testing

Four basic groups of inservice tests have been identified for the AP600. These testing groups are outlined in the following

Remote Valve Position Indication Inservice Tests

Valves have remote valve position indication inservice tests when there is a safety-related remote valve position indication and this remote valve position indication is used to satisfy the safety-related mission of the valve.

Valve Leakage Inservice Tests

Valves with safety-related seat leakage limits will be tested to verify their seat leakage. These valves include:

- Pressure Isolation - valves that provide isolation between high and low pressure systems.
- Temperature Isolation - valves whose leakage may cause unacceptable thermal loading to piping or supports.
- Containment Isolation - valves that provide isolation of piping/lines that penetrate the containment.

In some cases pressure isolation is satisfied by performing a flow test and observing that the valve position indication is accurately representing the valve positions during this flow test. This approach is applied to the passive core cooling system accumulator check valves and the in-containment refueling water storage tank injection check valves.

Containment isolation valves are tested in accordance with 10CFR50, Appendix J. Depending on the function and configuration, some valves are tested during the integrated leak rate testing (Type A) or individually as a part of the Type C testing or both.

The ASME Code does not require additional leak testing for valves that demonstrate operability during the course of plant operation. Therefore, valves that meet this criteria are not leak tested.

Valve Exercise Inservice Tests

Manual/Power Operated Valve Tests

Manual/Power Operated Valve Exercise Tests - Safety-related active valves, both manual and power-operated (motor-operated, air-operated, hydraulically-operated, solenoid-operated, etc.) will be exercised periodically. The ASME code specifies a quarterly valve exercise frequency. Exceptions to this frequency are taken when meeting

this test frequency results in a plant trip or equipment damage. The AP600 test frequencies are identified in the attached table.

If an exception is taken to performing quarterly full exercise testing of a valve, then full stroke testing will be performed during cold shutdowns on a frequency not more often than quarterly. If this is not possible, then the stroke testing will be performed once each refueling cycle.

The inservice testing requirement for measuring stroke time for valves in the AP600 will be completed in conjunction with a valve exercise inservice test. The stroke time test is not identified as a separate inservice test.

The ASME Code does not specify exercise testing for valves that demonstrate operability during the course of plant operation. Therefore, exercise testing is not identified for valves that meet this criteria.

Power Operated Valve Operability Tests - A valve operability test will be performed on powered-operated valves that operate under high differential pressures to perform their safety-related functions. Valves which operate under low differential pressures to perform their safety-related missions and incorporate additional margins are exempted from this testing. This inservice operability testing includes exercising a valve to the positions that fulfill the safety-related functions, at operating conditions (differential pressure and flow) as near as practicable to those expected during their safety-related missions. The safety-related missions for power-operated valves include transferring open, transferring closed, and throttling. This test is performed once each refueling cycle.

If design basis operating conditions can not be reasonably used, then partial flow / differential pressure conditions will be used. If partial conditions can not be used, then alternate means should be used to determine operability. Alternate means include methods such as non-intrusive / diagnostic techniques or valve disassembly and inspection. The test program for motor-operated valves is developed using the appropriate guidelines (including NRC Generic Letter 89-10).

Power Operated Valves That Fail to a Specified Position - Safety-related valves that fail to the safety-related actuation position to perform the safety-related missions, and rely upon nonsafety-related systems to provide actuation power to establish and maintain the nonsafety-related position or positions are subject to a valve exercise inservice test. The test must verify that the nonsafety-related systems do not assist the valve in repositioning to the safety-related position and the nonsafety-related systems do not prevent the valve from repositioning to the safety-related position. This inservice test confirms that the valves fail to the safety-related position without assistance from nonsafety-related systems and should be accomplished using only safety-related components. The specified frequency for these tests is once each refueling cycle.

Check Valve Tests

Check Valve Flow Tests - Safety-related check valves identified with specific safety-related missions to transfer open or transfer closed are tested periodically. Exercising a check valve confirms the valve capability to move to the position(s) to fulfill the safety-related mission(s). Forward flow and reverse flow inservice testing is individually specified. Alternatives to exercising that can verify obturator travel to the required position are allowed when flow testing is not practicable. Alternatives include non-intrusive / diagnostic techniques or valve disassembly and inspection.

The specified frequency for this inservice test is once each refueling cycle.

Check Valve Low Differential Pressure Tests - Safety-related check valves that perform a safety-related mission to transfer open or transfer closed under low differential pressure conditions will have periodic exercising inservice testing to verify the capability of the valve to operate with these safety-related conditions.

The intent of this inservice test is to exercise the valve to the position to fulfill the safety-related function(s), at or near design basis conditions (differential pressure and flow). Forward flow and reverse flow inservice testing is individually specified. This inservice test is performed in addition to forward and/or reverse flow inservice tests.

The specified frequency for this inservice test is once each refueling cycle.

Other Valve Inservice Tests

Explosively-Actuated Valves - Explosively-actuated valves are subject to periodic test firing of the explosive actuator charges. The inservice tests for these valves is specified in the ASME code. At least 50 percent of the charges installed in the plant in explosively-actuated valves shall be fired and replaced at least once each refueling cycle. The firing of the explosive charge may be performed inside of the valve or outside of the valve in a test fixture.

Pressure/Vacuum Relief Devices - Pressure relief devices that provide safety-related functions or that protect equipment in systems that perform AP600 safety-related missions are specified by ASME to have periodic inservice testing. The inservice tests for these valves are identified in ASME IST, Appendix I.

The periodic inservice testing include visual inspection, seat tightness determination, set pressure determination, and operational determination of balancing devices, alarms, and position indication as appropriate. The frequencies for this inservice test is specified as every five years for ASME Class 1 or every ten years for ASME Classes 2 and 3

Add the following reference to section 3.9.8

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

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
CAS-PL-V040	Service Air Containment Isol.	Globe / Diaph w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown Fail Safe / Refueling		9.3.1-1
CAS-PL-V041	Service Air Contain Isol Check	Check / Self-Actuated	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.3.1-1
CAS-PL-V084	Breathing Air Containment Isol	Globe / Hand	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.3.1-1
CAS-PL-V085	Breathing Air Cont Isol Check	Check / Self-Actuated	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.3.1-1
CCS-PL-V200	Containment Isolation-Inlet	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown		9.2.2-2
CCS-PL-V201	Containment Isolation-Inlet	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown		9.2.2-2
CCS-PL-V207	Containment Isolation-Outlet	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown		9.2.2-2
CCS-PL-V208	Containment Isolation-Outlet	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown		9.2.2-2
CCS-PL-V209	Containment Isolation-Outlet	Stop Check / Hand	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.2.2-2
CCS-PL-V257	Containment Isolation - Inlet	Stop Check / Hand	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.2.2-2
CVS-PL-V001	RCS Purification Stop	Gate / Motor	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling RCS Press Bound leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		9.3.6-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
CVS-PL-V002	RCS Purification Stop	Gate / Motor	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling RCS Press Bound leak test / Refueling Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		9.3.6-2
CVS-PL-V040	Resin Flush IRC Isol	Ball / Hand	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.3.6-2
CVS-PL-V041	Resin Flush ORC Isol	Ball / Hand	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.3.6-2
CVS-PL-V042	Flush Ln Cont Isol Relief	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Class 2/3 RV Tests / Every 10 Years during Refueling		9.3.6-2
CVS-PL-V045	Letdown Containment Isol IRC	Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		9.3.6-2
CVS-PL-V047	Letdown Containment Isol ORC	Globe / Diaph w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		9.3.6-2
CVS-PL-V056	Letdown Line Cont Isol Relief	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Class 2/3 RV Tests / Every 10 Years during Refueling		9.3.6-2
CVS-PL-V079	RNS Pump Suction Cont Isol	Ball / Hand	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.3.6-2
CVS-PL-V081	RCS Charging Stop	Stop Check / Diaphragm	A	Maintain Close Transfer Close	Active to Failed RCS Press Boundary	B	Check Rvrs Flow, closing / Refueling Fail Safe / Refueling		9.3.6-2
CVS-PL-V082	RCS SG Makeup Line	Check / Self-Actu	A	Maintain Close	Active	BC	Check Rvrs Flow, closing / Refueling		9.3.6-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type - Frequency	IST Notes	SSAR Fig. No.
CVS-PL-V084	Aux Pzr Spray Line Isolation	Stop Check / Diaphragm	A	Maintain Close Transfer Close	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Check-Rvrs Flow, closing / Refueling		9.3.6-2
CVS-PL-V085	Aux Pzr Spray Line	Check / Self-Actuated	A	Maintain Close Transfer Close	Active RCS Press Boundary	BC	Check-Rvrs Flow, closing / Refueling Check-Initial Close Flow / Refueling		9.3.6-2
CVS-PL-V090	Makeup Line Cont Isolation	Globe / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years		9.3.6-2
CVS-PL-V091	Makeup Line Cont Isolation	Globe / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years		9.3.6-2
CVS-PL-V092	Hydrogen Add Cont Isol	Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Refueling Fail Safe / Refueling		9.3.6-2
CVS-PL-V094	Hydrogen Add IRC Isol	Check / Self-Actuated	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.3.6-2
CVS-PL-V136A	Demin Water System Isolation	Gate / Motor	C	Maintain Close Transfer Close	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		9.3.6-2
CVS-PL-V136B	Demin Water System Isolation	Gate / Motor	C	Maintain Close Transfer Close	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		9.3.6-2
CVS-PL-V171	PXS Makeup Line Cont Isol	Globe / Diaphragm	B	Maintain Close	Active to Failed Containment Isolation Safety Seat Leakage	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		9.3.6-2
CVS-PL-V172	PXS Test Hdr IRC Cont Isol	Check / Self-Actuated	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.3.6-2
DWS-PL-V030	Demin. Water Containment Isol	Globe / Hand	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.2.4-1

Table 3.9.6-1

Valve Inservice Test Requirements

Valve Tag No.	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
[illegible]	[illegible]	Check / Self-Actuated	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.2.4-1
[illegible]	...ainment [illegible]	Globe / Hand	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.5.1-1
[illegible]	Cont Iso [illegible]	Check / Self-Actuated	B	Maintain Close	Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling		9.5.1-1
[illegible]	...T Isolation	Butterfly / Diaphragm	C	Transfer Open	Active to Failed Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.2.2-1
[illegible]	...WST Isolation	Butterfly / Diaphragm	C	Transfer Open	Active to Failed Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.2.2-1
[illegible]	...olation	Gate / Motor	C	Maintain Open	Active	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		6.2.2-1
[illegible]	[illegible]	Gate / Motor	C	Maintain Open	Active	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Exercise Full Stroke - Rflg / Refueling		6.2.2-1
[illegible]	[illegible]	[illegible]	C	Transfer Open	Active	B	Exercise Full Stroke - Rflg / Refueling		6.2.2-1
[illegible]	[illegible]	[illegible]	C	Transfer Open	[illegible]	B	Remote Position Indicator / Refueling Operability Test, Full D/P / Every 5 Years during Refueling Fail Safe / Refueling		9.3.3-2
[illegible]	[illegible]	[illegible]	[illegible]	Transfer Open	Active to Failed RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Operability Test, Full D/P / Every 5 Years during Refueling Fail Safe / Refueling		9.3.3-2
[illegible]	[illegible]	[illegible]	[illegible]	...n Close	Active to Failed ...oundary	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.3.3-2
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	...e Position Indicator / Refueling ...k test / Refueling [illegible]		9.3.3-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
PSS-PL-V010A	Cont Isol - Liquid Sample Line	Globe / Solenoid	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak sest / Refueling Operability Test, Full D/P / Every 5 Years during Refueling Fail Safe / Refueling		9.3.3-2
PSS-PL-V010B	Cont Isol - Liquid Sample Line	Globe / Solenoid	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years during Refueling Fail Safe / Refueling		9.3.3-2
PSS-PL-V011	Cont Isol - Liquid Sample Line	Globe / Solenoid	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Operability Test, Full D/P / Every 5 Years during Refueling Fail Safe / Refueling		[illegible]
PSS-PL-V023	Cont Isol - Sample Return Line	Globe / Solenoid	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		9.3.3-2
PSS-PL-V024	Cont. Sample Return Check	Check / Self-Actuated	B	Maintain Close	Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling		9.3.3-2
PSS-PL-V046	Cont Isol - Air Sample Line	Globe / Solenoid	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Postion Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.3.3-2
PXS-PL-V002A	CMT A CL Inlet Isolation	Gate / Motor	A	Maintain Open			Remote Postion Indicator / Refueling		6.3-1
PXS-PL-V002B	CMT B CL Inlet Isolation	Gate / Motor	A	Maintain Open			Remote Postion Indicator / Refueling		6.3-1
PXS-PL-V014A	CMT A Discharge Isolation	Globe / Diaphragm	A	Maintain Open Transfer Open	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.3-1
PXS-PL-V014B	CMT B Discharge Isolation	Globe / Diaphragm	A	Maintain Open Transfer Open	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		[illegible]
PXS-PL-V015A	CMT A Discharge Isolation	Globe / Diaphragm	A	Maintain Open Transfer Open	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.3-1
PXS-PL-V015B	CMT B Discharge Isolation	Globe / Diaphragm	A	Maintain Open Transfer Open	Active to Failed Remote Position	d	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.3-1

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
PXS-PL-V016A	CMT A Discharge Check	Check / Self-Actuated	A	Maintain Open Transfer Open Transfer Close	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Rvrs Flow, closing / Refueling		6.3-1
PXS-PL-V016B	CMT B Discharge Check	Check / Self-Actuated	A	Maintain Open Transfer Open Transfer Close	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Rvrs Flow, closing / Refueling		6.3-1
PXS-PL-V017A	CMT A Discharge Check	Check / Self-Actuated	A	Maintain Open Transfer Open Transfer Close	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Rvrs Flow, closing / Refueling		6.3-1
PXS-PL-V017B	CMT B Discharge Check	Check / Self-Actuated	A	Maintain Open Transfer Open Transfer Close	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Rvrs Flow, closing / Refueling		6.3-1
PXS-PL-V022A	Accumulator A Pressure Relief	Safety/Relief / Self-Actuated	C	Maintain Close		C	Class 2/3 RV Tests / Every 10 Years during Refueling		6.3-1
PXS-PL-V022B	Accumulator B Pressure Relief	Safety/Relief / Self-Actuated	C	Maintain Close		C	Class 2/3 RV Tests / Every 10 Years during Refueling		6.3-1
PXS-PL-V027A	Accum A Discharge Isolation	Gate / Motor	C	Maintain Open		B	Remote Postion Indicator / Refueling		6.3-1
PXS-PL-V027B	Accum B Discharge Isolation	Gate / Motor	C	Maintain Open		B	Remote Postion Indicator / Refueling		6.3-1
PXS-PL-V028A	Accumulator A Discharge Check	Check / Self-Actuated	A	Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling		6.3-1
PXS-PL-V028B	Accumulator B Discharge Check	Check / Self-Actuated	A	Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling		6.3-1
PXS-PL-V029A	Accumulator A Discharge Check	Check / Self-Actuated	A	Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling		6.3-1
PXS-PL-V029B	Accumulator B Discharge Check	Check / Self-Actuated	A	Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling		6.3-1
PXS-PL-V042	Nitrogen Supply Cont Isol	Globe / Diaphrag	B	Maintain Close	Active to Failed	• A	Remote Postion Indicator / Refueling		6.3-1

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
PXS-PL-V043	Nitrogen Supply Cont Isol IRC	Stop Check / Hand	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling Check Rvrs Flow, closing / Refueling		6.3-1
PXS-PL-V044	N2 Supply Header Press Relief	Safety/Relief / Self-Actuated				B	Class 2/3 RV Tests / Every 10 Years during Refueling		6.3-1
PXS-PL-V101A	PRHR HX A Inlet Isolation	Gate / Motor	A	Maintain Open		B	Remote Postion Indicator / Refueling		6.3-2
PXS-PL-V101B	PRHR HX B Inlet Isolation	Gate / Motor	A	Maintain Open		B	Remote Postion Indicator / Refueling		6.3-2
PXS-PL-V103A	PRHR HX A Outlet Isol	Gate / Hand	A	Maintain Open		B	Remote Postion Indicator / Refueling		6.3-2
PXS-PL-V103B	PRHR HX B Outlet Isol	Gate / Hand	A	Maintain Open		B	Remote Postion Indicator / Refueling		6.3-2
PXS-PL-V108A	PRHR HX Discharge Control	Globe / Diaph w/ Pos	A	Maintain Open Transfer Open	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.3-2
PXS-PL-V108B	PRHR HX Discharge Control	Globe / Diaph w/ Pos	A	Maintain Open Transfer Open	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		6.3-2
PXS-PL-V117A	Rercirc Sump A Isolation	Gate / Motor	C	Maintian Open Maintain Close Transfer Open	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		6.3-2
PXS-PL-V117B	Recirc Sump B Isolation	Gate / Motor	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		6.3-2
PXS-PL-V118A	Recirc Sump A Isolation	Gate / Motor	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		6.3-2
PXS-PL-V118B	Recirc Sump B Isolation	Gate / Motor	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly		6.3-2
PXS-PL-V119A	Recirc Sump A Check	Check / Self-Actuated	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling		6.3-2
PXS-PL-V119B	Recirc Sump B Check	Check / Self-Actuated	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling		6.3-2
PXS-PL-V120A	Recirc Sump A Check	Check / Self-Actuated	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling		6.3-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
PXS-PL-V120B	Recirc Sump B Check	Check / Self-Actuated	C	Maintain Open Maintain Close Transfer Open	Active Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling		6.3-2
PXS-PL-V121A	IRWST Injection A Isol	Gate / Motor	C	Maintain Open Transfer Open	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Refueling		6.3-2
PXS-PL-V121B	IRWST Injection B Isol	Gate / Motor	C	Maintain Open Transfer Open	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Refueling		6.3-2
PXS-PL-V122A	IRWST Injection A Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V122B	IRWST Injection B Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V123A	IRWST Injection A Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V123B	IRWST Injection B Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
PXS-PL-V124A	IRWST Injection A Check	Check / Self-Actuated	C	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Position Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V124B	IRWST Injection B Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Position Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V125A	IRWST Injection A Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Position Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V125B	IRWST Injection B Check	Check / Self-Actuated	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Position Indicator / Refueling Check-Fwd Flow, full open / Refueling Check-Initial Open D/P / Refueling	6	6.3-2
PXS-PL-V301A	pH Adjustment Tank Disch Isol	Squib / Explosive	C	Maintain Open Transfer Open	Active Remote Position	D	Remote Position Indicator / Every 10 Years during Refueling Exercise Full Stroke - Rflg / Every 10 Years during Refueling Charge Test Fire / Refueling		6.3-4
PXS-PL-V301B	pH Adjustment Tank Disch Isol	Squib / Explosive	C	Mainta[illegible] Transfe[illegible]	Active Remote Position	BD	Remote Position Indicator / Every 10 Years during Refueling Exercise Full Stroke - Rflg / Every 10 Years during Refueling Charge Test Fire / Refueling		6.3-4
PXS-PL-V307	pH Tank N2 Supply Press Relief	Safety/Relief / Self-Actuated	C			C	Class 2/3 RV Tests / Every 10 Years during Refueling		6.3-4
PXS-PL-V311	pH Tank Pressure Relief	Safety/Relief / Self-Actuated	C			C	Class 2/3 RV [illegible] / Every 10 Years during Refueling		6.3-4

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
PXS-PL-V315A	pH Tank Vacuum Breaker	Vac Brkr / Self-Actuated	C	Maintain Open Transfer Open	Active	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		6.3-4
PXS-PL-V315B	pH Tank Vacuum Breaker	Vac Brkr / Self-Actuated	C	Maintain Open Transfer Open	Active	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		6.3-4
RCS-PL-V001A	First Stage ADS	TBD / Motor	A	Maintain Open Maintain Close Transfer Open Transfer Close Throttle Flow	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1, 5	5.1-5
RCS-PL-V001B	First Stage ADS	TBD / Motor	A	Maintain Open Maintain Close Transfer Open Transfer Close Throttle Flow	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1, 5	5.1-5
RCS-PL-V002A	Second Stage ADS	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1, 5	5.1-5
RCS-PL-V002B	Second Stage ADS	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1, 5	5.1-5
RCS-PL-V003A	Third Stage ADS	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1, 5	5.1-5
RCS-PL-V003B	Third Stage ADS	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1, 5	5.1-5
RCS-PL-V004A	Fourth Stage ADS	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling Charge Test Fire / Refueling	2, 4, 5, 7	5.1-5
RCS-PL-V004B	Fourth Stage ADS	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling Charge Test Fire / Refueling	2, 4, 5, 7	5.1-5

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
RCS-PL-V004D	Fourth Stage ADS	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling Charge Test Fire / Refueling	2 4 5 7	5.1-5
RCS-PL-V005A	Pressurizer Safety Valve	Safety/Relief / Self-Actuated	A	Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Every 5 Years during Refueling Class 1 RV Tests / Every 5 Years during Refueling		5.1-5
RCS-PL-V005B	Pressurizer Safety Valve	Safety/Relief / Self-Actuated	A	Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	BC	Remote Postion Indicator / Every 5 Years during Refueling Class 1 RV Tests / Every 5 Years during Refueling		5.1-5
RCS-PL-V010A	ADS Dschg Hdr A Vacuum Relief	Vac Brkr / Self-Actuated	C	Transfer Open	Active	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		5.1-5
RCS-PL-V010B	ADS Dschg Hdr B Vacuum Relief	Vac Brkr / Self-Actuated	C	Transfer Open	Active	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		5.1-5
RCS-PL-V011A	First Stage ADS Isolation	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1 5	5.1-5
RCS-PL-V011B	First Stage ADS Isolation	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1 5	5.1-5
RCS-PL-V012A	Second Stage ADS Isolation	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1 5	5.1-5
RCS-PL-V012B	Second Stage ADS Isolation	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1 5	5.1-5
RCS-PL-V013A	Third Stage ADS Isolation	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1 5	5.1-5

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
RCS-PL-V013B	Third Stage ADS Isolation	TBD / Motor	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	1 5	5.1-5
RCS-PL-V014A	Fourth Stage ADS Isolation	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Every 5 Years Operability Test, Part D/P / Every 5 Years during Refueling	3 5 7	5.1-5
RCS-PL-V014B	Fourth Stage ADS Isolation	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Every 5 Years during Refueling Operability Test, Part D/P / Refueling	3 5 7	5.1-5
RCS-PL-V014C	Fourth Stage ADS Isolation	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years during Refueling	3 5 7	5.1-5
RCS-PL-V014D	Fourth Stage ADS Isolation	TBD / TBD	A	Maintain Open Maintain Close Transfer Open	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / See Notes Operability Test, Part D/P / Every 5 Years	3 5 7	5.1-5
RCS-PL-V152	RCS Head Vent	Globe / Solenoid	B	Maintain Open Maintain Close Transfer Open Transfer Close	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		5.1-5
RCS-PL-V153	RCS Head Vent	Globe / Solenoid	B	Maintain Open Maintain Close Transfer Open Transfer Close	Active RCS Press Boundary Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		5.1-5
RCS-PY-K001A	SV Dschrge Chmb Rupture Disk	TBD / Self-Actuated	C	Transfer Open	Active	B	Replace Cl 2/3 Rupt Disk / Every 5 Years		5.1-5
RCS-PY-K001B	SV Dschrge Chmb Rupture Disk	TBD / Self-Actuated	C	Transfer Open	Active	B	Replace Cl 2/3 Rupt Disk / Every 5 Years		5.1-5

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
RNS-PL-V001A	RCS Inner HL Suction Isolation	Gate / Motor	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling RCS Press Bound leak test / Cold Shutdown Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		5.4-7
RNS-PL-V001B	RCS Inner HL Suction Isolation	Gate / Motor	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling RCS Press Bound leak test / Cold Shutdown Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		5.4-7
RNS-PL-V002A	RCS Outer HL Suction Isolation	Gate / Motor	A	Maintain Close Transfer Close	Active RCS Press Boundary Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling RCS Press Bound leak test / Cold Shutdown Exercise Full Stroke - Rflg / Refueling Operability Test, Part D/P / Every 5 Years during Refueling		5.4-7
RNS-PL-V002B	RCS Outer HL Suction Isolation	Gate / Motor	A	Maintain Close Transfer Close	Active RCS Press Boundary Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling RCS Press Bound leak test / Cold Shutdown Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years during Refueling		5.4-7
RNS-PL-V003A	RCPB Iso. Vlv. Thermal Relief	Stop Check / Hand	A	Maintain Close Transfer Open	Active RCS Press Boundary Safety Seat Leakage	A	RCS Press Bound leak test / Refueling Check-Fwd Flow, full open / Refueling		5.4-7
RNS-PL-V003B	RCPB Iso. Vlv. Thermal Relief	Stop Check / Hand	A	Maintain Close Transfer Open	Active RCS Press Boundary Safety Seat Leakage	A	RCS Press Bound leak test / Refueling Check-Fwd Flow, full open / Refueling		5.4-7
RNS-PL-V011	RNS Control/Containment Isol.	Globe / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly		5.4-7
RNS-PL-V013	RNS Discharge Containment Iso.	Check / Self-Actuated	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		5.4-7
RNS-PL-V015A	RNS Discharge RCPB Isolation	Check / Self-Actuated	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage	AC	RCS Press Bound leak test / Refueling Check-Rvrs Flow, closing / Refueling		5.4-7
RNS-PL-V015B	RNS Discharge RCPB Isolation	Check / Self-Actuated	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage	AC	RCS Press Bound leak test / Refueling Check-Rvrs Flow, closing / Refueling		5.4-7

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig No.
RNS-PL-V017A	RNS Discharge RCPB Isolation	Check / Self-Actuated	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage	AC	RCS Press Bound leak test / Refueling Check-Rvrs Flow, closing / Refueling		5.4-7
RNS-PL-V017B	RNS Discharge RCPB Isolation	Check / Self-Actuated	A	Maintain Close Transfer Close	Active RCS Press Boundary Safety Seat Leakage	AC	RCS Press Bound leak test / Refueling Check-Rvrs Flow, closing / Refueling		5.4-7
RNS-PL-V021	RNS HL Suction Pressure Relief	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open Transfer Close	Active Remote Position	BC	Remote Position Indicator / Refueling Class 2/3 RV Tests / Every 10 Years during Refueling		5.4-7
RNS-PL-V022	RNS Suction Header Cont. Isol.	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling		5.4-7
RNS-PL-V023	RNS Suct. from IRWST Cont. Iso	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling		5.4-7
SFS-PL-V034	SFS Suction Line Cont. Isol.	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling		9.1-8
SFS-PL-V035	SFS Suction Line Cont. Isol.	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling		9.1-8
SFS-PL-V037	SFS Disch. Line Cont. Isol.	Check / Self-Actuated	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage	AC	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.1-8
SFS-PL-V038	SFS Disch. Line Cont. Isol.	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling		9.1-8
SFS-PL-V048	SFS Suction CIV Thermal Relief	Stop Check / Hand	B	Maintain Close Transfer Open Transfer Close	Active Containment Isolation Safety Seat Leakage	A	Cont Isol leak test / Refueling Check-Rvrs Flow, closing / Refueling		9.1-8
SGS-PL-V027A	PORV Block Valve SG 01	Gate / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Part D/P / Every 5 Years during Refueling		10.3.2-1
SGS-PL-V027B	PORV Block Valve SG 02	Gate / Motor	B	Maintain Close	Active	B	Remote Position Indicator / Refueling		10.3.2-1

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
SGS-PL-V030A	Main Steam Safety Valve SG 01	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open	Active Containment Isolation	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		10.3.2-1
SGS-PL-V030B	Main Steam Safety Valve SG 02	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open	Active Containment Isolation	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		10.3.2-1
SGS-PL-V031A	Main Steam Safety Valve SG 01	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open	Active Containment Isolation	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		10.3.2-1
SGS-PL-V031B	Main Steam Safety Valve SG 02	Safety/Relief / Self-Actuated	B	Transfer Open Transfer Close	Active Containment Isolation	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		10.3.2-1
SGS-PL-V032A	Main Steam Safety Valve SG 01	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open	Active Containment Isolation	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		10.3.2-1
SGS-PL-V032B	Main Steam Safety Valve SG 02	Safety/Relief / Self-Actuated	B	Maintain Close Transfer Open	Active Containment Isolation	BC	Class 2/3 RV Tests / Every 10 Years during Refueling		10.3.2-1
SGS-PL-V036A	Steam Line Cond Drain Isolatio	Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V036B	Steam Line Cond Drain Isolatio	Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
SGS-PL-V040A	Main Steam Line Isolation	Gate / Piston, Dbl Ac	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Part Stroke - Pwr / Quarterly Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling	8	10.3.2-1
SGS-PL-V040B	Main Steam Line Isolation	Gate / Piston, Dbl Ac	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Part Stroke - Pwr / Quarterly Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling	8	10.3.2-1
SGS-PL-V057A	Main Feedwater Isolation	Gate / Piston, Dbl Ac	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Part Stroke - Pwr / Quarterly Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling	8	10.3.2-1
SGS-PL-V057B	Main Feedwater Isolation	Gate / Piston, Dbl Ac	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Part Stroke - Pwr / Quarterly Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling	8	10.3.2-1
SGS-PL-V067A	Startup Feedwater Isolation	Stop Check / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Operability Test, Full D/P / Every 5 Years		10.3.2-1
SGS-PL-V067B	Startup Feedwater Isolation	Stop Check / Motor	B	Maintain Close Transfer Close	Active Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Operability Test, Full D/P / Every 5 Years		10.3.2-1
SGS-PL-V074A	SG Blowdown Isolation	Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V074B	SG Blowdown Isolation	Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V075A	SG Series Blowdown Isolation	Globe / Diaphragm	C	Maintain Close Transfer Close	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / [illegible]		10.3.2-1
SGS-PL-V075B	SG Series Blowdown Isolation	Globe / Diaphragm	C	Maintain Close Transfer Close	Active to Failed Remote Position	B	Remote Postion Indicator / [illegible] Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
SGS-PL-V233A	Power Operated Relief Valve	Globe / Diaph w/ Pos	C	Maintain Close Transfer Close	Active to Failed	B	Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V233B	Power Operated Relief Valve	Globe / Diaph w/ Pos	C	Maintain Close Transfer Close	Active to Failed	B	Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V240A	MSIV Bypass Isolation	Globe / Diaph w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V240B	MSIV Bypass Isolation	Globe / Diaph w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Remote Position	B	Remote Postion Indicator / Refueling Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V250A	Main Feedwater Control	Globe / Diaph w/ Pos	C	Maintain Close Transfer Close	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Part Stroke - Pwr / Quarterly Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V250B	Main Feedwater Control	Globe / Diaph w/ Pos	C	Maintain Close Transfer Close	Active to Failed Remote Position	B	Remote Postion Indicator / Refueling Exercise Part Stroke - Pwr / Quarterly Exercise Full Stroke - S/D / Cold Shutdown Operability Test, Part D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V255A	Startup Feedwater Control	Globe / Diaph w/ Pos	C	Maintain Close Transfer Close	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
SGS-PL-V255B	Startup Feedwater Control	Globe / Diaph w/ Pos	C	Maintain Close Transfer Close	Active Remote Position	B	Remote Postion Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		10.3.2-1
VES-PL-V002A	Pressure Regulating Valve A	Globe / Process Fluid	C	Throttle Flow	Active	B	Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years		6.4-2
VES-PL-V002B	Pressure Regulating Valve B	Globe / Process Fluid	C	Throttle Flow	Active	B	Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years		6.4-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
VES-PL-V003A	Relief Valve A	Safety/Relief / Self-Actuated	C			C	Class 2/3 RV Tests / Every 5 Years		6.4-2
VES-PL-V003B	Relief Valve B	Safety/Relief / Self-Actuated	C			C	Class 2/3 RV Tests / Every 5 Years		6.4-2
VES-PL-V004A	Refill Connection Isolation A	Globe / Hand	C	Transfer Open	Active	B	Exercise Full Stroke - S/D / Cold Shutdown		6.4-2
VES-PL-V004B	Refill Connection Isolation B	Globe / Hand	C	Transfer Open	Active	B	Exercise Full Stroke - S/D / Cold Shutdown		6.4-2
VES-PL-V005A	Outlet Isolation Valve A	Globe / Solenoid	C	Transfer Open	Active to Failed Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		6.4-2
VES-PL-V005B	Outlet Isolation Valve B	Globe / Solenoid	C	Transfer Open	Active to Failed Remote Position	B	Remote Position Indicator / Refueling Exercise Full Stroke - Pwr / Quarterly Operability Test, Full D/P / Every 5 Years Fail Safe / Refueling		6.4-2
VFS-PL-V003A	Cont. Air Filter Supply A	Butterfly / Piston w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VFS-PL-V003B	Cont. Air Filter Supply B	Butterfly / Piston w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VFS-PL-V004A	Cont. Air Filter Supply A	Butterfly / Piston w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VFS-PL-V004B	Cont. Air Filter Supply B	Butterfly / Piston w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VFS-PL-V009A	Cont. Air Filter Exhaust A	Butterfly / Piston w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VFS-PL-V009B	Cont. Air Filter Exhaust B	Butterfly / Piston w/ Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig No.
VFS-PL-V010A	Cont. Air Filter Exhaust A	Butterfly / Piston w/Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VFS-PL-V010B	Cont. Air Filter Exhaust B	Butterfly / Piston w/Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		9.4.7-1
VWS-PL-V058	Fan Coolers Supply Cont Isolat	Butterfly / Piston w/Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		9.2.7-1
VWS-PL-V062	Fan Coolers Supply Cont Isolat	Butterfly / Piston w/Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		9.2.7-1
VWS-PL-V082	Fan Coolers Return Cont Isolat	Butterfly / Piston w/Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		9.2.7-1
VWS-PL-V086	Fan Coolers Return Cont Isolat	Butterfly / Piston w/Pos	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		9.2.7-1
WLS-PL-V004	RCDT Containment Isolation IRC	Packless Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		11.2-2
WLS-PL-V006	RCDT Containment Isolation ORC	Packless Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		11.2-2
WLS-PL-V055	Sump Containment Isolatio	Packless Globe /	B	Maintain Close	Active to Failed	[illegible]	Remote Position Indicator / Refueling		11.2-2

Table 3.9.6-1

Valve Inservice Test Requirements

Tag Number	Description	Valve Type/Actr	AP600 Equip Class	Safety Related Missions	Safety Related Functions	ASME IST Category	Inservice Testing Type / Frequency	IST Notes	SSAR Fig. No.
WLS-PL-V057	Sump Containment Isolation ORC	Packless Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Fail Safe / Refueling		11.2-2
WLS-PL-V067	RCDT Gas Containment Isolation	Packless Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Pwr / Quarterly Fail Safe / Refueling		11.2-2
WLS-PL-V068	RCDT Gas Containment Isolation	Packless Globe / Diaphragm	B	Maintain Close Transfer Close	Active to Failed Containment Isolation Safety Seat Leakage Remote Position	A	Remote Position Indicator / Refueling Cont Isol leak test / Refueling Exercise Full Stroke - Rflg / Refueling Fail Safe / Refueling		11.2-2

Table 3.9.6-1

Valve Inservice Test Requirements

Standard Notes Reference

Number	Note
1	Valve body type will be determined outside of design certification during valve procurement. IST requirements illustrate two examples of valve body types (gate and globe).
2	Valve type will be determined outside of design certification during valve procurement. IST requirements illustrate two examples of valve types (air-piston gate and squib).
3	Valve type will be determined outside of design certification during valve procurement. IST requirements illustrate two examples of valve types (air-piston gate and motor-operated gate).
4	Charge Test Fire / Refueling applies only if squib valves are selected for this valve application. Other tests apply only if air-piston valves selected for this application.
5	The frequency for stroke testing ADS valves will be determined based on a probabilistic safety assessment - quarterly testing was assumed for 6/92 PRA report.
6	Forward flow testing together with check valve position indication verifies RCS pressure boundary integrity.
7	During valve fail safe testing the pnematic accumulator is leak tested.
8	During valve fail safe testing the pnematic operator is leak tested.

Question 210.33

The response to Q210.15 dated December 22, 1992 requires more detailed information relative to the methods for verification of computer programs. Section 3.9.1.2 of the SSAR and the response to Q210.15 both reference Chapter 17 of the SSAR, "Quality Assurance," for this information. However, Chapter 17 does not contain the level of detail that the staff is seeking. As a minimum, the staff requests that each program used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-code items for the AP600 be verified by at least one of the following methods:

1. Hand calculations
2. Analytical results published in the literature
3. Acceptable experimental tests
4. Results from a similar program previously endorsed by the staff
5. Comparison with the benchmark problems in NUREG/CR 1677, "Piping Benchmark Problems."

The following programs listed in Table 3.9.15 of the SSAR have been reviewed and endorsed by the staff:

ABAQUS	Finite element structural analysis
ANSYS	Finite element structural analysis
GAPPIPE	Static and dynamic analysis of piping systems
WECAN	Finite element structural analysis
Westdyn	Static and dynamic analysis of piping systems

For the remainder of the programs in Table 3.9.15 and all other applicable programs that will be listed in the ASME Code Design Reports, revise Section 3.9.1.2 of the SSAR to identify one or more of the above verification methods. In addition, delete the exception to SRP 3.9.1, Section II.2, in Revision 1 to WCAP-13054.

Response:

The Quality Assurance plan applicable to the verification of computer programs is contained in WCAP-8370 as discussed in Chapter 17 of the SSAR. Westinghouse internal procedures developed to implement the commitments described in WCAP-8370 provide for the verification of both Westinghouse developed and externally developed computer codes. This verification must be performed by at least one of the following methods.

a. Hand calculations
b. Alternate verified calculational methods
c. Results of other verified programs
d. Results obtained from experiments and tests
e. Known solutions for similar or standard problems
f. Measured and documented plant data
g. Confirmed published data and correlations
h. Results of standard programs and benchmarks
i. Parametric sensitivity analysis

j. Reference to a verification and validation that has been reviewed and accepted by an independent third party.

Other applicable programs that will be listed in the ASME Code Design Reports will be added to Table 3.9-15 as shown below.

The exception to Standard Review Plan 3.9.1, Section II.2, will be revised in the next revision of WCAP-13054

SSAR Revision:

Revise Subsection 3.9.1.2 as follows:

3.9.1.2 Computer Programs Used in Analyses

A number of computer programs that are used in the dynamic and static analyses of mechanical loads, stresses, and deformations of seismic Category I components and supports, **and in the hydraulic transient load analyses**, are listed in Table 3.9-15. A complete list of programs will be included in the ASME Code Design Reports.

The development process, verification, validation, configuration control and error reporting and resolution for computer programs used in these analyses for the AP600 are completed in compliance with an established quality assurance program. The quality assurance program is described in Chapter 17. **The verification must conform to at least one of the following methods:**

- **Hand calculations**
- **Alternate verified calculational methods**
- **Results of other verified programs**
- **Results obtained from experiments and tests**
- **Known solutions for similar or standard problems**
- **Measured and documented plant data**
- **Confirmed published data and correlations**
- **Results of standard programs and benchmarks**
- **Parametric sensitivity analysis**
- **Reference to a verification and validation that has been reviewed and accepted by an independent third party.**

Revise Table 3.9-15 to read as follows:

Table 3.9-15

Computer Programs for Seismic Category 1 Components

Program	Application
ABAQUS	Finite element structural analysis
ANSYS	Finite element structural analysis
CAEPIPE	Static analysis of piping analysis
FATCON	ASME fatigue analysis of piping components
GAPPIPE	Static and dynamic analysis of piping systems
MAXTRAN	Transient stress evaluation of piping components
PIPSAN	Structural and ASME stress analysis of component supports
PS+CAEPIPE	Static and dynamic analysis of piping systems
STAAD-III	Static and dynamic analysis of structural frames
THERST	Transient heat transfer analysis of piping components
WECAN	Finite element structural analysis
WEGAP	Dynamic structural response of the reactor core
WECEVAL	ASME stress evaluation of mechanical components
WESTDYN	Static and dynamic analysis of piping systems
ITCH	Transient hydraulic analysis
FORFUN	Computes unbalanced hydraulic forces between piping elbows
RELAP-5	Transient dynamic analysis
THRUST	Computes time-history hydraulic forcing functions
MULTIFLEX	Thermal-hydraulic-structural system analysis
MULTIFLEX-SG	Transient dynamic analysis
GEC2	Computes time-history hydraulic forcing functions
FATSTR	ASME stress evaluation of piping components

Question 210.42

Sections 3.6.2.3.2, 3.9.3.4, and 3.10.1.3 of the SSAR mention an analysis approach for transient loading conditions and Service Level D conditions that allows a limited number of pipe supports to fail, provided that the consequences of these failures are evaluated and that adequate support exists for deadweight and steady state pressure conditions following the event. Since these are ASME Class 1, 2, or 3 supports, they are designed to ASME Subsection NF, and the loading combinations in Table 3.9-8 of the SSAR and, therefore, should withstand Service Level D loads without failure. What is the advantage of postulating failures of such supports? Provide a more detailed discussion of this procedure and how it will be implemented.

Response:

There will be no support failure for service level D conditions. The approach will be deleted from the SSAR

SSAR Revision:

See response to Question 210.68 for revisions to Subsection 3.9.3.4

Revise the second paragraph of Subsection 3.6.2.3.2 as follows:

~~During the transient loading period a limited number of pipe supports may be permitted to fail provided that adequate support exists for deadweight and steady state pressure conditions.~~

Revise the fourth paragraph of Subsection 3.6.2.3.2 as follows:

These analyses may consider nonlinear geometric and material characteristics of the piping system. ~~Supports that may fail are designed in such a way that a local pipe pressure boundary failure does not occur at the support connection point.~~

Revise the first paragraph of Subsection 3.10.1.3 as follows:

Seismic and dynamic loading qualification demonstrates that Category I instrumentation and electrical equipment and active valves and dampers are capable of performing their designated safety-related functions under applicable plant loading conditions, including the safe shutdown earthquake. The qualification also demonstrates the structural integrity of seismic Category I nonactive valves, mechanical supports, and structures. Some permanent deformation of supports and structures is acceptable at the safe shutdown earthquake level, provided that the capability to perform the designated safety-related functions is not impaired. **In this case the system analysis will be done using inelastic behavior of the piping and supports (see subsection 3.9.3.4)**

Question 210.52

Section 3.9.1.1 of the SSAR, "Design Transients," discusses pressure, temperature, and flow transients, but does not include seismic events. The last paragraph in this section states that where applicable, in addition to the effects produced by the above transients, earthquake loadings must be considered, and references Section 3.9.3 for a description of how these loads are considered for the AP600. To be consistent with the guidelines in Section 3.9.1 of the SRP, Section 3.9.1 and Table 3.9-1 of the SSAR should include seismic events as one of these transients. Section IM, "Elimination of OBE," in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," dated Apr'' 2, 1993, presents the current staff position relative to accounting for earthquake cycles in fatigue analyses. Revise Section 3.9.1 and Table 3.9-1 to include the seismic events and the number of cycles consistent with this staff position.

Response:

The guidelines in Regulatory Guide 1.70, and the Standard Review Plan recommend that transients resulting from seismic events be included in the design transients in Subsection 3.9.1. The AP600 is designed for seismic loads and has identified no fluid system transients resulting from seismic events. Seismic events are loading conditions and are discussed in Subsection 3.9.3 with the other loading conditions. The final paragraph of subsection 3.9.1.1 notes that the seismic loads are described in Subsection 3.9.3. Section 3.9.1 and Table 3.9-1 will not be revised to include seismic loading conditions.

The discussion of seismic loading conditions in Subsection 3.9.3 and the tables in Section 3.9 that define the load combinations used for analysis of seismic loads are revised in other RAI responses in response to the NRC position on elimination of operating basis earthquake as a loading condition. The effect of earthquake cycles will be considered in the fatigue analyses. See RAIs 210.60, 210.79 and 210.80 for additional discussion of these issues.

SSAR Revision:

See responses to RAIs 210.79, and 210.80 for SSAR revisions related to NRC position on elimination of operating basis earthquake as a loading condition.

Question 210.60

Section 3.7 of the SSAR states that the operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Section IM, "Elimination of OBE," in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," dated April 2, 1993 contains the staff's recommendations to the Commission relative to this issue. The staff has evaluated the impact of this proposal, and has identified the necessary changes to the current seismic design criteria and the appropriate technical actions necessary for Westinghouse to implement these changes for the AP600. The current staff positions relative to these changes are discussed in the attachment to this enclosure. Revise applicable portions of Section 3.9 of the SSAR to implement these positions.

Response:

See the response to RAI 210.79 for a discussion of this issue.

SSAR Revision:

See the response to RAI 210.79 for SSAR Revisions.

Question 210.62

To be consistent with Section 3.9.3 of the SRP, Appendix A, Table 1, the staff's position is that Table 3.9.3-7 of the SSAR, "Minimum Design Loading Combinations for ASME Class 2 and 3 Piping," and Table 3.9.3-8, "Minimum Design Loading Combinations for Supports for ASME Class 1, 2, and 3, Piping and Components," should include SSE + DF in the loading combinations for the Level D condition for all Class 1, 2, and 3 components. Revise Table 3.9.3-7 to add this combination, and revise Table 3.9.3-8 to delete Note 3. Delete Note 7 to Table 3.9-6, if applicable. In addition, revise the exception to Section 3.9.3 of the SRP, Appendix A, Section C.1.2 in WCAP-13054, as applicable.

Response:

See the response to RAI 210.79 for a discussion of this issue.

SSAR Revision:

See the response to RAI 210.79 for SSAR revisions.

Question 210.65

Section 3.9.3.1.7 of the SSAR states that no special stress limits are required to provide functional capability of ASME Class 2 and 3 piping. The current staff position on this issue is documented in NUREG-1367, "Functional Capability of Piping Systems," dated November 1992. Revise applicable portions of the SSAR to commit to the positions in NUREG-1367 for all seismic Category I piping.

Response:

See the response to RAI 210.79 for a discussion of this issue.

SSAR Revision:

See the response to RAI 210.79 for SSAR revisions.

Question 210.70

Section 3.9.5.2.4 of the SSAR states that the AP600 core barrel, upper and lower support plates, support columns, and radial key supports are considered core support structures and constructed to ASME Subsection NG. It further states that for other internal structures, Article NG-3000, "Design," does not specifically apply and that these other internals are designed and fabricated using the ASME Code as a guide. In Sheet 38 of Table 3.2-3 of the SSAR, all of the safety-related reactor internals are listed as AP600 Class C and the principal construction code is identified as ASME III, CS for all internals. This implies that all internals are constructed to ASME Subsection NG, which does not agree with the statement in Section 3.9.5.2.4 of the SSAR relative to other internals. If internal structures other that those identified as core support structures will not be constructed to NG-3000, revise this section to provide a more detailed description of the design criteria that is used for such items. Include a discussion describing how selected code rules and other requirements are used together to ensure structural adequacy and functionality of various internal structures at various conditions. In addition, revise Table 3.2-3 to be consistent with the revised Section 3.9.5.2.4.

Response:

Section 3.9.5.2.4 is entitled: " Design Loading Categories", consequently, the discussion of classification of core support structures is inappropriate for this Section. Therefore, the second paragraph of Section 3.9.5.2.4 should be deleted. The appropriate section for classification of reactor internals should be in Table 3.2-3 (Sheet 38 of 107). That table will be revised as shown below. Table 3.2-3 has been expanded to include core support structures and internal structures and the method or principal construction code used in manufacturing and analysis.

SSAR Revision:

Revise Section 3.9.5.2.4 as follows:

The combination of design loadings fit into either the service level A, B, C or D conditions shown on Figures ~~NG-3221-1~~ NG-3221-1, ~~NG-3224-1~~ NG-3224-1, NG-3232-1 and by Appendix F of the ASME Code, Section III.

~~The AP600 core barrel, upper and lower core support plates, support columns, and radial key supports are considered core support structures. Core support structures are certified to the requirements of Subsection NG of the ASME Code. For other internal structures, Article NG-3000 does not specifically apply. These other internals structures are designed and fabricated using the ASME Code as a guideline.~~

AP600

Revise Table 3.2-3 (Sheet 38 of 107) as follows:

Table 3.2-3 (Sheet 38 of 107)

Component Description	Loc	AP600 Class	Seismic Category	Principal Construction Code	Comments
Lower core supp plate	11		I	ASME III, CS	
Secondary core support	11	~~C~~ D	~~I~~ NS	ASME III, CS	Note 1
Vortex suppression plate	11	D	~~II~~ NS	ASME III, CS	Note 1
Radial reflector	11	~~C~~ D	~~I~~ NS	ASME III, CS	Note 1
Radial supports	11	C	I	ASME III, CS	
Upper support	11	C	I	ASME III, CS	
Upper core plate	11	C	I	ASME III, CS	
Support columns	11	C	I	ASME III, CS	
Guide tube assemblies	11	D	~~II~~ NS	ASME III, CS	Note 1
Core barrel	11	C	I	ASME III, CS	
Hold down spring	11	D	NS	ASME III, CS	Note 1
Core barrel nozzle	11	D	NS	ASME III, CS	Note 1
Head and vessel pins	11	D	NS	ASME III, CS	Note 1
Fuel alignment pins	11	C	I	ASME III, CS	
Upper core plate inserts	11	C	I	ASME III, CS	
Safety injection deflector	11	D	NS	ASME III, CS	Note 1
Irradiation specimen guide	11	D	NS	ASME III, CS	Note 1
Head cooling nozzles	11	D	NS	ASME III, CS	Note 1
Threaded struc fasteners	11	C	I	ASME III, CS	
Fasteners	11	D	NS	ASME III, CS	Note 1

Notes:

1. Part is an internal structure, not a core support sructure. Although not a Code requirement, AP600 internal structures are designed to ASME Code, Section III, Subsection NG.

Westinghouse

Question 210.73

The ASME Code requires that a design specification be prepared for all ASME Class 1, 2, and 3 components. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations and other design data inputs. The Code also requires a design report for all such components. In the past, as a part of its review of plants under construction, the staff reviewed documents related to design specifications and design reports for a small number of ASME Class 1, 2 and 3 pumps, valves, and piping systems. The staff intends to perform such a review for the first AP600 plant. The objective of this review will be to provide the staff with the basis for concluding that the AP600 design documentation meets the applicable requirements of ASME Section III, Subsection NCA. In the interim, either revise Section 3.9.3 of the SSAR, or submit a separate document referenced in the SSAR, to provide a detailed description of the procedures used for generating design specifications for procurement of ASME Class 1, 2 and 3 components. Include a specific commitment to state whether Westinghouse or the COL will provide the final documentation for the staff's review.

Response:

Westinghouse internal procedures are used for generating design specifications for procurement of ASME Class 1, 2 and 3 components and piping. These documents are available for staff review. Design Reports for ASME Class 1, 2 and 3 components and piping are prepared by the responsible AP600 design agent or vendor, utilizing their respective quality assurance program and procedures that meet ASME Code, Section III requirements. The Design Specifications and Design Reports must be completed by the Owner or Owner's agent prior to affixing an N-Stamp to any ASME Class 1, 2 or 3 components or piping systems. Design Specifications and Design Reports in progress are available at Westinghouse's facilities for staff review.

SSAR Revision:

Add the following paragraph to subsection 3.9.3.

The ASME Code, Section III requires that a design specification be prepared for ASME Class 1, 2 and 3 components. The specification conforms to and is certified to the requirements of ASME Code, Section III. The Code also requires a design report for safety related components, to demonstrate that the as-built component meets the requirements of the relevant ASME Design Specification and the applicable ASME Code. The Design Specifications and Design Reports will be completed by the Combined License applicant or his agent prior to affixing an N-Stamp to any ASME Class 1, 2 or 3 components or piping systems. Design Reports for ASME Class 1, 2 and 3 components and piping are prepared by the responsible AP600 design agent or vendor, utilizing procedures that meet the ASME Code.

Question 210.77

The table in Revision 1 to WCAP-13054 that addresses Section 3.6.2 of the SRP lists MEB 3-1, Sections B.1.c.(5) and B.3.c.(4) as acceptable for the AP600 design. Both of these guidelines relate to qualifying equipment for environmental (temperature, pressure, and humidity) effects. Several portions of Section 3.6.2 of the SSAR briefly mention requirements for considering environmental effects. For example, Section 3.6.2.1.1.4 provides a commitment to evaluate leakage cracks in main steam and feedwater lines in the containment penetration area. However, Section 3.6.2 does not appear to contain any detailed discussion relative to the guidelines in the two MEB 3-1 sections. Revise Section 3.6.2 of the SSAR to include a commitment to these guidelines and provide a description of how environmental effects will be considered in the AP600 design of high and moderate energy piping systems.

Response:

SSAR Section 3.6.2 references the sections of the SSAR that provide detailed discussion relative to the guidelines in the two MEB 3-1 sections as summarized below.

- MEB 3-1, Section B.1.c.(5) for high energy piping requires that "safety-related equipment must be environmentally qualified in accordance with Standard Reviwew Plan 3.11. Required pipe ruptures and leakage cracks (whichever controls) must be included in the design bases for environmental qualification of electrical and mechanical equipment both inside and outside the containment."

  The environmental qualification of equipment is described in SSAR Section 3.11. Environmental conditions (temperature, pressure, and humidity) are specified in Appendix 3D. These environmental conditions are based on the LOCA and main steamline break inside containment which envelope other postulated breaks inside containment. High energy piping outside containment is identified in SSAR Appendix 3F. It includes piping in the mainsteam, feedwater, steam generator blowdown, and chemical and volume control systems. Environmental temperatures, specified in Appendix 3D for the main steam isolation valve area, are based on the one square foot break in the mainsteam line. Environmental conditions will be added in Appendix 3D by December, 1994 for the valve room in which the steam generator blowdown piping is located. Environmental conditions need not be specified for the rooms housing the chemical and volume control piping since this portion of the chemical and volume control system is a high pressure cold system. Subsections 3.6.2.1, 3.6.2.2, and 3.6.2.1.1.4 contain the commitments on environmental effects.

- MEB 3-1, Section B.3.c.(4) for high and moderate energy piping requires that "the flow from a leakage crack should be assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period required to take corrective actions." Subsection 3.6.2.1.3.2 is revised as shown below.

  Flooding effects are described in SSAR Section 3.4. Spray effects, including wetting of unprotected equipment, will be included in SSAR Section 3.4, Revision 2.

SSAR Revision:

Revise the first paragraph of Subsection 3.6.2.1.3.2 as follows:

High-and moderate-energy through-wall crack openings are assumed to be a circular orifice with cross-sectional flow area equal to that of a rectangle one-half the pipe inside diameter in length and one-half pipe wall thickness in width. The flow from a leakage crack is assumed to result in an environment that wets all unprotected components within the compartment with consequent flooding in the compartment and communicating compartments, unless analysis shows otherwise. Flooding effects are determined on the basis of a conservatively estimated time period required to take corrective actions.

Question 210.80

Revision 1 to WCAP-13054 lists an exception to Section C.3.2 of Appendix A to Section 3.9.3 of the SSAR, that states that one-half SSE is evaluated to level C limits. This does not appear to be consistent with the staff's current position relative to the use of a single-earthquake design for the AP600. The attachment to Q210.60 contains this position. Revise the above exception to Section 3.9.3 of the SRP to be consistent with this staff position.

Response:

The response to RAI 210.79 discusses the position on the use of single earthquake design and includes SSAR revisions including loading combination revisions to implement the position. The SSAR revisions shown below implement the position for fatigue analyses. For AP600 the option of using 20 full cycles of the maximum safe shutdown earthquake (SSE) stress range or the alternative of five events each resulting in 63 full cycles with a magnitude equal to one third of the maximum SSE stress range will be used. The exception to Section C.3.2 of Appendix A to Section 3.9.3 of the Standard Review Plan will be revised in the next revision of WCAP-13054.

SSAR Revision:

Revise the third paragraph of Subsection 3.9.3.1 as follows:

The design transients for the AP600 are defined in Subsection 3.9.1. The transients are classified into Level A, B, C, and D Service conditions and test conditions, depending on the expected frequency of occurrence and severity. The description of the transients in Subsection 3.9.1 provides the initial plant operating condition and identifies any different component operating condition. The design transients for Levels A and B are used in the evaluation of cyclic fatigue for the Class 1 components and piping. **The effects of seismic events are also included in the evaluation of cyclic fatigue (See Subsection 3.9.3.1.2).** Level D and up to 25 strong stress cycles of Level C Service conditions are not required by the rules of the ASME Code to be included in the fatigue evaluation.

Revise the first paragraph of Subsection 3.9.3.1.1 as follows:

Seismic Category I ~~structures,~~ systems~~,~~ and **components, including core support structures,** ~~subsystems~~ are designed for one occurrence of the safe shutdown earthquake (SSE) which is evaluated as a Service Level D condition for pressure boundary integrity. In addition, ~~sub~~systems **and components** sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. These effects are considered by including **20 full cycles of the maximum safe shutdown earthquake stress range or** five seismic events, each resulting in 63 full stress cycles with a magnitude equal to one third ~~percent~~ of the calculated safe shutdown earthquake response for structures and components using linear elastic methods. ~~The seismic input to the subsystem is taken as 50 percent of the safe shutdown earthquake input for subsystems qualified by nonlinear analysis methods or by testing.~~

Revise the seventeenth and eighteenth paragraphs of Subsection 3.9.3.1.2 as follows:

To provide integrity for the reactor coolant system, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions. Generally only Level A and B service condition design transients are evaluated in the analysis of cyclic fatigue. Up to 25 stress cycles for Level C service conditions may be excluded from cyclic fatigue analysis in conformance with ASME Code, Section III criteria. Any Level C service conditions which are in excess of the 25-cycle limit are evaluated for the effect on cyclic fatigue using Level B criteria. **For the evaluation of cyclic fatigue, the cycles included for seismic events are evaluated using Level B criteria and are not excluded from the fatigue evaluation regardless of the size of the stress range considered.** The determination of which transients ~~and seismic events~~ are included in the 25-cycle exclusion is made separately for each component and line of piping.

The effects of seismic events on the design of components other than piping are considered in one of the following ways. **The effects of seismic events are considered by including 20 full cycles of the maximum safe shutdown earthquake stress range in the fatigue analysis.** ~~When the SSE event is evaluated as a Level C event, there is no seismic contribution to the fatigue evaluation. In this case the SSE accounts for 20 cycles, corresponding to two SSE events with 10 stress cycles per event, of the 25 Level C stress cycles.~~ Alternatively, ~~a seismic event with an amplitude equal to one-third of the SSE is evaluated as a Level B event. In this case,~~ the seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the SSE and 63 cycles per event. The seismic evaluation of piping components is discussed in Subsection 3.9.3.1.5.

Revise the eighth paragraph of Subsection 3.9.3.1.5 (ASME Piping) as follows:

The Level A and B service condition and test condition transients identified in Subsection 3.9.1.1 are included in the fatigue evaluation. For each thermal transient, two load-sets are defined representing the maximum and minimum stress states for that transient. **The effects of seismic events on the design of piping are considered in one of the following ways. The effects of seismic events are considered by including 20 full cycles of the maximum safe shutdown earthquake stress range in the fatigue analysis. Alternatively, the seismic contribution to the fatigue evaluation is based on five seismic events with an amplitude of one-third the SSE and 63 cycles per event.**

Question 210.95

Traditionally, the design of PWR internals is dominated by the LOCA loads due to postulating large breaks in the coolant loop. Thus, the internals have ample margins to resist an SSE, operation transients, and flow-induced vibrations, that generally induce less significant stress levels and deflections than that induced by the LOCA. Due to the application of leak-before-break (LBB), LOCA loads become less important in the internals design. In Section 3.9.2.5 of the SSAR, identify the largest LOCA used in the design of AP600 internals, and provide a discussion regarding how margins were maintained to ensure adequate defense of reactor internals against uncertainties of SSE and operational loads.

Response:

Consistent with past practice, the AP600 reactor internals are designed for the dynamic effects of 1 square foot hot leg and cold leg breaks. The AP600 core support structures and threaded structural fasteners analysis and ASME Section III Subsection NG allowable stresses ensure acceptability of the design loadings for Level A, Level B, Level C and Level D conditions. Uncertainties of SSE and operational loadings are usually enveloped by the conservatism in the generation of the operating loads, the conservatisms in the analysis, and the conservatisms inherent in the ASME Code. Seismic loading is governed by the SSE events as a Level C condition, or equivalently as a Level B condition with adjusted cycles and magnitude.

SSAR Revision:

Revise the final paragraph of Subsection 3.9.2.5 as follows:

Although the AP600 design loads for LOCA conditions are based on the use of mechanistic pipe break criteria (see Subsection 3.6.3), enveloping LOCA loads for 1 square foot hot and cold leg breaks are used in the analysis of the reactor internals.

~~The conditions evaluated for pipe rupture are based on the application of mechanistic pipe break criteria to the reactor coolant system piping. High energy piping of four-inch nominal size or larger containing reactor coolant is evaluated for leak-before-break considerations. The pipe ruptures considered for evaluation of dynamic effects are those not excluded by application of mechanistic pipe break criteria.~~

~~Based on the successful application of mechanistic break, the reactor internals do not have to be evaluated for the dynamic effects of a rupture of the reactor coolant piping or large branch line piping connected to the reactor coolant loop piping. See Subsection 3.5.3 for a description of the leak-before-break criteria.~~

Question 210.95

Traditionally, the design of PWR internals is dominated by the LOCA loads due to postulating large breaks in the coolant loop. Thus, the internals have ample margins to resist an SSE, operation transients, and flow-induced vibrations, that generally induce less significant stress levels and deflections than that induced by the LOCA. Due to the application of leak-before-break (LBB), LOCA loads become less important in the internals design. In Section 3.9.2.5 of the SSAR, identify the largest LOCA used in the design of AP600 internals, and provide a discussion regarding how margins were maintained to ensure adequate defense of reactor internals against uncertainties of SSE and operational loads.

Response:

Consistent with past practice, the AP600 reactor internals are designed for the dynamic effects of 1 square foot hot leg and cold leg breaks. The AP600 core support structures and threaded structural fasteners analysis and ASME Section III Subsection NG allowable stresses ensure acceptability of the design loadings for Level A, Level B, Level C and Level D conditions. Uncertainties of SSE and operational loadings are usually enveloped by the conservatism in the generation of the operating loads, the conservatisms in the analysis, and the conservatisms inherent in the ASME Code. Seismic loading is governed by the SSE events as a Level C condition, or equivalently as a Level B condition with adjusted cycles and magnitude.

SSAR Revision:

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Although the AP600 design loads for LOCA conditions are based on the use of mechanistic pipe break criteria (see Subsection 3.6.3), enveloping LOCA loads for 1 square foot hot and cold leg breaks are used in the analysis of the reactor internals.

~~The conditions evaluated for pipe rupture are based on the application of mechanistic pipe break criteria to the reactor coolant system piping. High-energy piping of four-inch nominal size or larger containing reactor coolant is evaluated for leak-before-break considerations. The pipe ruptures considered for evaluation of dynamic effects are those not excluded by application of mechanistic pipe break criteria.~~

~~Based on the successful application of mechanistic break, the reactor internals do not have to be evaluated for the dynamic effects of a rupture of the reactor coolant piping or large branch line piping connected to the reactor coolant loop piping. See Subsection 3.6.3 for a description of the leak-before-break criteria.~~

Question 210.97

In Section 3.9.5.3 of the SSAR, provide a more detailed discussion of the basis for the deflection allowables listed in Table 3.9-14.

Response:

In Table 3.9-14 the upper barrel radial inward deflection limit is based on preventing contact between the barrel and the peripheral upper guide tubes during a LOCA event. The rod cluster control assembly can be dropped during the LOCA event is the guide tubes are not contacted by the barrel. The radial outward (uniform) deflection is based on maintaining flow in the downcomer annulus between the core barrel and pressure vessel wall. Since the calculated deflection of the barrel is usually radially non-uniform, i.e., somewhat elliptical, the radial outward (non-uniform) deflection during a LOCA event could result in a locally peak deflection exceeding the uniform allowable deflection. A peak deflection greater than the uniform allowable is acceptable provided that the annulus blockage from the deflected core barrel is less than the non-uniform radial outward deflection limit. The upper package allowable deflection is based on the clearance between the upper core plate and guide tube support pin shoulder. Exceeding this value could result in potential buckling of the guide tube and potential loss of function during operating or accident conditions. The rod cluster guide tube allowable lateral deflection is based on test data that indicates the rod cluster control assembly drop time will not be impaired. These limits are consistent with past practice.

SSAR Revision:

Revise the second paragraph of Subsection 3.9.5.3.1 as follows:

The functional limitations for the core support structures and internal structures during operating and accident conditions are shown in Table 3.9-14. ~~To provide for no column loading of rod cluster control guide tubes, the upper package deflection is limited so as not to exceed the values shown in Table 3.9-14.~~

Revise the fourth paragraph of Subsection 3.9.5.3.2 as follows:

~~For the load combinations of pipe rupture and the safe shutdown earthquake condition, the deflection criteria of critical internals structures are the limiting values given in Table 3.9-14.~~ For normal operating and accident conditions the deflection criteria of critical core support structures and internal structures are the limiting values given in Table 3.9-14. The upper barrel radial inward deflection limit is based on preventing contact between the barrel and the peripheral upper guide tubes during a LOCA event. The rod cluster control assembly can be dropped during the LOCA event is the guide tubes are not contacted by the barrel. The radial outward (uniform) deflection is based on maintaining flow in the downcomer annulus between the core barrel and pressure vessel wall. A peak deflection greater than the uniform allowable is acceptable provided that the annulus blockage from the deflected core barrel is less than the non-uniform radial outward deflection limit. The upper package allowable deflection is based on the clearance between the upper core plate and guide tube support pin shoulder. Exceeding this value could result in potential buckling of the guide tube and potential loss of function during operating or

accident conditions. The rod cluster guide tube allowable lateral deflection is based on test data that indicates the rod cluster control assembly drop time will not be impaired.

Question 210.100

The response to Q210.16 dated January 1, 1993 indicates that preoperational test data from several operating plants and from scale-model flow tests are used for the assessment of flow-induced vibrations of the AP600 reactor internals. The response also indicates that the assessment has not yet been finalized and the effort was planned to be completed in the first quarter of 1994. When this assessment is complete, revise Section 3.9.2 of the SSAR to provide a more detailed summary of the assessment results used (a) for verifying your conclusions on the adequacy of the AP600 reactor internals design to withstand flow induced vibration, and (b) to provide the basis for classifying the first AP600 plant as Non-Prototype, Category II in accordance with Position C.1.5 of RG 1.20.

Response:

a. Response Revision 1 to RAI 210.16 outlines the vibration assesment performed on the reactor internals for the AP600. The SSAR revisions to include the additional information for the assement results are provided below.

b. The basis for classifying the first AP600 plant as Non-Prototype, Category II is provided below in paragraphs to be added as a SSAR revision.

SSAR Revision:

Revise the tenth paragraph of Section 3.9.2.3 as follows:

These tests confirmed that the internals behaved as expected and that the vibration levels were within allowable values. The vibration testing for 17x17 fuel internals and inverted hat upper internals is reported in WCAP-8766 (Reference 4) and WCAP-8516-P (Reference 5). The vibration testing of three-loop XL type lower core support structure in DOEL 4 is reported in WCAP 10846 (Reference 6). The vibration evaluations of upper and lower internals assemblies for a four-loop XL plant, including reference to the test results in Paluel 1 (four-loop XL type without neutron pads), are reported in WCAP-10865 (Reference 7).

The results of the Doel 3, Doel 4 and Paluel 1 reactor internals vibration test programs are utilized to perform the vibration assessment of the AP600 reactor internals. The measured responses from Doel 3 and Doel 4 are adjusted to the lower AP600 flow rate to determine the expected upper internals and lower internals vibration levels respectively. The AP600 flow velocities are 85 per cent or less of the Doel 4 values at various locations in the reactor internals which results in lower responses in the AP600.

Revise the thirteenth paragraph of Section 3.9.2.3 as follows:

The vibrations of the upper internals components are well characterized by previous plant testing based on the following: The control assembly guide tubes and support column designs are similar to those in a previously tested plant. The outlet nozzle velocity is less than the outlet nozzle velocity of previously tested three-loop plants.

The AP600 upper internals design is substantially the same as that measured in the Doel 3 plant and 3XL scale model tests. The AP600 support column, guide tube and upper support assembly are nearly identical to these measured components. There are slight differences in the numbers of guide tubes and support columns but little effect on the vibrational responses is expected since these components respond as individual beams. The

corresponding AP600 responses were calculated to be less than the previous plant responses due to the lower flow velocities in the AP600.

Revise the fifteenth paragraph of Section 3.9.2.3 as follows:

The core barrel outside diameter and inside diameter and the reactor vessel inside diameter are the same as the tested three-loop plants. The core barrel length is one foot shorter. The coolant velocity in the downcomer annulus between the core barrel and the reactor vessel wall is lower in the AP600 design than in previous three-loop plants because the AP600 has no thermal shield or neutron pads in the annulus and the total reactor coolant flow rate is lower.

The vibrational response of the core barrel was measured during the Doel 4 reactor internals vibration measurement program. The diameter, length and thickness are nearly identical to the AP600 core barrel and the both utilize the single combined lower core support plate. The cantilever beam mode frequency and amplitude of the AP600 core barrel are calculated to be similar to the measured Doel 4 responses. Comparison of the 4XL scale model to the Paluel plant test results indicate that the removal of the neutron panels has little effect on core barrel vibration.

Revise the sixteenth paragraph of Section 3.9.2.3 as follows:

The reflector is shorter than the core barrel, has a larger cross sectional area and a smaller diameter than the core barrel, and is more rigidly clamped at its axially supported end, so that its vibration is expected to occur primarily coupled to the core barrel and to have small vibration levels relative to the core barrel.

The replacement of the baffle-former structure with the radial reflector reduces the stiffness of the core barrel. The AP600 shell mode amplitudes are estimated to be higher than the standard 3 loop core barrel responses based on scaling the measured responses to the AP600 reduced core barrel stiffness.

The AP600 core barrel and reflector will be instrumented during the preoperational testing of the first plant to determine the shell mode frequencies and amplitudes.

Revise the third paragraph of Section 3.9.2.4 as follows:

With respect to reactor internals, the first AP600 plant is classified as a Non-Prototype Category II according to Regulatory Position C.1.5 in Regulatory Guide 1.20. The comparison in Subsection 3.9.2.3 of the AP600 reactor internals with previous Westinghouse designs supports this classification.

The prototype (reference) plant for the AP600 is H. B. Robinson which has substantially the same size and operating conditions as the AP600. Structural differences include modifications resulting from the use of 17x17 fuel, the removal of the thermal shield and the change to the inverted top hat upper internals support assembly. These design changes were incorporated into the Doel 3 and Doel 4 reactor internals as well as the AP600.

The effects of these design evolutions from the reference plant were shown by instrumented preoperational testing at the Doel 3 (upper internals) and Doel 4 (lower internals) plants. The vibrational responses of the AP600 reactor internals are characterized by the Doel 3 and 4 vibration measurement programs.

Question 210.102

The response to Q210.18 dated January 14, 1993 indicates that the preoperational vibration test program for the initial AP600 plant remains to be developed. Thus, detailed information regarding the program, including types and locations of sensors to be installed, the bases used to establish expected and acceptable vibration levels, and the conditions at which data are to be acquired, is not available at this time. The staff's position is that such information is essential for ensuring design adequacy of reactor internals to withstand flow-induced vibrations under operational transients. Subsequent to the staff receiving an acceptable response to Q210.100, develop and provide such information in the SSAR for design certification review.

Response:

As noted in Revision 1 of the response to RAI 210.16 the reactor internals flow-induced vibration assessment program has been completed. As noted in Revision 1 of the response to RAI 210.18 a test plan for reactor internals vibration has been developed. This test will be similar to previous plant tests and will provide added confidence of the adequacy of the reactor internals design, in addition to the vibration assessment program. The following is a summary of the plan.

The reactor internals flow-induced vibration measurement program will be conducted during preoperational tests of the first AP600 plant. The major structural components of the first AP600 plant reactor lower internals will be instrumented during pre-operational testing. Transducers will be installed on the reactor vessel and the internals prior to the cold hydrostatic test. The integrity of these transducers and the operability of the data acquisition equipment will be verified during this test.

The response of the reactor and the internals due to flow-induced vibration will be measured during the hot functional testing. As shown by the results of the vibration assessment program, the dominant vibration modes of the internals with no core present are similar to those with the core in place and their vibration amplitudes are expected to be more than 10% higher.

Data will be acquired at several temperatures from cold startup to hot standby (529°F) conditions. Data will be recorded for pump startup and shutdown transients as well as for all possible combinations of steady-state pump operation. In addition, data will be recorded with none of the pumps operating in order to determine the background noise level.

Transducer signals will be monitored as they are being recorded to insure the validity of the data. A spectrum analyzer will be used during the test as an additional check on transducer performance. The spectrum analyzer will also provide preliminary information on the natural frequencies and responses of the instrumented components. The majority of the data, however, will be analyzed from the magnetic tapes.

The leads for these internally mounted transducers will be routed through the top mounted instrumentation guide tube conduits. The combined in-core detectors/core exit thermocouples will not be installed during the hot functional test. Special fittings, designed to ASME Section III, Class 1 pressure boundary rules, will be used to seal the transducer leads during this test. These fittings will be removed following the test.

All transducers and associated hardware will be removed after the completion of the Hot Functional testing.

Location of Transducers

Transducer locations and their directions of sensitivity are listed in the table included in the SSAR Revision. The measurement objectives for the instrumented components are listed below:

1. Four radially sensitive accelerometers mounted near the top of the radial reflector. These transducers are to detect shell mode vibration of the radial reflector and provide additional information on the core barrel beam modes.

2. Six axially sensitive strain gages mounted just below the core barrel flange. These transducers will detect axial vibration of the lower internals and core barrel beam modes.

3. Two axially sensitive strain gages (one inside and one outside) mounted on the upper support assembly skirt to detect vertical motion of the upper support structure. Alternatively, this information may be obtained using axially sensitive accelerometers.

4. Four axially (2 inside and 2 outside) sensitive strain gages located on the core barrel to lower core plate weld. These strain gages will provide direct information on the stresses at this location. Alternatively, this information may be obtained using axially sensitive accelerometers.

5. Four axially sensitive strain gages mounted on two lower support columns that attach the vortex suppression plate to the lower support plate. These gages will be mounted at 90 degree separation on two different support columns such that lateral displacement of the vortex suppression plate assembly can be determined. Alternatively, four horizontally sensitive accelerometers will be considered to obtain this information.

6. Two axially sensitive strain gages located on the upper support column extension. These transducers will detect the lateral displacement of the extension.

7. Four vertically sensitive and two horizontally sensitive accelerometers mounted on the reactor head closure studs at 90° intervals. These transducers will detect motion of the reactor vessel and the upper and lower internals flanges.

8. Four radially sensitive accelerometers will be installed at the upper core plate elevation to determine the shell mode responses of the core barrel.

The bases used to establish expected and acceptable vibration levels and expected natural frequencies are found in the vibration assessment program. The final values established for expected and acceptable levels will be established prior to the start of testing.

SSAR Revision:

Revise the last paragraph of Subsection 3.9.2.4 as follows:

The reactor internals flow-induced vibration measurement program will be conducted during preoperational tests of the first AP600. Transducers will be installed on the reactor vessel and the internals prior to the cold hydrostatic test. The response of the reactor and the internals due to flow-induced vibration will be measured during the hot functional test. Data will be acquired at several temperatures from cold startup to hot standby conditions. The location of the transducers is outlined in Table 3.9-X(1). The leads for the internally mounted transducers will be routed through the top mounted instrumentation guide tube conduits through special fitting that will be removed following the test.

The bases used to establish expected and acceptable vibration levels and expected natural frequencies are found in the vibration assessment program. The acceptance standards for the inspection of reactor internals before and after the hot functional testing are the same as required in the shop by the original design drawings and specifications.

(1) The table number in the SSAR will depend on the number of tables added for other RAI responses.

Add a table to Section 3.9 for the transducer locations for the first plant AP600 reactor internals vibration measurement program as follows:

AP600

Table 3.9-X First Plant AP-600 Reactor Internals Vibration Measurement Program Transducer Locations

Instrumented Component	Number and Type of Transducers	Transducer Locations	Direction of Sensitivity
Radial Reflector (Inner Wall)	4 accelerometers	0°, 180°, 225°, 270°	Radial
Core Barrel Flange (Outer Wall)	4 strain gages	0°, 90, 180°, 270°	Axial
Core Barrel Flange (Inner Wall)	2 strain gages	180°, 270°	Axial
Core Barrel Mid-elevation	4 accelerometers	0°, 180°, 225°, 270°	Radial
Upper Support Skirt (Inside and Outside)	2 strain gages	180°	Axial
Lower Support Plate Weld (Inside and Outside)	4 strain gages	0°, 90°	Vertical
Vortex Suppression Plate Support Columns (2)	4 strain gages or	On column near lower core support plate	Axial
	4 accelerometers	or on vortex suppression ring	Horizontal
Reactor Vessel (Head Studs)	4 accelerometers	0°, 90°, 180°, 270°	Vertical
	2 accelerometers	0°, 90°	Horizontal
Support Column Extension	2 strain gages	0°, 90°	Axial

Question 210.103

The response to Q210.19 dated January 8, 1993 identifies ASME Code criteria applicable to AP6CJ core support structures. Provide specific values of stress limits, deflection limits, and buckling stability limits for various core support structures. Also, provide the design limits of internal structures other than the designated core support structures.

Response:

The response to this question is addressed in responses to RAI 210.70, RAI 210.88, RAI 210.96, and RAI 210.97. The stress limits and buckling stability limits are taken from ASME Code, Section III, Subsection NG and Appendix F for Level A, Level B, Level C, and Level D conditions for core support structures and threaded structural fasteners. The deflection limits are shown in Table 3.9-14 and discussed in response to RAI 210.97. The stability limit for the core drop scenario is discussed in Section 3.9.5.3.2. The design limits of internal structures are discussed in the response to RAI 210.70.

SSAR Revision: NONE

Question 210.104

In Revision 1 to WCAP-13054, under Section 3.9.2 of the SRP, an exception is taken to Position C.1 in RG 1.20, and Position C.2 in RG 1.20. These positions are listed as not applicable to the AP600 design certification because it applies to preoperational and initial startup testing. The staff cannot evaluate these issues until it receives acceptable responses to Q210.100 and Q210.102. Revise WCAP-13054 to agree with the resolution of those RAIs.

Response:

The AP600 position on Position C.1 in Regulatory Guide 1.20 remains an exception as stated in WCAP-13054. The response to RAI 210.100 provides additional information on the definition of Non-Prototype Category II as the applicable Regulatory Guide 1.20 category for reactor internals vibration testing.

Position C.2 in Regulatory Guide 1.20 recommends implementation of a vibration assessment program. The AP600 position on the implementation remains as not applicable as stated in WCAP-13054. As outlined in Revision 1 to the response for RAI 210.16, a vibration assessment analysis has been completed for the AP600 reactor internals. As outlined in response revision 1 to RAI 210.18 a test plan for a vibration measurement program has also been developed for the first AP600 reactor internals. The response to RAI 210.102 contains an outline of the vibration measurement program. WCAP-13054, Revision 1, Section 3.9.2 remains valid for Positions C.1 and C.2 in Regulatory Guide 1.20.

SSAR Revision: NONE

Question 210.105

In Revision 1 to WCAP-13054, Section 4 in Section 3.9.2 of the SRP is listed as acceptable. However, in the "Comments/Summary of Exception" column, it states that the reactor internals for the first AP600 are classified as Non-Prototype Category II as defined in position C.1.5 of RG 1.20. The staff has not yet accepted this classification for the AP600. This decision will be made by the staff as a part of its review of the responses to Q210.100 and Q210.102. Revise WCAP-13054 to agree with the resolution of those RAIs.

Response:

The response to RAI 210.100 provides additional information on the definition of Non-Prototype Category II as the applicable Regulatory Guide 1.20 category for reactor internals vibration testing. Based on that response, WCAP-13054 does not have to be revised for Criteria 4 of the Standard Review Plan Section 3.9.2. The response to RAI 210.102 addresses the vibration measurements plan and does not address Criteria 4 of the Standard Review Plan Section 3.9.2.

SSAR Revision: NONE

Question 210.107

In Section 3.9.3.4 of the SSAR, provide a commitment that for pipe support base plate designs, the applicable action items in IE Bulletin 79-02, Revision 2, dated November 8, 1979 will be met. The staff's position on this issue is as follows:

- If "undercut" type expansion anchor bolts will be used in the AP600, and, if the safety factors used for such bolts are different from those in IE Bulletin 79-02, provide the factors which will be used in the design of the "under-cut" type of expansion anchor bolt and the basis for these factors.
- Irrespective of the type of expansion anchor bolt that will be used, the staff requires a commitment to the action item in IE Bulletin 79-02 relative to pipe support base plate flexibility.

Response:

The factor of safety for undercut type anchor bolts is determined in accordance with Appendix B of ACI 349. The AP600 response to the NRC position on the use of Appendix B of ACI 349 is described in the response to RAI 220.84.

SSAR Revision:

Add new paragraph at the end of Section 3.9.3.4 as follows:

Use of baseplates with concrete expansion anchors is minimized in the AP600. Concrete expansion anchors may be used for pipe supports. For these pipe support baseplate designs, the baseplate flexibility requirements of IE Bulletin 79-02, Revision 2, dated November 8, 1979 are met by accounting for the baseplate flexibility in the calculation of anchor bolt loads.

Question 210.109

In Section 5.2.1.2 of the SSAR, provide a list of the ASME Code Cases to be used in the AP600 plant design.

Response:

A list of ASME Code Cases planned to be used in the AP600 will be added to the SSAR. The final list of Code Cases used can not be established until component vendors are selected and components are procurred.

Use of ASME Code Cases in addition to those listed will be controlled as identified in Subsection 5.2.1.2. A number of the code cases included in the list are not included in the list of approved ASME Code Cases in Regulatory Guide 1.84 Revision 29 and Regulatory Guide 1.85, Revision 29. A separate letter (NTD-NRC-94-4182) has been transmitted to the NRC requesting approval, as required by 10 CFR 50.55a(a)(3), of the use of the code cases identified in this response that are not listed in the Regulatory Guides.

SSAR Revision:

Revise the first paragraph in Subsection 5.2.1.2 as follows:

ASME Code Cases used in the AP600 are listed in Table 5.2-3. In addition, pPressure boundary components may use other ASME Code Cases found in Regulatory Guides 1.84 and 1.85, as discussed in Section 1.9, in effect at the time of the Design Certification. Use of Code Cases approved in revisions of the Regulatory Guides issued subsequent to the Design Certification follows the requirements of the Design Certification. Use of any Code Case not approved in Regulatory Guides 1.84 and 1.85 on Class 1 components is authorized as provided in 50.55a(a)(3) and the requirements of the Design Certification.

Add Table 5.2-3 as follows:

Table 5.2-3

ASME Codes Cases

Code Case Number	Title
N-4-11	Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and Class CS.
N-20-3	SB-163 Nickel-Chromium-Iron Tubing (Alloys 600 and 690) and Nickel-Iron-Chromium Alloy 800 at a Specified Minimum Yield Strength of 40.0 ksi and Cold Worked Alloy 800 at Yield Strength of 47.0 ksi, Section III, Division 1, Class 1.
N-60-5	Material for Core Support Structures, Section III, Division 1.

N-71-15	Additional Material for Subsection NF, Class 1, 2, 3 and MC Compo..ent Supports Fabricated by Welding, Section III Division 1
N-122-1	Stress Indices for integral Structural Attachments Section III, Division 1, Class 1
N-201-3	Class CS Components in Elevated Temperature Service, Section III, Division 1.
N-249-11	Additional Materials for Subsection NF, Class 1, 2, 3, and MC Supports Fabricated Without Welding, Section III, Division 1
N-282	Metal containment Shell Buckling Design Methods, Section III, Division 1 Class MC
N-318-4	Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping Section III, Division
N-319-2	Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping Section III, Division 1
N-391-1	Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping Section III, Division 1
N-392-2	Procedure for valuation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping Section III, Division 1
N-474-2	Design Stress Intensities and Yield Strength Values for UNS06690 With a Minimum Yield Strength of 35 ksi, Class 1 Components, Section III, Division 1.
2142	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX.
2143	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX.

Question 210.110

The response to Q210.25 dated January 22, 1993 and Section 3.9.6 of the SSAR both state that the AP600 inservice testing program will include safety-related ASME Class 1, 2, or 3 valves. The staff's position for passive plants, as recommended in Section H, "Inservice Testing of Pumps and Valves," of SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 is that those important non-safety-related pumps and valves identified by the regulatory treatment of non-safety systems (RTNSS) process should be designed to accommodate testing in accordance with ASME Code, Section XI. Specific positions on the inservice testing requirements for these components will be finalized when the staff completes its review of the RTNSS issue. Revise the response to Q210.25 and Section 3.9.6 of the SSAR to reflect this staff position.

Response:

Revision 1 to the response to RAI 210.24 provides the AP600 inservice testing plan. There are no nonsafety-related pumps or valves in the AP600 inservice testing plan. The AP600 evaluation of the regulatory treatment of nonsafety-related systems (RTNSS) described in Reference 210.110-1 identifies that the only important nonsafety-related fluid systems (as defined by RTNSS) are the normal residual heat removal system and supporting systems (component cooling water system and service water system) during reduced reactor coolant system inventory conditions (midloop). These systems are normally used during reduced reactor coolant system inventory conditions. As recommended in Reference 210.110-1, redundant pumps and subsystems in these three systems are to be demonstrated to be operable prior to entering reduced inventory conditions. The nonsafety-related pumps and valves in these systems are not required to be included in the inservice testing plan. The evaluation described in Reference 210.110-1 did not result in a determination that would require revision of the response to RAI 210.25 or of Subsection 3.9.6.

Reference:

210.110-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process Summary Report, September, 1993.

SSAR Revision: NONE

Question 220.49

The seismic Category II structures, such as the turbine building, the annex buildings I and II, and the solid radwaste building are sufficiently close to the nuclear island such that their collapse could affect the safety function of Category I structures. The structural integrity is the requirement for seismic Category II structures. Therefore, provide the reason why the seismic Category II structures are excluded for the foundation analyses (Section 3.8.5 of the SSAR).

Response: (Revision 1)

As described in the response to RAI 230.54, the only seismic Category II structure adjacent to the nuclear island is the high bay area of the radwaste building. The foundation for this building is a reinforced concrete mat on grade as described in SSAR Subsection 1.2.7. Design criteria for the structure including the foundation mat are described in SSAR Subsection 3.7.2.8, as revised in the response to RAI 230.54. The annex and turbine building have been reclassified to non-seismic. SSAR Subsection 3.7.2.8, as revised by the response to RAI 230.54, provides the basis to demonstrate that the design of these non-seismic structures that are located adjacent to the nuclear island is such that the SSE will not cause unacceptable interaction with, or failure of, any seismic Category I items.

SSAR Revision: NONE

Question 220.58

In the nuclear island foundation design, consider the seismic shear and moments due to the out-of-phase vibration between the shield building, containment shell, and structures.

Response:

The shield building, containment vessel and containment internal structures are concentric structures supported on the nuclear island basemat. For portions of the basemat outside the shield building, out-of-phase vibration of these structures reduces the total load on the basemat, and seismic shear forces and moments are lower than those predicted by the analyses assuming the three structures to be in phase.

The portion of the basemat inside the shield building is a circular slab varying in thickness from six to twenty two feet. The safe shutdown earthquake loads on this circular slab are the reactions from the containment vessel (including the containment internal structures), the soil reactions, and the equivalent static accelerations applied to the slab itself. Loads due to the equivalent accelerations are small. Similarly, as shown in Tables 3.7.2-11, 3.7.2-12, and 3.7.2-13, the shear forces and moments for the steel containment vessel and containment internal structures at elevation 100 feet are much less than the shear forces and moments for the coupled auxiliary and shield buildings at that elevation. Therefore, it is judged that the out-of-phase vibration between the shield building, steel containment vessel, and containment internal structures would have no effect on the nuclear island basemat.

Confirmatory analyses will be performed to confirm this judgement. The following steps will be followed to determine whether out-of-phase vibration between the shield building, containment vessel, and the containment internal structures govern the design of the portion of the basemat inside the shield building:

1. Using the current N.I. basemat analysis results for the basemat, establish two load cases where the bearing pressure between the basemat and containment internal structures are maximum in the first iteration (i.e. linear elastic analysis).

2. For the two cases established above, reverse the direction of the horizontal seismic forces in the containment vessel and containment internal structures. Perform linear elastic analysis to obtain forces and moments in the basemat.

3. Compare the new forces and moments established in step 2 with the current forces and moments obtained in the first iteration. If the current forces and moments are greater than the forces and moments established in step 3, then it can be concluded that out-of-phase vibration does not govern the design of the basemat.

4. If the current forces and moments are less than the forces and moments established in step 3, then establish additional cases and design the basemat accordingly.

SSAR Revision: NONE

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:

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).

The structural design is defined as a series of structural design calculations, drawings and analysis reports. As required in Appendix B to Standard Review Plan 3.8.4, conduct of the structural audit will include a presentation by Westinghouse giving an overview of each of the key structures and identification of the related AP600 design documents.

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 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).

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 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 PCS tank, precast panels and roof to cylinder connection)
- Summary of Results
- Conclusions
- List of Engineering Drawings and Design Calculations
- References

Question 220.89

Identify the number of critical sections chosen for the detailed structural design and the basis for this selection. In addition, provide drawings indicating the reinforcement details of all selected critical sections in the SSAR.

Response:

Critical sections representative of the structural design are identified in Table 220.89-1. Information on the design of these critical sections will be included in the design summary reports described in the response to RAI 220.83. This will include summaries of the forces, moments, required reinforcing steel and the areas provided. The summary reports will identify the reinforcement details. Typical reinforcing drawings are already included in the SSAR (see SSAR Figures 3.8.3-4, 3.8.4-2, 3.8.4-3, and 3.8.5-3).

SSAR Revision: NONE

Table 220.89-1

Critical Sections

Location	Basis for Selection
Passive containment cooling system water storage tank	This is a unique area of the roof and water tank.
Shield building roof to cylinder location at columns	This is the junction between the shield building roof and the cylindrical wall of the shield building.
Shield building to auxiliary building connection at elevation 180'	This is the connection between the two buildings at the highest elevation.
South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'	This exterior wall illustrates typical loads such as soil pressure, surcharge, seismic, tornado, etc.
Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6"	This is one of the most highly stressed shear walls.
West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0"	This illustrates design of the main control room wall.
North wall of auxiliary building (column line 11 between Q and P) elevation 117'-6" to 153'-0"	This wall acts as an anchor for the main steam line.
Floor slab on metal decking (battery room, elevation 82'-6")	This is a typical slab on metal decking and structural steel framing.
Basemat below the containment vessel	This illustrates the reinforcement design for the basemat under the containment vessel.
Column in the containment internal structures	This is a typical column in the containment internal structures.
West wall of the refueling cavity. It will include the connections to the IRWST outer wall, the operating floor (elevation 135') and the base connection to the basemat.	This illustrates the design of a typical 4'-0" thick concrete filled steel structural module, including its connections to adjacent modules and the basemat.
West wall of steam generator cavity	This illustrates the design of a typical 2'-6" thick concrete filled steel structural module.

IRWST wall (steel wall adjacent to containment vessel)	This illustrates the design of a steel structural module.
2'-0" slab in auxiliary building	This illustrates the design of a 2'-0" thick concrete slab.
Finned floor above the main control room at elevation 135'-3"	This illustrates the design of the finned floors.
Spent fuel pool liner details	This provides typical spent fuel pool details.

Question 410.117

Section 9.2.2.1.1 of the SSAR states that the component cooling water system (CCS) serves no safety-related function except for containment isolation. Explain why AP600 CCS does not have any safety-related function, other than containment isolation, as compared to the CCS of the current PWR plants, that has safety-related functions. Explain the function of each of the components in AP600 CCS to demonstrate that none of them perform any safety function. The discussion should consider CCS function during normal plant operation, LOCA or transient, LOOP, and shutdown plant operation.

Response:

The component cooling water system (CCS) for current plants is required to be safety-related because a number of components supplied by the component cooling water system require cooling water to perform their safety-related functions to establish and maintain safe shutdown conditions following an event. The safety-related components supplied by the component cooling water system in current plants include charging pumps that provide high head safety injection and heat exchangers that provide core and containment cooling. In addition, the component cooling water system for current plants also includes safety-related containment isolation valves for the portions of the lines that penetrate the containment.

The AP600 component cooling water system provides cooling water to various nonsafety-related components and also to the following safety-related components:

- Reactor coolant pumps (ASME Class 1)
- Chemical Volume and Control System letdown heat exchangers (ASME Class 3)
- Normal RHR heat exchangers and pumps (ASME Class 3)

These safety-related components do not require component cooling water to perform safety-related functions during normal plant operation, shutdown conditions, or following events such as LOCA, transients, or loss of offsite power (LOOP). The safety-related functions of these components are limited to maintaining reactor coolant system pressure boundary integrity, whether or not the component is operating. The reactor coolant pumps also have a safety-related function to provide for reactor coolant system flow coast down for events such as a loss of offsite power where the reactor coolant pumps are de-energized. Maintenance of pressure boundary integrity for these components and reactor coolant pump coastdown capability, which are safety-related functions, are independent of component cooling water system operation. See SSAR Section 5.4.1.3.4 for a discussion of the coastdown capability of the reactor coolant pump. The component cooling water system is also required for normal residual heat removal operation to provide plant cooldown and core decay heat removal functions during shutdown conditions. Safety-related cooldown and core decay heat removal functions are provided by the passive core cooling system and containment cooling systems (see Sections 6.3 and 6.2). Therefore, the component cooling water system is not required to operate to perform any safety-related functions.

Although the component cooling water system is a nonsafety-related system, segments of the system piping that penetrate the containment and the associated isolation valves are safety-related and perform a safety-related

containment isolation function. These portions of the component cooling water system are designed using components with safety-related AP600 equipment classifications.

The component cooling water system function to support normal residual heat removal operation to provide core decay heat removal during reduced reactor coolant system inventory shutdown conditions is identified as a significant nonsafety-related function in Reference 410.117-1. Reference 410.117-1 also provides the appropriate additional regulatory oversight for those portions of the component cooling water system that perform this function. The oversight proposed includes short-term availability recommendations that verify operability of these functions prior to entering the reduced reactor coolant system inventory condition.

Please see the response to RAI 210.36 for additional information.

Reference:

410.117-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

Question 410.118

Section 9.2.2 of the SSAR states that the component cooling water system is a non-safety-related, closed loop cooling system that transfers heat from various plant components to the service water system during normal phases of operation and removes core decay heat and sensible heat for normal reactor shutdown and cooldown from various plant components. The components that were referred to in the above statement are listed only in the proprietary version of the SSAR.

Without a description of those components, it is not possible to discuss and determine that the CCS is a non-safety-related system. The staff has determined that the description of those components should not be treated as proprietary information. Revise the SSAR to bring this information into the non-proprietary version of the SSAR.

Response:

The components to which the component cooling water system (CCS) provides cooling are identified on Figure 9.2.2-1, Component Cooling Water System Simplified Flow Diagram. This figure is in the non-proprietary portion of the SSAR. See the response to RAI 410.117 for a discussion of why the component cooling water system is nonsafety-related. Since the component cooling water system components used for transferring heat from various plant components to the service water system (SWS) are nonsafety-related, no further description of these components is needed in the SSAR. Ongoing discussions are planned between Westinghouse and the NRC regarding agreement on the scope of proprietary information.

SSAR Revision: NONE

Question 410.121

For the SWS, Section 9.2.1.1 of the SSAR states that failure of the service water system or its components will not affect the ability of safety-related systems to perform their intended safety function. There is no similar statement in the SSAR for the CCS.

Clarify if the same statement is applicable for the CCS. Does the CCS meet the guidance of Position C.2 of RG 1.29 for the non-safety-related portions? Provide the bases for this position.

Response:

The statement that failure of the component cooling water system or its components will not affect the ability of safety-related systems to perform their intended safety function is applicable to the component cooling water system. The component cooling water system conforms to the guidance of Position C.2 of Regulatory Guide 1.29, as described in the response to RAI 210.36. Please refer to the response to RAI 410.117 for a discussion of the non-safety related nature of the CCS.

SSAR Revision: Revise Subsection 9.2.2.1.1 as follows:

9.2.2.1.1 Safety Design Basis

Failure of the component cooling water system or its components will not affect the ability of safety-related systems to perform their intended safety functions. The component cooling water system serves no safety-related function except for containment isolation and therefore has no nuclear safety design basis except for containment isolation (see Subsection 6.2.3).

Question 410.122

It is indicated in WCAP-13054 regarding compliance with Section 9.2.2 of the SRP that the AP600 design will meet the requirements of GDC 4. GDC 4 requires that SSCs important to safety shall be designed to accommodate environmental and dynamic effects. Demonstrate in the SSAR how GDC 4 is met by the CCS.

Response:

The Component Cooling System (CCS) is not a safety-related system. The only safety function performed by the component cooling water system is containment isolation. The component cooling water system components which provide the containment isolation function are designed to accomodate environmental and dynamic effects and thus, satisfy GDC4. (See SSAR Chapter 3 and Section 6.2.3)

SSAR Revision: NONE

Question 410.123

It is indicated in WCAP-13054 regarding compliance with Section 9.2.2 of the SRP that the AP600 design will meet the requirements of GDC 44, 45, and 46 for the CCS. Demonstrate in the SSAR how the above GDC are met by the CCS.

Response:

The response to RAI 410.117 demonstrates that the component cooling water system (CCS) is nonsafety-related except for containment isolation valves. GDC 44, 45 & 46 relate to the capability to transfer heat loads from safety-related structures, systems and components to a heat sink. Accordingly, GDC 44, 45 & 46 do not apply to the component cooling water system.

The next revision of WCAP 13054 will change the indication of compliance to GDC 44,45, & 46 from "Acceptable" to "Not Applicable".

SSAR Revision: NONE

Question 410.124

It is indicated in WCAP-13054 regarding compliance with Section 9.2.2 of the SRP that the AP600 design will meet IEEE 279 for CCS instrumentation. Demonstrate in the SSAR how this commitment for the AP600 CCS instrumentation is met. Specifically, provide additional information to address the staff review guidance stated in Paragraph III.4.d of Section 9.2.2 of the SRP with respect to meeting IEEE 279.

Response:

SSAR Subsection 5.4.1.2.1 provides a description of the instrumentation used to monitor RCP performance, alert operators of abnormal conditions and automatically terminate operation.

As stated in SSAR, Subsection 7.1.2.2, a reactor trip signal is generated by a high reactor coolant pump bearing water temperature. This signal is not from an instrument in the component cooling water system. None of the signals used for monitoring or controlling the reactor coolant pumps originate from the component cooling water system. Since the component cooling water system function to supply cooling water to the reactor coolant pumps is nonsafety-related and none of the comp. nent cooling water system instrumentation performs a safety-related function, the component cooling water system is not required to be in compliance with IEEE 279.

The next revision of WCAP 13054 will change the indication of compliance to IEEE 279 from "Acceptable" to "Not Applicable".

SSAR Revision: NONE

Question 410.125

Section 9.2.2.6 of the SSAR, "Inspection and Testing Requirements," describes operational testing of CCS components without specifics, such as test frequency and acceptance criteria. However, Section C11.2.3 and Table C11-5 of the PRA states that it is assumed that most of components (pumps, heat exchangers, and valves involved in train operation change) are tested quarterly, and other maintenance valves are assumed to be tested once every refueling cycle.

Confirm that the tests assumed in the PRA are a design requirement for COL applicants to carry out. If that is the case, revise Section 9.2.2.6 of the SSAR to delineate the testing and inspection program. In addition, because there are no technical specifications for the CCS, where will the test requirements in terms of the test frequency and acceptance criteria be specified?

Response:

Since the CCS is in operation during plant operation, there are no in-service testing design requirements imposed to support PRA reliability assumptions. The AP600 In-Service Testing Program is discussed in the response to RAI 210.24.

The CCS provides a significant nonsafety-related function during reduced RCS inventory conditions. As described in Reference 410.125-1, the following surveillance will be performed to verify the short-term availability of those portions of the CCS that support the RTNSS-important function:

- Demonstration of system availability prior to entering reduced reactor coolant system inventory shutdown operations.

Reference:

410.125-1 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

Question 410.129

Section 9.2.8.2.2 of the SSAR states that codes and standards applicable to the turbine building closed cooling water system are listed in Section 3.2. Table 3.2-3 of Section 3.2, "Classification of Components and Systems," is supposed to list this information. However, the staff finds that the turbine building closed cooling water system is not included in the table. Revise Table 3.2-3 to include the above information for the turbine building closed cooling water system.

Response:

The turbine building closed cooling system (TCS) is classified as AP600 Class E. TCS is listed in Table 3.2-3 sheet 107 of 107 as one of the systems that contain no components that are Class A, B, C, or D. Class E components are designed to industrial standards as discussed in note 12 of Table 3.2-1.

SSAR Revision:

Revise Subsection 9.2.8.2.2 as follows:

9.2.8.2.2 Component Description

~~Codes and standards applicable to the turbine building closed cooling water system are listed in~~Turbine building closed cooling water system component classification is given in Section 3.2. Equipment parameters are given in Table 9.2.8-2.

Question 410.131

Describe how the AP600 turbine building closed cooling water system is designed to minimize the potential for water hammer.

Response:

The turbine building closed cooling water system is a closed loop system. A surge tank keeps the system full of water and provides a pressure head on the system. Drain down or voiding of system piping or components does not occur during transients or upon system shutdown; therefore, the potential for water hammer due to water column rejoining is minimized. Temperatures in the system are moderate and the surge tank keeps a sufficient head on the system such that the pressure of the TCS fluid is kept above its saturation pressure; therefore, the potential for water hammer due to thermodynamic voiding and subsequent vapor collapse is minimized. There are no fast acting power operated valves in the system, and the only check valves in the normal operating flow path are in a standard configuration at the discharge of each TCS pump; therefore, the design of the system minimizes water hammer potential due to rapid valve actuation.

SSAR Revision: NONE

Question 410.132

Demonstrate that the turbine building closed cooling water system pumps have sufficient available net positive suction head (NPSH) at the pump suction locations.

Response:

As can be seen from SSAR Table 9.2.8-2, temperatures in the turbine building closed cooling water system (TCS) are moderate and fluid vapor pressures are well below atmospheric. The TCS surge tank maintains a minimum head available at the suction of the TCS pumps. The TCS pumps are located at the base slab of the turbine building (elevation 100') and the TCS surge tank is located above the operating deck (elevation 161'). The gravity head from the surge tank maintains an ample available net positive suction head (NPSH) at the TCS pump suction.

SSAR Revision: NONE

Question 410.134

Provide information with respect to the analysis of postulated cracks in moderate-energy piping systems for the turbine building closed cooling water system.

Response:

Piping and fluid containing components of the turbine building closed cooling water system are located entirely within the turbine building. As stated in SSAR Subsection 1.2.8, no safety-related equipment is located in the turbine building. Thus, postulated cracks in the system piping will not impact safety-related components through mechanisms such as jet impingement. Possible flooding effects from failure of the system piping are enveloped by flooding effects of the circulating water system which, as described in Subsection 10.4.5.2.3, will not result in detrimental effects on safety-related equipment. Water from the failure of TCS components cannot reach safety-related equipment.

SSAR Revision: NONE

Question 410.141

Section 10.2.2 of the SSAR states that there are no safety-related systems or components located within the turbine building. Are there any safety-related structures that need to be considered for turbine-missile protection?

Response:

As discussed in SSAR Subsection 3.5.1.3, due to the orientation of the turbine generator relative to safety-related structures, systems, and components, the potential for damage from turbine missiles is negligable. Safety-related structures, systems, and components are located outside of the high velocity, low tragectory missile strike zone as defined by Regulatory Guide 1.115.

SSAR Revision: NONE

Question 410.148

Section 10.4.2.2.2 of the SSAR states that Section 3.2 provides the quality group and associated quality standards for the condenser air removal system. However, the staff cannot find the information on the quality group and associated quality standards of the condenser air removal system in Table 3.2-3 or Section 3.2 of the SSAR. Provide the above information.

Response:

The condenser air removal system (CMS) is classified as AP600 Class E. CMS is listed in SSAR Table 3.2-3 sheet 107 of 107 as one of the systems that contain no components that are Class A, B, C, or D. Class E components are designed to industrial standards as described in note 12 to Table 3.2-1.

SSAR Revision: NONE

Question 410.156

Section 9.3.1.1.2 of the SSAR indicates that the CAS provides ". . . **essentially** oil-free air for pneumatic instruments" while Section 3.1.2.2 discusses "The three **oil-free** rotary compressors . . ." If the air compressors are not truly oil-free, then what is the maximum total oil or hydrocarbon content, excluding the non-condensables?

Response:

The rotary screw air compressors are oil free. Even the cleanest supply air in an industrial environment may contain some oil and/or hydrocarbon impurities from sources other than the compressor. To provide for desiccant and instrumentation protection, a coalescing oil filter is placed upstream of the air dryers to remove the atmospheric hydrocarbon impurities. The maximum hydrocarbon content downstream of the filter is 1.0 ppm. Actual contaminant levels should be much lower in this system.

SSAR Revision: NONE

Question 410.172

Explain how the deleterious effects that dust and dirt have on diesel generator operation and reliability will be minimized (including the effects on electrical equipment).

Response:

The deleterious effects of dust and dirt on the non-safety related diesel generator operation and reliability (including the effects on electrical equipment) are addressed as follows:

1) The equipment procurement specifications for electrical equipment (require dust tight enclosures).

2) The ventilation system to the service module (which includes most of the electric switchgear) utilizes inlet air filters.

3) The standby ventilation system to the diesel generator building (utilizes inlet air filters).

SSAR Revision: NONE

Question 410.173

Provide a list of all of the diesel generator trips and state whether they are in effect only during testing or during all operational modes.

Response:

The nonsafety-related diesel generator trip conditions are as follows:

TRIP CONDITIONS	TEST MODE	OPERATION* MODE
Engine Overspeed	X	X
Engine Crankcase Pressure High	X	X
Lube Oil Pressure Low-Low	X	X
Engine Jacket Water Pressure Low	X	X
Engine Jacket Water Temperature High	X	X
Generator Differential Current (87)	X	X
Generator Bs Fault Lockout (86)	X	X
Loss of Generator Field (40)	X	X
Generator Reverse Power Actuation	X	
Generator Overcurrent (51V)	X	
Generator Negative Sequence Overcurrent (46)	X	

* During all OPERATION modes, the diesel generator operates as a standby power source, and is not connected to the other offsite power sources. During TEST mode, the diesel generator is paralleled with the offsite power source.

SSAR Revision: NONE

Question 410.189

Section 10.4.9.1.2 of the SSAR indicates that the instruments and electric valves for each of the two startup feedwater pumps are powered by the standby source motor control center circuitry. Describe the failure position of the electrically operated valves at pump suction and discharge lines.

Response:

The startup feedwater pump discharge isolation valves are motor-operated. These valves are normally closed and interlocked with the startup feedwater pumps so that they open only when their respective startup feedwater pump is in service. If both the normal ac power and the onsite standby ac power sources are unavailable, these valves fail as-is.

The startup feedwater pump suction header isolation valves are pneumatically-actuated. The isolation valve for the deaerator storage tank is normally open and fails close. The isolation valve for the condensate storage tank is normally closed and fails open. The individual pump suction isolation valves are manual valves that are normally open.

SSAR Revision:

SSAR figure 10.4.7-1 will be updated to show the startup feedwater pump suction header isolation valves as pneumatically-actuated. SSAR subsection 10.4.9.2.2 will be revised as follows:

Startup Feedwater Pump Suction Isolation Valves

~~Motor-operated gate valves are provided on both the suction lines from the deaerator storage tank and condensate storage tank to the startup feedwater pumps. The deaerator valve receives an automatic signal to open with loss of main feedwater pumps or on low deaerator level, and the condensate storage tank valve opens on low deaerator storage tank level. Opening of the valves occurs during periods of operation of the startup feedwater for plant startup, hot standby, or shutdown (automatic link with pump start/stop).~~

Pneumatically-actuated gate valves are provided on both the suction lines from the deaerator storage tank and condensate storage tank to the startup feedwater pumps. The isolation valve from the deaerator storage tank is normally open, and fails close. The isolation valve from the condensate storage tank is normally closed and fails open. The system is designed to automatically switch valve positions from their normal position on low deaerator storage tank level.

The individual pump suction isolation valves are normally open, manually-operated valves.

Question 410.237

Figure 1.7-2 of the SSAR refers to "Reference C," but no such reference is provided. Provide Reference C.

Response:

Reference C is the list of AP600 plant systems, which provides the 3-letter system abbreviations and the full descriptions of the names of the AP600 plant systems. This information for plant systems mentioned in the AP600 SSAR is provided in SSAR table 1.7-2.

SSAR Revision: NONE

Question 435.74

Chapter 3 of the AP600 PRA indicates that the recovery probability used for loss of offsite power was taken from Revision 2, Volume III, Appendix A to Chapter 1 of the EPRI ALWR Requirements Document. The recovery probability given in that revision of the Requirements Document is not appropriate for use in the AP600 design because that probability was derived from plants with two sources of offsite power available during power operating modes. The AP600 design has only one source of offsite power available during power operating modes. The second source of offsite power is described as a maintenance source to be used during plant outages. Revision 4 to Volume III of the EPRI Requirements Document provides appropriate recovery probabilities for offsite power sources like those used in the AP600 design. Therefore, the recovery probabilities provided in Revision 4 of the EPRI document that are appropriate to the AP600 offsite power system design should be used in the AP600 PRA. Address this concern.

Response:

There are two offsite power recovery probabilities modeled for the loss of offsite power event tree. The probabilities are for offsite power recovery in one-half and 24 hours, event tree top events R05 and R24, respectively. These probabilities will be recalculated during the AP600 PRA update. Reflected in this calculation will be the appropriate AP600 offsite power source design.

PRA Revision:

The PRA will be revised by December 31, 1994

SSAR Revision: NONE

Question 440.55

As stated in Section 6.7 of NUREG-1449, the staff has noted instances in which the failure of temporary RCS boundaries (such as freeze seals used to temporarily isolate fluid systems, temporary plugs for nuclear instrument housings, and nozzle dams installed in the hot-leg and cold-leg penetrations to steam generators) can lead to a rapid non-isolable loss of reactor coolant. This concern is of special importance in PWRs because the emergency core cooling system (ECCS) is not designed to automatically mitigate accidents initiated at pressures below a few hundred psig and is not normally fully available for manual use during these conditions. Address this concern with respect to failure of temporary boundaries in the AP600 design (see also Q440.53, Q440.56, Q440.58, Q440.71, and Q440.72).

Response:

The AP600 passive safety-related systems provide the safety-related means for protecting the plant during all modes of operation including shutdown and refueling. The AP600 includes design features to mitigate accidents that occur at low pressures. The passive safety-related systems are designed to either automatically mitigate events that occur during shutdown, or are available f... ...nual actuation, as described in Reference 440.55-1. The AP600 technical specifications identify when the variou... ,ortions of the passive safety-related systems are required to be available. The response to RAI 440.58 provides the AP600 technical specifications for shutdown.

Portions of the passive systems are required to be operable during modes of operation until mode 6, when the refueling cavity has been flooded, and the upper internals have been removed. At this point in time, the stored heat capacity of the water in the refueling cavity is sufficient to maintain the reactor in a safe condition for an extended period of time (at least 72 hours) without need for operator recovery actions in case of a loss of decay heat removal. In all other modes, the availability of the passive safety-related systems is maintained via technical specficiations and the safety-related systems' redundant design features.

The availability of the passive safety-related systems during shutdown as described above significantly reduces the risk associated with failures of temporary RCS boundaries. In addition, the following AP600 design features reduce the risks associated with temporary RCS boundaries:

- Steam Generator nozzle dams - the AP600 steam generator nozzle dams are classified as AP600 Equipment Class C so that the design, manufacture, installation, and inspection of this boundary (when installed) are controlled by the following requirements: 10CFR21; 10CFR50, Appendix B; Regulatory Guide 1.26 Quality Group C; and ASME Boiler and Pressure Vessel Code, Section III, Class 3. In addition, this pressure boundary is classified as Seismic Category I so that it is protected from failure following a safe shutdown earthquake.

- Elimination of temporary plugs for nuclear instrumentation - The AP600 does not contain bottom mounted intrumentation that require temporary plugging during shutdown and refueling. The AP600 utilizes a fixed incore system.

- Current plants remove the excore detectors from above the excore housings through the floor of the refueling cavity. During refueling operations, these holes are plugged to facilitate flooding of the refueling cavity. The AP600 has eliminated these temporary plugs by designing the excore instrumentation to be inserted from below the excore housings.

- Reduced reliance on freeze seals - the AP600 has reduced the potential applications for freeze seals by reducing the number of lines that connect to the RCS and by providing the ability to perform operability tests on many valves that connect to the reactor coolant pressure boundary. This improved IST reduces the requirements for dissasembly of reactor coolant pressure boundary valves to test their operability. See the response to RAI 210.24 for a discussion of the AP600 IST requirements. The use of freeze seals during a forced outage will typically occur in cold shutdown (Mode 5), when the passive core cooling system is required to be available.

References:

440.55-1 WCAP-13793, AP600 Systems / Event Matrix, June 1994.

SSAR Revision: NONE

Question 440.59

Provide a description of the design features that are incorporated into the AP600 design to increase the allowable water level operating band during mid-loop operation to prevent a loss of decay heat removal capability and water flooding the steam generator and containment.

Response:

SSAR section 5.4.7.2.1 describes the design features that are incorporated into the AP600 to increase the allowable water level operating band during mid-loop operation.

SSAR Revision: NONE

Question 440.62

Provide a discussion of design features of non-safety-related systems for normal shutdown operations, procedures, and technical specifications that will reduce the risk to challenge passive safety systems and active non-safety-related systems for accident mitigation during shutdown operations.

Response:

The nonsafety-related systems, including those features that provide functions for normal shutdown operati s, are discussed in their appropriate SSAR sections. Please see Reference 440.62-1 for a discussion of how these nonsafety-related systems reduce challenges to the passive safety-related systems.

Based on the results of the regulatory treatment of nonsafety-related systems (RTNSS) evaluations documented in Reference 440.62-2, no technical specifications are required for the nonsafety-related systems. Reference 440.62-2 provides short term availability recommendations for those portions of nonsafety-related systems that perform functions identified as RTNSS-important.

References:

440.62-1 WCAP-13793, AP600 Systems / Events Matrix, June 1994.

440.62-2 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

Question 440.63

Provide a discussion of analyses that demonstrate that the design of passive safety-related systems and active non-safety-related systems is adequate for mitigating the consequences of loss of ac power, inadvertent RCS drainage, and loss of normal residual heat removal system events during shutdown and mid-loop operations. The information should include the assumptions and methods used in the analyses with a discussion of testing data to support the adequacy of the assumptions and methods, and analytical results.

Response:

The AP600 passive safety-related systems provide the safety-related means for protecting the plant during all modes of operation including shutdown and refueling. The events described (loss of ac power, inadvertent RCS drainage, and loss of normal residual heat removal system) are mitigated as described in Reference 440.63-1. Shutdown events and midloop operations are evaluated in the AP600 PRA and results indicate a low contribution to core damage from shutdown events.

In addition, shutdown events are included in the AP600 implementation of the regulatory treatment of nonsafety-related systems process. As described in Reference 440.63-2, the shutdown events are a low contributor to the focused PRA. The evaluation of shutdown initiating event frequencies does identify RTNSS-important nonsafety-related functions. Those functions and the corresponding regulatory oversight proposed are included in Reference 440.63-2.

For events that occur at hot shutdown, hot standby, or cold shutdown (prior to reduced inventory operations), the full compliment of passive safety-related systems are available to mitigate an event. For reduced inventory conditions, the passive RHR heat exchangers, accumulators, and core makeup tanks are unavailable or ineffective. Prior to and during reduced inventory operations, precautions are taken such as opening the automatic depressurization system (ADS) valves connected to the pressurizer and assuring containment closure capability. During midloop operation, protection is provided by IRWST injection (motor-operated isolation valves in the injection line are closed and operable). These isolation valves would be manually opened following a loss of normal residual heat removal, ac power, or low RCS inventory. In addition, these valves receive signals from the nonsafety-related diverse actuation system to automatically open on a low RCS inventory signal (after a 30 minute time delay).

References:

440.63-1 WCAP-13793, AP600 Systems / Events Matrix, June 1994.

440.63-2 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision: NONE

Question 440.65

Outage and maintenance activities require only minimal isolation of important systems and components. How does the AP600 design take this into consideration?

Response:

The AP600 passive safety-related systems provide the safety-related means for protecting the plant during all modes of operation including shutdown and refueling, when outage and maintenance activities occur. As such, the AP600 technical specifications identify when the various portions of the passive safety-related systems are required to be available. The response to RAI 440.58 provides technical specifications for shutdown.

Portions of the passive safety-related systems are required to be operable during all modes of operation until mode 6, when the refueling cavity has been flooded, and the upper internals have been removed. At this point in time, the stored heat capacity of the water in the refueling cavity is sufficient to maintain the reactor in a safe condition for an extended period of time (at least 72 hours) without need for operator recovery actions in case of a loss of decay heat removal. In other modes, the availability of the passive safety-related systems is maintained via technical specifications and the safety-related systems' redundant and diverse design features.

SSAR Revision: NONE

Question 440.69

Generic Letter (GL) 87-12 requested information regarding lower reactor coolant system (RCS) inventory operation. How does the AP600 design comply with GL 87-12?

Response:

AP600 compliance with Generic Letter (GL) 87-12 , as well as GL 88-17 are discussd in SSAR section 1.9.5, under the paragraph entitled Midloop Operations. In addition, SSAR section 5.4.7.2.1 discusses the AP600 design features addressing midloop operations.

SSAR Revision: NONE

Question 440.71

As discussed in Chapter 2 of NUREG-1449, the staff reviewed operating experience to ensure that its evaluation encompassed the range of events encountered during shutdown and low-power operation. The data base considered by the staff includes licensee event reports, NRC internal reports, and various inspection reports to determine the types of events that take place during refueling, cold and hot shutdown, and low-power operation. Discuss the range of operating experience during shutdown operations considered in the AP600 design and demonstrate that the AP600 design adequately encompasses the shutdown events occurred in the industry (see also Q440.53, Q440.55, Q440.56, Q440.58, and Q440.72).

Response:

In the design of the AP600, Westinghouse has built upon its proven design base of previous plant designs, and has incorporated the lessons learned from operating plants. The AP600 designers review License Event Reports, NRC Generic Letters and Information Notices, and review the INPO SER's and SOER's. References 440.71-1 and 440.71-2 provide a description of the range of shutdown operating experience considered in the AP600 design. In addition, shutdown and low power events are included in the AP600 implementation of the regulatory treatment of nonsafety-related systems process. As described in Reference 440.71-3, the shutdown events are a low contributor to the focused PRA. The evaluation of shutdown initiating event frequencies identifies RTNSS-important nonsafety-related functions. Those functions and the corresponding regulatory oversight proposed are included in Reference 440.71-3.

Chapter 2 of NUREG-1449 provides a retrospective review of events at operating reactors and includes evaluations of the following events:

- loss of shutdown cooling
- loss of reactor coolant inventory
- breach of containment integrity
- loss of electrical power
- overpressurization of reactor coolant system
- flooding and spills
- inadvertent reactivity addition

Two major observations were made as a result of the evaluation of these operating events. The first observation was that a greater percentage of the events were caused by human errors than by equipment errors. The second observation was that the events did not reveal new unanalyzed issues, but instead, appeared to represent an accumulation of errors or equipment failures or a combination of the two.

Westinghouse has considered the range of events evaluated in NUREG-1449 and specific design features have been incorporated to address the potential shutdown events and these are discussed below:

Loss of Shutdown Cooling

As described in NUREG-1449, the loss of shutdown cooling can be initiated by the loss of flow in the RHR or support systems. The AP600 passive safety-related systems have shutdown cooling capability, and provide protection from a loss of shutdown cooling event. See the response to RAI 440.63 for a discussion of how the passive safety-related systems provide shutdown cooling following an event. The nonsafety-related shutdown cooling systems (normal RHR, component cooling water, service water) provide the normal means of shutdown heat removal.

NUREG 1449 explains that loss of shutdown cooling arises due to the problems associated with midloop operations. The AP600 design features that address reduced reactor coolant system inventory shutdown operations are discussed in SSAR Subsection 5.4.7.2.1. These design features are specifically incorporated as a result of the industry experience described in the referenced reports. These features include:

- Loop piping offset that allows midloop operations to be performed at a much higher RCS water level
- Step nozzle connection that significantly lowers the minimum acceptable RCS level during midloop
- No RHR throttling is required during midloop, even with the RCS at saturated conditions
- Self-venting suction line that eliminates local high points allows for immediate restart of the RHR pumps after air-binding
- Redundant, remote hot leg level readout in the main control room
- Improved RCS draining procedures

In addition, SSAR Subsection 1.9.5 addresses the licensing issues contained in SECY 90-016 which includes midloop operations. Please refer to the responses to RAI 210.037 and 210.061 for SSAR revisions of these sections.

Loss of Reactor Coolant Inventory

A review of the operating experience shows that loss of reactor coolant inventory events at shutdown have occurred for various reasons such as opening of the RHR relief valve, PORV and block valves opening simultaneously during testing, and loss of pressure in the reactor cavity seal ring allowing drainage from the cavity. The AP600 has addressed this issue by providing safety-related protection against a loss of inventory during shutdown. See the response to RAI 440.63 for a description of the safety-related protection against a loss of RCS inventory.

In addition, the AP600 has incorporated other features to reduce the probability of a loss of inventory during shutdown. The set pressure for the normal RHR relief valve is higher than typical PWRs, and therefore challenges to this relief valve should be fewer. The AP600 does not use power-operated relief valves nor does it employ power-operated valves to provide low temperature overpressure protection. This reduces the probability of a spurious opening of a power-operated valve during shutdown operations.

Breach of Containment Integrity

As described in the NUREG, a breach of containment coupled with postulated events could increase the severity of an event. The AP600 has reduced the requirements for containment integrity at shutdown operations to mid-loop operations. Results provided in SSAR Chapter 15 demonstrate acceptable offsite doses for the limiting fuel drop accident without containment integrity. Therefore, the AP600 is less susceptible to problems associated with a breach of containment integrity.

Loss of Electrical Power

The safety significance of a loss of electrical power depends on the part of the plant affected. Since the AP600 safety-related systems do not rely on ac power to perform their safety-related function, the probability of a loss of electrical power resulting in a loss of safety function is greatly reduced. This is demonstrated by the results of the AP600 PRA report, and the results of the focused PRA presented in Reference 440.71-3.

Overpressurization of Reactor Coolant System

These events are precursors to exceeding the reactor vessel brittle fracture limits. The AP600 has addressed this issue with design features and technical specifications. Whereas current PWRs rely on the power-operated relief valves to protect the RCS from a cold overpressure transient, the AP600 employs a passive relief valve to protect the RCS. The reliability of the RHR relief valve to protect the plant is higher than of power-operated valves. Technical specifications and administrative procedures are employed to reduce the probability of a low temperature overpressure event.

Flooding and Spills

The safety significance of flooding or spills depends on the safety-related equipment affected. The AP600 includes flooding protection for safety-related equipment, valves, and instrumentation. The AP600 the passive safety-related systems are designed to flood portions of the containment to mitigate an accident. Consequently, safety-related equipment, valves, and instrumentation that are located in areas designed for flooding are qualified to perform their safety-related functions under flooded conditions. This results in the AP600 being less susceptible tan current PWRs to flooding events that occur at shutdown. SSAR Section 3.4 provides the description of the AP600 provisions for flood protection. PRA section 9.1 provides the AP600 internal flooding probabilistic risk assessment.

Inadvertent Reactivity Addition

The AP600 has safety-related protection from boron dilution events that could occur during shutdown from mode 1 through mode 5. Technical specifications are provided to protect the plant from reactivity events that occur in mode 6.

References:

440.71-1 WCAP-14115, Review of Plant Operating Experience in the Application of the AP600, July 1994.

440.71-2 WCAP-13559, Operational Assessment for AP600, December 1992.

440.71-3 WCAP-13856, AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, September 1993.

SSAR Revision:

Please refer to the responses to RAI 210.037 and 210.061 for revisions of SSAR sections 1.9.5 and 5.4.7.2.1.

Question 440.76

Section 5.4.9.2 of the SSAR states that the routing of pipe between the pressurizer and the safety valves does not include a loop seal, and that any condensation of steam in the connecting pipe up to the valve will drain back to the pressurizer and not collected as a slug of water to be discharged during the initial opening of the valves.

a. Though the safety valves are designed for relief of saturated steam, they may still be subject to water discharge. Provide a discussion regarding the effects of passage of a water slug on the safety valves, such as water hammer.

b. What are the water relief rates assumed in the loading analysis? Are the water relief rates used consistent with test results obtained from similar valves?

Response:

a. The AP600 safety valves are designed and qualified for relief of steam. As demonstrated by the SSAR Chapter 15 safety analysis of design basis events, the safety valves are not required to relieve water. Consistent with the design bases of most current PWRs, safety valve operation with water is considered beyond the design basis of the valves.

b. The loading analysis considers a conservative steam relief flow rate based on a performance evaluation of the safety valve. Water relief is not considered in the loading analysis.

SSAR Revision: NONE

Question 440.81

Technical Specification LCO 3.4.13 for the LTOP system requires that, with the accumulators isolated, either the NRHRS suction relief valve or the RCS depressurized with an open RCS vent of greater than or equal to 5.4 square inches be operable.

a. This appears to allow the use of an RCS vent when the RCS is depressurized in lieu of the NRHRS relief valve for LTOP. Discuss this matter in Section 5.2.2 of the SSAR? How is the RCS vent size determined?

b. As a resolution of Generic Issue 94 regarding LTOP, Generic Letter (GL) 90-06 states that the LTOP availability should be ensured by limiting the allowable outage time to 24 hours for a single LTOP channel while operating in Modes 5 and 6. Explain how LCO 3.4.13 requiring operability of either the relief valve or the RCS depressurization comply with GL 90-06.

c. If the NRHRS suction relief valve is inoperable, Action Item C of LCO 3.4.13 requires either that the relief valve be restored to operable status or that the RCS be depressurized and the RCS vent be established within 8 hours. Has an analysis been made to determine whether the RCS can be overpressurized within the 8 hours when the relief valve is inoperable and the RCS depressurization is not initiated? Address this concern.

Response:

The normal residual heat removal system (RNS) relief valve provides low temperature overpressure protection during shutdown operations at low reactor coolant system temperatures. If the normal residual heat removal system relief valve is inoperable, the Technical Specifications require the reactor coolant system to be depressurized, and a vent of 5.4 square inches to be opened to protect the reactor vessel from low temperature overpressure events. A vent size was selected to bound the area of the normal residual heat removal system relief valve inlet pipe. Since the actual valve area will be less than the area of the inlet piping, an area equal to that of the valve inlet area is judged to be sufficient as a minimum vent area required when the relief valve is out of service. This allows the operator to use one or more automatic depressurization system paths in order to provide a vent path when the normal residual heat removal system relief valve is out of service. See the response to RAI 440.078 for a discussion of the sizing basis for the normal residual heat removal system relief valve.

b. An LTOP channel refers to a low temperature overpressure protection system (LTOP) system that requires instrumentation to actuate relief valves. The AP600 LTOPS employs a passive device to provide this function. If the normal residual heat removal system relief valve is not available, the reactor coolant system is required to be depressurized, and a vent path opened. With the reactor coolant system depressurized and the vent path opened, the Appendix G pressure limits for the AP600 reactor vessel will not be exceeded for the design basis LTOP events.

c. The reactor coolant system could become pressurized above the Appendix G limits if a design basis LTOP event occurred when the valve was inoperable, and the reactor coolant system vent was not established. However, the probability of such an event is very low considering:

- the amount of time that the valve is required to be available is very low (modes 4 and 5, reactor coolant system temperature $\leq$ 350°F, until mode 6 when the reactor vessel head is removed)

- the normal residual heat removal system relief valve is a passive device that requires no power or control logic and has a very low unavailability frequency

Current LTOP systems that use active valves can have one channel out for 24 hours. The lower allowable outage time for the AP600 LTOP is an improvement.

SSAR Revision: NONE

Question 440.85

In the discussion of the passive core cooling system (PXS) design basis for emergency core makeup and boration for non-LOCA events, Section 6.3.1.1.2 of the SSAR states that following either small or large steam line break events, the RCS is automatically brought to a subcritical condition, consistent with the passive containment cooling capabilities. Clarify the relationship between subcriticality and the passive containment cooling capabilities, and the phrase "automatically."

Response:

The passive core cooling system (PXS) provides boration and makeup to the reactor coolant system as well as core cooling. The passive containment cooling system provides containment cooling and pressure reduction.

During a steam line break, the reactor trip initially shutdowns the reactor to a subcritical condition. Safety analysis with conservative reactivity assumptions indicates that the reactor may return critical for a time until core makeup tank boration is able to make it subcritical again. Refer to the response to RAI 440.107 for additional discussion. During a steam line break, the core makeup tanks operate in a natural circulation mode with hot water flowing to the core makeup tanks while cold core makeup tank water flows to the reactor. This mode of operation provides boration and makeup to the reactor coolant system. In a steam line break accident, the core makeup tanks are automatically actuated on a low reactor coolant system cold leg temperature signal and no operator action is needed to allow for the core makeup tanks to complete their boration function. The mass/energy input to the containment is affected by the period of time that the reactor returns to power during a steam line break. The passive containment system is sized to remove this energy.

SSAR Revision: NONE

Question 440.87

Section 6.3.1.1.2 of the SSAR states that for safe shutdown, the passive core cooling system is designed to supply sufficient boron to the RCS to maintain the Technical Specification (TS) requirements for shutdown margin (with the CVCS unavailable). TS LCO 3.1.1 and 3.1.2 specify that the required shutdown margins for various modes of plant conditions, and if they are not met, require the operator to initiate boration to restore shutdown margins to within the specified limits without identifying the boron injection source, i.e., the CVCS or the CMT.

a. Discuss the required boron concentrations to maintain the shutdown margin for cold and post-depressurization conditions, and the source of boration to be used to implement LCO 3.1.1 and 3.1.2.

b. Describe how boron injection to the RCS is accomplished using the passive core cooling system alone.

c. Identify any deviations from the boron concentrations in current operating plant requirements.

Response:

a. For cold shutdown conditions (pressurized or depressurized), the required boron concentration to maintain shutdown margin is less than 1000 ppm. The chemical and volume control system is normally used to provide this Reactor coolant system boration and to correct it in case the technical specifications shutdown margin is not maintained. If the nonsafety-related chemical and volume control system is not available the core makeup tanks provide a safety-related means to add boron to the reactor coolant system in most operating modes (i.e. T.S. modes 2 - 5 filled). When the core makeup tanks are not required to be available by the technical specifications (modes 5 not filled), the IRWST provides the safety-related boration function. The IRWST also provides a backup to the core makeup tanks.

b. The core makeup tanks provide boration in these shutdown modes by natural circulation, with hot cold leg water circulating up to the core makeup tank and cold core makeup tank water circulating back to the reactor. The core makeup tanks can also provide boration when the cold legs are drained.

   In mode 5 not filled, the IRWST provides boration through injection by gravity head. In this mode, automatic depressurization system stages 1/2/3 are required by the technical specifications to be open which allows for IRWST injection. In other shutdown modes, the IRWST provides a backup to the core makeup tanks, although the automatic depressurization system valves may have to be opened to reduce the reactor coolant system pressure and allow the IRWST to inject.

c. The AP600 boron concentrations are similar to the boron concentrations in current operating plants. AP600 stores boric acid reduced concentrations, less than 4500 ppm, that do not require heat tracing or special room heating.

   Current plants store boric acid at concentrations as high as 21,000 ppm, which requires heat tracing.

SSAR Revision: NONE

Question 440.88

Section 6.3.1.1.6 of the SSAR discusses the PXS reliability requirements, and states that "Subsection 6.3.1.2 includes specific non-safety-related design requirements" that help to confirm satisfactory system reliability. Section 6.3.1.2 does not appear to address these design requirements. Provide a discussion of these specific non-safety-related design requirements.

Response:

Section 6.3.1.2 of the SSAR contains the following specific passive core cooling system (PXS) nonsafety-related design requirements that relate to system reliability.

- Two 100% capacity passive residual heat removal heat exchangers are required. This capability provides redundancy to allow for the isolation of an individual heat exchanger in the event of tube leakage, without forcing a plant outage in order to repair the leaking heat exchanger tubes.

- The frequency of automatic depressurization system (ADS) actuation must be limited to a low probability. This requirement is intended to improve plant availability by avoiding unnecessary automatic depressurization system actuations.

- Passive core cooling system equipment is located so that following a automatic depressurization system actuation it is not flooded or it is designed so that it is not damaged by flooding. This requirement reduces the plant outage time associated with automatic depressurization system actuation.

- The passive core cooling system is also capable of supporting component maintenance, including capabilities to isolate and drain equipment.

SSAR Revision: NONE

Question 440.92

Sections 6.3.1.1.4 and 6.3.2.1.1 of the SSAR state that the passive residual heat removal (PRHR) heat exchangers, in conjunction with the passive containment cooling system (PCCS), has the capability to bring the plant to safe shutdown conditions, cooling the RCS to about 400°F in 72 hours for non-LOCA events, and can provide core cooling for an indefinite period of time based on the assumption that the PCCS is operable for indefinite period of time to promote condensation of the steam from and return the condensate to the IRWST.

a. Is this PRHR system capability of cooling RCS to 400°F in 72 hours consistent with the requirement specified in the EPRI ALWR Utility Requirements Document for passive plants that the PRHR system shall have sufficient capacity to reduce coolant temperature to 420°F within 36 hours of reactor shutdown?

b. The PCCS is designed with a 72-hour capacity, and relies on non-safety equipment to replenish the lost water in the reservoir beyond 72 hours. Justify Westinghouse's position that the PCCS will be operable for an indefinite period of time.

c. For the condensate to return directly to the IRWST, an isolation valve in each of the two gutters that normally drain to the containment sump will be shut when the PRHR heat exchangers actuate. What is the design requirement of the isolation valve to ensure its closure upon PRHR actuation?

d. The IRWST water inventory is sufficient to provide the PRHR heat exchanger operation for 72 hours without recovery of the condensate. Will the heat exchangers remain totally submerged in the IRWST water to maintain full operational capability for 72 hours? If not, how are the progression of the uncovery of the heat exchangers during transients and the degradation of the heat removal capacity accounted for in the safety analyses?

Response:

a) The Advance Light Water Reactor (ALWR) Utility Requirements Document (URD) does not represent regulatory requirements. However, the design of the passive residual heat removal system is consistent both with the ALWR URD requirement 5.3.2.2 and with the above indicated SSAR statement. The passive residual heat removal heat transfer capability is very effective when there are relatively high differential temperatures between the reactor coolant and the IRWST cooling water. As a result the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to 420°F in 36 hours and to 400°F in 72 hours. This long term operation of the passive residual heat removal heat exchanger is dependent on two nonsafety-related features, ac power to the 1E battery chargers and the condensate return to the IRWST. If either nonsafety-related feature is not available then automatic depressurization system is automatically actuated which places the plant in a safe shutdown condition.

In case ac power is not available, a timer automatically actuates automatic depressurization system at about 23 hours after loss of power. In case condensate does not return to the IRWST, the passive residual heat removalheat exchanger will boil down the IRWST level such that after 72 hours the reactor coolant system

temperature will increase to a level that actuates core makeup tanks. As the IRWST continues to boil down the passive residual heat removal heat exchanger will become less effective and the reactor coolant system will start to steam through the pressurizer safety valves. If this condition continues the core makeup tank level will eventually drop to the point where automatic depressurization system is actuated.

b) The safety case assumes that within 72 hours offsite assistance is secured to provide makeup water to the passive containment cooling system. Safety-related connections are provided to facilitate this makeup. Nonsafety-related makeup to the passive containment cooling system from installed plant equipment is not credited in the safety case. Westinghouse has also performed a beyond design basis analysis where makeup water to the passive containment cooling system was not provided. In this case, the passive containment cooling system water cooling ended at 72 hours and after that time there was only air cooling. Air cooling alone was sufficient to keep the containment pressure below its design pressure.

c) The condensate return to the IRWST is not necessary to meet NRC safety criteria. As a result the isolation valve design is nonsafety-related.

d) With no condensate recovery the passive residual heat removal heat exchanger does not remain completely covered during the 72 hours. Uncovery of the heat exchangers would not occur for about 20 hours, which is beyond the time frame of the detailed safety analysis contained in Chapter 15 of the SSAR. Calculations of long term passive residual heat removal heat exchanger performance have been performed. The performance of the heat exchanger is based on the passive residual heat removal heat exchanger test. These calculations indicate that the degradation of the passive residual heat removal heat exchanger heat transfer performance as it uncovers is compensated for by the reduction in decay heat. The reactor coolant system temperature gradually increases in the long term, approaching hot standby conditions in the 72 hour time frame.

SSAR Revision: NONE

Question 440.100

Table 6.3-4 of the SSAR provides the normal-, actuation-, and failed-positions, respectively, of the remotely actuated valves used by the various passive core cooling system components.

a. There are many valves which are normally closed and whose opening positions are important in the prevention/mitigation function of the PXS, such as the ADS MOVs and the sump recirculation line MOVs. Explain why these valves are designed to "fail as is" rather than move to the "fail safe" positions, e.g., move to the open position, even though the failure modes and effects analysis in Table 6.3-6 indicates that there is no safety-related effects for "fail as is."

b. Section 6.3.2.2.7.4 of the SSAR states that these motor-operated isolation valves have various interlocks, automatic features, and position indications. Provide a description of the interlocks and the bases for these interlocks.

Response:

a. For the automatic depressurization system (ADS) motor-operated valves (MOVs), fail-as-is is the fail safe position. During normal operation and the initial phase of an accident (prior to automatic depressurization system actuation) these valves are clo ed. If these valves opened it would lead to an unnecessary automatic depressurization system actuation. Therefore, fail open valves would not be appropriate. Following automatic depressurization system actuation, these valves are open. These valves must remain open for continued core cooling. Therefore, fail closed valves would not be appropriate. These valves are safety-related and use safety-related power (1E dc power) to change position.

   For the containment recirculation sump valves (V117 A/B, V118 A/B) fail-as-is is the safe position. During normal plant operation and during the initial phase of an accident (prior to containment recirculation) these valves are closed. If these valves were to open, the in-containment refueling water storage tank (IRWST) water would drain directly into the sump, thus depleting IRWST inventory available for injection. Therefore, fail open valves would not be appropriate. After these valves are opened for recirculation, they must stay open for continued core cooling. Therefore, fail closed valves would not be appropriate. These valves are safety-related and use safety-related power (1E dc power) to change position.

b. Refer to SSAR section 7.6.2 for a description of passive core cooling system interlocks.

SSAR Revision: NONE

Question 440.104

Section 6.3.2.5.1 of the SSAR states that the PXS has been "specifically designed to treat check valves failures to reposition as active failures." The core makeup tank discharge line contains two tilt-disc check valves in series. The FMEA in Table 6.3-6 does not consider the failure modes for these check valves because they are not considered active failures as they are normally open and remain in the same position on demand. However, for an accident where the accumulators discharge into the RCS, these check valves will close to prevent backflow into the CMT, and will have to reopen to inject borated water into the RCS.

a. The arrangement of two check valves in series does not meet the single failure consideration. Either modify the CMT discharge line check valve arrangement, or provide justification for treating these check valves as passive components. Also, provide the results of the FMEA analysis for these CMT discharge line check valves.

b. Describe the CMT discharge line check valve design to discuss how they are normally maintained open.

c. Technical Specification 3.5.2 specifies the LCO and Surveillance Requirements (SR) for the CMT. Why are there no SRs to verify that the CMT outlet check valves are open, and no action requirement when the check valves are not open?

Response:

Since these check valves sit in an open position for up to two years between inservice tests that exercise them, the main concern is their failure to close. As a result, series check valves have been installed to account for the possibility of a single normally open check valve failing to close. The differential pressure available to close them is limited to the differential pressure generated in the direct vessel injection line during accumulator injection. This differential pressure is sufficient to overcome the counterbalance weight that normally keeps the valves open. Because of the counterbalance weight, no flow/differential pressure is required to re-open the valve. Since more flow/differential pressure is required to close the valve than open it, it is unlikely that the check valves would not re-open as soon as the accumulator flow slows down. This position is consistent with the PRA which treats standby check valve failures as 2.0 E-7 fail/hr. The chance of a standby core makeup tank check valve failing between refueling outages is then 1.75 E-3. The chance of a core makeup tank check valve failing to re-open 5 minutes after it closes is therefore only 1.66 E-8, or 5 orders of magnitude less. As a result, parallel check valves have not been incorporated. Adding parallel check valves would increase the complexity of the piping design (thermal stresses), increase the chance of a reactor coolant system leak or a LOCA, reduce the chance that the accumulator bypass flow will be isolated, and increase testing and maintenance which increases radiation exposure.

These check valves are tilt disk check valve designs which are counter balanced such that they normally hang in a fully open position. Because the disk sits completely in the flow stream it closes quickly with reverse flow.

The technical specifications for the core makeup tanks do not address the normal position of these check valve because they will normally be in their proper position and during power operation there is no way to change their position. This approach is consistent with how the accumulator check valves are handled in current plants.

SSAR Revision:

Revise the fourth paragraph of Subsection 6.3.2.5.1 to include re-opening of the core makeup tank check valves as an exception to the general treatment of check valves as single failures as follows:

There are two exceptions to this treatment of check valve failures in the passive core cooling system. One ~~The only~~ exception to this treatment ~~in the passive core cooling system~~ is made for the accumulator check valves, which is consistent with the treatment of these specific check valves in currently licensed plant designs. The other exception is made for the core makeup tank check valves failure to re-open after they have just closed during an accident. This exception is based on the low probability of these check valves not re-opening within a few minutes after they have cycled closed during accumulator operation.

Question 440.112

Section 6.3.3.4.2 of the SSAR states that, with a loss of the normal RHR system when the RCS pressure boundary is intact, the PRHR heat exchangers provide the safety-related heat removal path, and that the PRHR heat exchangers can remove sufficient heat to maintain the RCS within the NRHRS design limits (400°F) and permit the NRHRS to be placed back in operation when it becomes available.

a. How long can the NRHRS be out of service while relying on the PRHR system to maintain the RCS at 400°F?

b. Provide a safety analysis of LOCA and other transient events that may be initiated at this condition.

Response:

a. The normal residual heat removal system (RNS) can remain out of service indefinitely during the scenario described above. The normal residual heat removal system is not a safety-related system and is not required to maintain safe shutdown conditions, and is not required to mitigate design basis accidents. The normal residual heat removal system is not credited in Chapter 15 accident analyses.

b. The LOCA and other transient analyses provided in Chapter 15 of the AP600 SSAR bound a LOCA that could occur at the conditions described above. See the response to RAI 440.63 for a discussion of the analyses that demonstrate the adequacy of the passive safety-related systems to mitigate the consequences of events that occur at shutdown.

SSAR Revision: NONE

Question 440.113

In the discussion of the loss of the NRHRS during mid-loop operation, Section 6.3.3.4.3 of the SSAR states that the IRWST isolation valves automatically open via a signal from diverse actuation system, after a delay, if the non-safety-related RCS hot leg level indication decreases below an established setpoint. This section also states that the operator can remotely open the CMT and accumulator isolation valves to provide additional makeup water injection, if required.

a. What is the justification for crediting the non-safety-related hot leg level indication to cope with a loss of the NRHRS?

b. The AP600 design basis is to use the passive safety systems and preclude operator action for 72 hours. What is the basis for crediting operator action during a loss of the NRHRS?

c. Provide a safety analysis of the loss of the NRHRS during mid-loop operation taking no credit for the non-safety related systems or components, nor operator actions.

Response:

a. The reactor coolant system hot leg level indication has recently been upgraded to be safety-related. The SSAR will be updated to reflect this change. Please see the responses to RAI 440.056 and 440.162 for a related discussion.

b. The design basis of the AP600 does not preclude the use of operator actions for 72 hours following design basis events. Operator actions are not credited for at least 30 minutes following an event. The results of the AP600 accident analyses are provided in Chapter 15 of the SSAR.

c. See the response to RAI 440.63 for a discussion of the analyses that demonstrate the adequacy of the passive safety-related systems to mitigate the consequences of events that occur at shutdown.

SSAR Revision:

SSAR Table 7.5-1 will be modified as follows:

Table 7.5-1 (Sheet 3 of 12)

Post-Accident Monitoring System

Variable	Range/ Status	Type/ Category	Qualification		Number of Instruments Required (Note 1)	Power Supply	QDPS Indication (Note 2)	Remarks
			Environmental	Seismic				
Startup feed-water flow	0-1000 gpm	D2, F2	Harsh	Yes	1/pump	1E	Yes	
Startup feed-water control valve status	Open/ Closed	D2	None	None	1/valve	Non-1E	No	
Containment pressure	-5 to 10 psig	B1, C2, D2, F2	Harsh/Mild	Yes	3 (Note 4)	1E	Yes	Located outside containment
Containment pressure (extended range)	0 to 180 psig	C1	Harsh	Yes	3 (Note 4)	1E	Yes	
Containment area radiation (high range)	10^0-10^7 R	B1, C2, E2, F2	Harsh	Yes	2 1/division	1E	Yes	Diverse measurement: sampling
Hot leg water level	0-100%	B3	Harsh~~None~~	Yes~~None~~	1/unit	~~Non-~~1E	Yes~~No~~	
RCS boric acid concentration	N/A	B3	None	None	N/A	N/A	No	Manual sampling
RCS activity	N/A	C3	None	None	N/A	N/A	No	Manual sampling
Plant vent radiation	(Note 3)	C2, E2	Mild	None	1	Non-1E	No	
Containment isolation valve status	Open/ Closed	C2	Harsh/mild	Yes	1/valve	1E	Yes	Separate divisions on series valves
Boundary environs radiation	N/A	C3, E3	None	None	N/A	N/A	No	Site specific
Hydrogen concentration	0-10%	B1	Harsh	Yes	3 (Note 4)	1E	Yes	
Hydrogen recombiners	On/Off	D2	None	None	1/recombiner	Non-1E	No	

Question 440.123

Section 5.4.1.2.1 of the SSAR states that the canned-motor reactor coolant (RC) pumps have been used in commercial nuclear plant service. Provide historical information regarding the use of these pumps in the commercial nuclear power plants and their reliability and performance records.

Response:

Canned reactor coolant pumps have been used in the commercial nuclear industry and in fossil fuel plants since the early 1960's. The following is the list of the Westinghouse canned-motor reactor coolant pumps that are or have been in commercial nuclear operation.

Westinghouse Canned Motor Reactor Coolant Pumps

Plant	Number
Shippingport Core 1	4
Yankee Rowe	4
Indian Point 1	8
SELNI(Italy)	4
SENA(Belgium)	4
Shippingport Core 2	4

Presently data is only available on four out of the six plants identified in the above list. Performance data is not presently available for the SELNI and SENA plants since they are foreign plants. The data that is available for the domestic applications must be reviewed to determine the applicability of the performance records to the AP600 canned-motor pump design. The lessons learned from these early plants have been incorporated into the AP600 to further enhance canned-motor pump reliability. Several examples of these design modifications include the removal of vent lines and penetrations from the canned-motor pumps. The canned-motor pumps in the above mentioned plants were oriented with the hydraulic end down, and thus a system was needed to relieve any entrapped air. Problems with the venting system were instrumental in the design decision to orient the AP600 canned-motor pumps with the hydraulic end up so that they are self venting. A second modification, that will be incorporated into the AP600 canned-motor pump, includes the method of sealing the main flange. Many of the early canned-motor pumps used hard nickel gaskets and as a result several pump flanges deformed and produced leakage. The design of gaskets has progressed significantly since the 1960's. The AP600 will utilize a softer compound gasket which has sealing capabilities superior to the hard gaskets.

A Failure Modes Analysis was performed in which the available data was reviewed for applicability. An unavailability allocation was then calculated using the applicable canned-motor data supplemented by applicable data recorded on shaft seal pumps. The resulting unavailability allocation is 10.5 hrs/yr per plant or 2.6 hrs/yr per pump.

SSAR Revision: NONE

Question 440.124

Table 5.4-1 of the SSAR provides the canned-motor RC pump design parameters, including the design flow rate, developed head, and motor/pump rotor moment of inertia.

a. Provide a pump head-capacity characteristic curve.

b. Does the value of the moment of inertia include the moment of inertia of the flywheel and all other rotating parts?

c. With regard to the canned-motor pump performance, Section 5.4.1.3.1 of the SSAR states that minimum NPSH requirements are not required to ensure safe pump operation because the required net positive suction head (NPSH) is provided with ample margin to minimize the potential for cavitation. It further states that the required NPSH for the impeller represents a 3 percent head drop-off condition, and the recommended plant available NPSH represents 100 percent margin, or twice the required value. Elaborate on the special requirements and/or restrictions on the design and operation of the AP600 canned-motor pumps to achieve these goals.

Response:

a. Table 440.124-1 provides the preliminary pump head-capacity curve.

b. The reactor coolant pump moment of inertia identified in Table 5.4-1 represents the moment of inertia of the entire rotating assembly. The rotating assembly includes the rotor and shaft, flywheel, impeller, and the other parts attached the rotating assembly.

c. There are no special requirements and/or restrictions related to the design and operation of the canned-motor pumps to achieve the net positive suction head (NPSH) margins, during normal operation. The required NPSH is well within the operating reactor coolant system pressure during heatup, cooldown, and power operations, with four pumps running. It may be necessary to restrict the operation of certain combinations of pumps when running at low reactor coolant pressures approaching the cut-in pressure of the normal residual heat removal system.

SSAR Revision: NONE

TABLE 440.124-1
PRELIMINARY PUMP HEAD-CAPACITY
CHARACTERISTIC CURVES

Flow (gpm)	Head (Feet)
0	463
10,000	405
20,000	393
27,500	391
35,000	360
40,000	326
45.000	288
51,000	240
55,000	206
60,000	163
65,000	117
70,000	55

Question 440.130

The RNS is designed to provide defense-in-depth functions for cooling the RCS during shutdown operation.

a. Provide an analysis to demonstrate that the RNS RHR has the capability to ensure that the specified design limits are not exceeded for the plant infrequent and moderate frequency events during shutdown cooling mode.

b. For an RNS pipe break outside containment that disables one train of the RNS while in a shutdown cooling mode operation, provide the following information: (1) the maximum discharge rate, (2) the capability of the intact RNS train, (3) the elapsed time before unacceptable consequence without operator recovery actions, (4) the alarms that are available to alert the operator of the event, (5) the recovery procedures, and (6) operator recovery action time.

Response:

a. The normal residual heat removal system (RNS) is not a safety-related system and as such, is not required to mitigate design basis accidents. The normal residual heat removal system is not credited in the Chapter 15 accident analyses. The capabilities of the normal residual heat removal system are discussed in SSAR section 5.4.7. An evaluation of events that occur at shutdown is provided in section 8.4 of the AP600 PRA report.

b. The break of an normal residual heat removal system line outside containment is not considered a design basis event because the system is not aligned during plant power operation. The system is considered a moderate energy system. The normal residual heat removal system is not in operation when the reactor coolant system is at high pressure, and is not assumed to be a credible break source for a LOCA and is therefore not considered in Chapter 15 accident analyses. The intersystem LOCA issue is discussed in SSAR subsection 5.4.7.2.2.

If a leak were to occur in the normal residual heat removal system during shutdown operations, the operator would be alerted to this by one or more of the following instruments:

- pressurizer level (wide range)
- hot leg level
- refueling cavity level
- spent fuel pit level
- auxiliary building sump level

During shutdown, safety-related core cooling is provided by passive means, either by the passive core cooling system, or the water in the refueling cavity. Sufficient time is available to the operator (> 30 minutes) to detect the leak in the normal residual heat removal system isolate the normal residual heat removal system (if necessary) and align the passive core cooling system (if necessary). In addition, the nonsafety-related spent fuel pit cooling system can be aligned to directly cool the refueling cavity. Passive safety-related systems are

available to provide makeup and core cooling during shutdown operation. See the response to RAI 440.63 for a discussion of the safety-related protection from events that occur at shutdown.

SSAR Revision: NONE

Question 440.134

The following questions are related to the RNS isolation valves:

a. Section 5.4.7.2.2 of the SSAR indicates that the RNS contains an instrumentation channel indicating the pressure in each pump suction line, and a high pressure alarm in the control room. Does the RNS also have position indication in the control room for the isolation valves on both the suction and discharge sides of the RNS as discussed in BTP RSB 5-1?

b. What evaluations have been made to address common cause failures (e.g., failure of the outboard containment isolation valve on the RNS common suction line, or failure of a pressure transmitter applied to the inner or outer isolation valves in the suction lines) that could disable both trains of RNS? What are the effects of these common cause failures?

c. If a failure of the pressure transmitter to the RNS suction valve interlocks occur, what operator actions and procedures are necessary and where must they be performed under those conditions to initiate the RNS when the RCS pressure is below 450 psig?

d. Because the closure of the manual maintenance valves upstream of the RNS pumps could isolate the RNS, describe the procedures and controls that eliminate the possibility of manual initiation of valve closure during shutdown operation.

Response:

a. Yes.

b. A common cause failure in the normal residual heat removal system (RNS) would require corrective action by the operator to establish the reactor decay heat removal via the normal residual heat removal system. If the corrective action cannot be accomplished, initiation of another means of decay heat removal is required to continue core cooling. Safety-related reactor decay heat removal is provided by the passive safety-related systems. In addition, nonsafety-related startup feedwater system and the nonsafety-related spent fuel pit cooling system could be available for decay heat removal.

c. To open the normal residual heat removal system suction isolation valves interlocked with the failed transmitter, the operator would have to bypass the failed transmitter's signal in the integrated protection cabinets.

d. The manual maintenance valves upstream of the normal residual heat removal system pump are administratively locked open during all plant conditions.

SSAR Revision: NONE

Question 440.137

Item A.1.d of Appendix A to Regulatory Guide (RG) 1.68 states that preoperational testing for several active systems, including residual heat removal system, should be performed to verify operability, redundancy, and electrical independence. Appendix 1A of the SSAR indicates that the AP600 position conforms to this RG, but further states that these systems have been eliminated by the design of the AP600 passive safety systems. Clarify this statement.

Response:

Item A.1.d of Appendix A to Regulatory Guide 1.68 requires operability testing of systems and design features provided or relied on to dissipate or channel thermal energy from the reactor to the atmosphere or to the main condenser or other systems following off-normal conditions or anticipated transients, including reactor scram.

The AP600 conforms to this item because the safety-related decay heat removal (as credited in the plant safety analyses) is accomplished by the safety-related passive core cooling system. The preoperational test program for the passive core cooling system satisfies Item A.1.d of Regulatory Guide 1.68.

Preoperational testing of the AP600 systems and components that perform nonsafety-related decay heat removal (i.e., not credited in the plant safety analyses) is done to demonstrate their design performance requirements.

The AP600 Preoperational Testing Program is presented in SSAR Chapter 14.

SSAR Revision:

Appendix 1A of the AP600 SSAR will be revised to clarify this statement as show below:

Reg. Guide 1.68, Rev. 2, 8/78 - Initial Test Program for Water-Cooled Nuclear Power Plants

C.1	App. A.1.d	Conforms	~~These systems have been eliminated due to the design of the AP600 passive safety systems. The functions of these systems are replaced by the PRHR heat exchangers of the passive core cooling system.~~ **The AP600 conforms to this item because the safety-related decay heat removal (as credited in the plant safety analyses) is accomplished by the safety-related passive core cooling system. The preoperational test program for the passive core cooling system satisfies Item A.1.d of Regulatory Guide 1.68.** **Preoperational testing of the AP600 systems and components that perform nonsafety-related decay heat removal (i.e., not credited in the plant safety analyses) is done to demonstrate their design performance requirements.**

Question 440.138

Section 5.4.7.5 of the SSAR states that the design of the RNS has been compared with the acceptance criteria set forth in Section 5.4.7 and BTP RSB 5-1 of the SRP, as appropriate. Discuss how the AP600 design complies with the Section 5.4.7 and BTP RSB 5-1 of the SRP. For those acceptance criteria or the staff positions that the AP600 design does not comply with, provide bases and justifications for these deviations. Address the following concerns:

a. BTP RSB 5-1 states that the RHR system isolation valves in the suction lines should have independent diverse interlocks to prevent the valves from being opened, unless the RCS pressure is below the RHR system design pressure. The two parallel sets of two in-series isolation valves in the RNS suction lines in the AP600 are designed with interlocks to prevent them from being opened by the operator when the RCS pressure is above 450 psig. Section 7.6.1.1.1 of the SSAR states that the logic for the outer valves is identical to that provided for the inner isolation valves, except that equipment diversity is provided by virtue of the fact that the pressure transmitter used for valve interlocks on the inner valves is diverse from the pressure transmitter used for the outer valve interlocks. Discuss the nature of the diversity of the pressure transmitters between the inner and outer isolation valves, and how this meets the diversity requirement of BTP RSB 5-1.

b. BTP RSB 5-1 states that the fluid discharged through the pressure relief valves must be collected and contained such that a stuck open relief valve will not result in flooding of any safety-related equipment. Has this been considered in the design of the AP600?

c. Item E of BTP RSB 5-1 states that the isolation valve operability and interlock circuits should be designed so as to permit on-line testing when operating in the RHR mode. Section 5.4.7.6 of the SSAR states that the testing requirements of the RNS pumps and the remaining MOVs are only those required to provide reliability consistent with the values assumed in the AP600 PRA. Does the RNS design meet the guidance of Item E of BTP RSB 5-1?

d. Will boron mixing tests and natural circulation cooldown tests be performed in the first plant with AP600 design per BTP RSB 5-1?

Response:

a. Typically, diversity between the pressure transmitters that control the inner and outer suction isolation valves is achieved by procuring wide range pressure transmitters either with similar pressure measurement principles from different vendors or with different pressure measurement principles (from either the same or different vendor). Specific instrumentation selection is a post-design certification activity which occurs during the equipment procurement phase.

b. Yes. The discharge of the normal residual heat removal pump suction relief valve is routed to the IRWST. The discharge of the normal residual heat removal pump discharge relief valve is routed to the liquid waste system effluent holdup tank.

c. Yes. The pressure interlock signals and logic are tested on line to the maximum extent possible without adversely affecting safety. This test includes the initiating signals for the interlocks from the division logic cabinets. For further information see SSAR Section 7.6.1.1.2.

d. SSAR Chapter 14 and the AP600 ITAAC describe the tests to be performed for the AP600 prior to startup. These tests adequately address the concerns of BTP RSB 5-1 that could be applicable to the AP600. Tests to be performed include natural circulation tests with heat removal from the steam generators and natural circulation with heat removal from the passsive residual heat removal heat exchangers. Tests are also planned to demonstrate operation of the core makeup tanks under natural circulation conditions. No tests are explicitly planned to verify boron mixing. As described in reference 440.138-1, the results of the natural circulation boron mixing and cooldown test performed at Diablo Canyon conclusively proved that boron mixing during natural circulation is not a concern. As the AP600 has a simplified loop design with no crossover leg, the possibility of "boron hideout" is reduced compared to a conventional design such as Diablo Canyon.

References:

440.138-1 WCAP 11086, Diablo Canyon Units 1 and 2 Natural Circulation / Boron Mixing / Cooldown Test Final Post Test Report, dated March 1986.

SSAR Revision: NONE

Question 440.140

Section 5.4.7.5 of the SSAR states that the compliance of the RNS design with General Design Criteria 2, 4, 5, 19 and 34 is found in Section 3.1. However, the statement of compliance with these GDCs is not clearly described in Section 3.1. For example, the statement of compliance indicates that the safety-related structures, systems, and components (SSCs) are designed to meet GDC 2 and that the SSCs important to safety will be designed to withstand the effe ts of natural phenomena without the loss of the capability to perform their safety functions. It further states that those SSCs vital to the shutdown capability of the reactor are designed to withstand the maximum probable natural phenomena at the intended site. Because the RNS is categorized to be a non-safety-related system that provides defense-in-depth functions, it is not clear whether it complies with GDC 2. Provide a specific description of GDC compliance for the RNS, and provide the bases for any noncompliance.

Response:

The conformance of the passive residual heat exchanger and the normal residual heat removal systems with General Design Criteria (GDC) 2, 4, 5, 19, and 34 is presented in WCAP-13054, Rev. 1--"AP600 Compliance with SRP Acceptance Criteria." Further clarification is provided below:

Safety-related decay heat removal is accomplished by the passive residual heat removal heat exchanger, which is designed in compliance with GDC 2, 4, 5, 19, and 34.

The decay heat removal function of the normal residual heat removal system (RNS) is not safety-related. However, the normal residual heat removal system maintains the selected safety-related functions of reactor coolant pressure boundary isolation and containment isolation. The design of the normal residual heat removal system components that provide these particular safety-related functions, comply with GDC 2, 4, 5, and 19.

The normal residual heat removal system is designed to be operated from the main control room, and RNS equipment redundancy is provided to achieve reliable nonsafety-related decay heat removal capabilities. The normal residual heat removal system is designed for a single nuclear power unit and is not shared between units. The portion of the normal residual heat removal system located outside containment is classified as AP600 Equipment Class C, Seismic Category I, not to protect its capabilities to remove decay heat, but to protect the integrity of the pressure boundary. The design of the normal residual heat removal system components utilized for nonsafety-related decay heat removal normal residual heat removal system pumps and normal residual heat removal system heat exchangers) is in compliance with GDC 4, 5, and 19 and partial compliance with GDC 2 and 34, even though compliance is not require since safety-related decay heat removal is provided by the passive residual heat removal heat exchangers.

SSAR Revision: NONE

Question 440.141

Section 5.4.12.1 of the SSAR states that the reactor vessel head vent system (RVHVS) is designed to remove noncondensable gases or steam from the RCS via remote manual operations from the control room (CR) through a pair of solenoid-operated isolation valves, and that a parallel path of manual valves is provided to perform similar functions during normal filling and venting operations.

a. Identify any operational differences between the solenoid-operated valves and the manual valves.

b. Do these valves have individual positive valve position indication and alarm in the CR?

c. Section 5.4.12.2 of the SSAR states that the RVHVS valves are included in the operability program. Does this mean that the vent system has provision to test for operability?

d. The solenoid-operated valves are powered by the diverse vital power supplies. Discuss the diverse power source and its compliance with the guidance of Section 5.4.12 of the SRP that they be powered from emergency buses.

Response:

a. As described in the June 30, 1994 AP600 Change Report (NTD-NRC-94-4175), the AP600 head vent has been modified so that both trains of valves are identical remotely-operated solenoid valves. There are no operational differences between these trains of valves.

b. The remotely-operated head vent valves have individual positive valve position indication and alarm in the control room.

c. These valves are included in the AP600 operability program and their inservise testing requirements are provided in the response to RAI 210.24.

d. The solenoid-operated valves are powered by the safety-related Class 1E DC and UPS System. This safety-related system is described in SSAR Subsection 8.3.2. The SSAR will be modified as shown below.

SSAR Revision:

The first two paragraphs of Subsection 5.4.12.2 will be revised as follows:

The reactor vessel head vent system consists of two flow paths, each with redundant isolation valves. Table 5.4-18 lists the equipment design parameters. The reactor vessel head vent system is shown on the reactor coolant system piping and instrumentation diagram (Figure 5.1-5).

~~The active portion of the system consists of two 1-inch,~~ The head vent system consists of two parallel paths of two 1-inch, open/close, solenoid-operated isolation valves ~~and two manual isolation valves~~ connected to a 1-inch vent pipe located near the center of the reactor vessel head. The system design with two valves in series in each

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flow path minimizes the possibility of reactor coolant pressure boundary leakage. The solenoid-operated isolation valves are powered by the ~~diverse vital power supplies~~ safety-related Class 1E DC and UPS Systems. The solenoid-operated isolation valves are fail-closed, normally closed valves. The valves are included in the valve operability program and are qualified to IEEE-323, IEEE-344, and IEEE-382.

SSAR Figure 5.1-5 RCS Piping and Instrumentation Diagram will be revised to reflect this design change.

Question 440.143

Section 5.4.12 of the SRP states that the size of the vent line should be kept smaller than the size corresponding to the definition of a LOCA to avoid unnecessary challenges to the ECCS. Does the one-inch vent pipe in the AP600 design meet this criterion?

Response:

The reactor vessel head vent contains orifices in the valve discharge lines that restrict the flow rate of the head vent to within the capabilities of the normal makeup system. Although the one-inch vent pipe is larger than the size of the size corresponding to the definition of a LOCA, the use of the orifices allow the AP600 to meet the intent of this criterion. See the response to RAI 440.141 for a related discussion.

SSAR Revision:

See the response to RAI 440.141.

Question 440.144

Describe the procedures used for venting and removing gases from the steam generator U-tubes to assure coolability of the core.

Response:

During plant startup operations, the reactor coolant system is filled water solid, with the high point vents opened to allow air to be vented from the system. After the vents are closed, a reactor coolant pump in each steam generator is started and allowed to run for a short time, and is then stopped. The high point vent lines are then reopened, to allow any air that collects in the high points to be vented. The vents are then reclosed, and the venting procedure is repeated until all of the air is removed from the reactor coolant system.

The AP600 does not require the steam generator U-tubes to be vented to provide coolability of the core during accident operations. The passive safety-related systems provide the safety-related function of core cooling following an accident.

SSAR Revision: NONE

Question 460.17

Discuss the applicability of IE Bulletin 80-05 to the AP600 design. The subject bulletin addresses the release of radioactive material or other adverse effects as a result of low-vacuum conditions that cause tank buckling. The low vacuum condition is created by cooling hot water in a low pressure tank. Design approaches that resolve this issue involve sizing the applicable tank vents adequately to prevent the collapse of the tank during drain down, or providing vacuum breakers to the tanks, as necessary. Section 11.2 of the SSAR should address how this concern will be resolved.

Response:

The liquid radwaste system has only one tank which can be exposed to hot water. This tank is the reactor coolant drain tank (RCDT) in the containment building. The reactor coolant drain tank can receive flashing primary coolant during loop drain operations. The reactor coolant drain tank has several features which eliminate the possibility of structural collapse due to condensation of steam. As shown in SSAR Table 11.2-2, the tank specification includes an external design pressure of 15 psig. Should the tank be exposed to a full vacuum, it will not collapse.

SSAR Revision: NONE

Question 460.24

The dilution flow for the liquid waste discharge appears to be very low. Reevaluate the dilution flow for diluting liquid radwaste discharge concentrations and revise, as appropriate. Justify your response.

Response:

The review of the dilution factor revealed an error in transcription of technical information. The required dilution factor is 270, not 400. The values in the table of SSAR section 11.2.3.3 are correct.

For example, to discharge 2150 gallons over an 8 hour period using a dilution factor of 270 requires:

$$\frac{2150 \text{ gallons}}{8 \text{ hours}} \times \frac{1 \text{ hour}}{60 \text{ min}} \times 270 = 1200 \text{ GPM of dilution flow}$$

SSAR Revision:

Revise the second paragraph of Subsection 11.2.3.3 as follows:

The dilution factor required to meet the 10 CFR 20 maximum permissible concentrations is 270 ~~400~~. The required dilution flow is dependent on the liquid waste discharge rate and, while the monitor tank pumps have a design flow rate of 100 gpm, the discharge flow is controlled to be compatible with the available dilution flow. With the average liquid waste release of 2150 gallons per day, the required dilution flow is:

Question 460.26

Table 11.1-8 of the SSAR should have an entry for N-16 primary coolant activity. Revise the table accordingly.

Response:

Table 11.1-8 of Revision 0 of the SSAR had a listing of N-16 activity. The listing of N-16 activity was removed as part of the response to RAI 460.8. Table 11.1-8 has been revised to show coolant activity concentrations calculated by the GALE computer code. The GALE code does not consider N-16 because it is not a consideration in regard to effluent releases.

N-16 is a major contributor to the shielding source term for the reactor coolant system. Table 12.2-3 provides N-16 concentrations at various points in the reactor coolant system.

SSAR Revision:

Revise the first paragraph of Subsection 11.1.1.4 as follows:

Activation of oxygen in the coolant results in the formation of N-16 which is a strong gamma emitter. Because of its short half-life of 7.11 seconds, N-16 is not of concern outside the containment. **Table 12.2-3 provides N-16 concentrations at various points in the reactor coolant system.**

Question 480.78

The staff is trying to determine whether the containment design of the AP600 can meet Service Level C limits for the first 24 hours after the onset of a core melt accident. For the AP600 containment, what are the pairs of (maximum pressure, temperature) and (pressure, maximum temperature) under severe accident conditions?

Response:

The staff's containment performance goal is outlined in SECY 93-087:

> "The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident containment challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products."

As shown in revision 1 of the AP600 Probabilistic Risk Assessment, the AP600 has a low frequency of core damage and containment design features which address severe accident challenges (e.g. reactor coolant system depressurization, hydrogen igniters, reactor cavity flooding, passive containment cooling, reactor cavity design). The severe accident sequences which fall into the category of "more likely severe accident containment challenges" do not produce the high energy severe accident events which would cause the containment pressure to approach ASME Service Level C. High energy containment challenges which could cause the containment pressure to approach Service Level C appear in accident sequences with frequencies less than the AP600 cut-off frequency of 1.0×10^{-7} per reactor-year for Service Level C. The frequencies of the severe accident containment challenges are presented in revision 1 of the AP600 Probabilistic Risk Assessment Report, chapter 12, section 12.5.

The following discussion presents the pressure and temperature transients for the top two severe accident containment challenges due to high energy phenomena. The maximum pressure, temperature and maximum temperature, pressure pairs are not sufficient to fully describe the effect of a severe accident on the containment shell temperature response. The transient pressure and temperature profiles of the containment gas are required since the duration at which the wall is exposed to elevated gas temperatures significantly affects the wall temperature response due to the "thermal inertia" of the wall.

The following two MAAP4 cases will be submitted with revision 1 of the AP600 Probabilistic Risk Assessment Report. They represent the worst severe accident cases in terms of containment pressure and temperature response. Both cases are based on the accident class 3BE in which failure of both trains of gravity injection lines from the in-containment refueling water storage tank to the reactor vessel downcomer are failed. The first case, 3BE.h3, models the global combustion of hydrogen generated from the oxidation of approximately one hundred percent of the active cladding in the reactor core. The second case, 3BE.cc4, models a severe accident case with the failure of the passive containment cooling system water which cools the outside the containment shell and failure of the operator to flood the reactor cavity which results in reactor vessel failure. The transient temperature response of five MAAP4 heat sinks are provided for each case. These heat sinks represent the wet and dry portions of the

containment dome and shell which are cooled by the passive containment cooling system and the portion of the containment shell below the operating deck which is not cooled.

In case 3BE.h3, the hydrogen burn produces severe peak pressures and containment gas temperatures (Figures 480.78-1 and 480.78-2) of 68.2 psia and 2060°F (4.7 bar and 2154°K), but the duration of the transient is short. The peak wall temperatures (Figure 480.78-3) are limited to less than 225°F (380°K).

In case 3BE.cc4, the failure of the passive containment cooling water limits the heat removal from the containment shell to natural convection to the passive cooling annulus air. The pressure and gas temperature in the containment (Figures 480.78-4 and 480.78-5) peak at 45 psia and 287°F (3.1 bar and 415°K) prior to reactor vessel failure. When the reactor vessel fails, the containment pressure and gas temperature peak at 78.3 psia and 310°F (5.4 bar and 427°K), and the pressure and temperature are sustained over a long term due to steaming from the reactor cavity. The maximum containment wall temperature after vessel failure is 295°F (419°K).

SSAR/PRA Revision: NONE

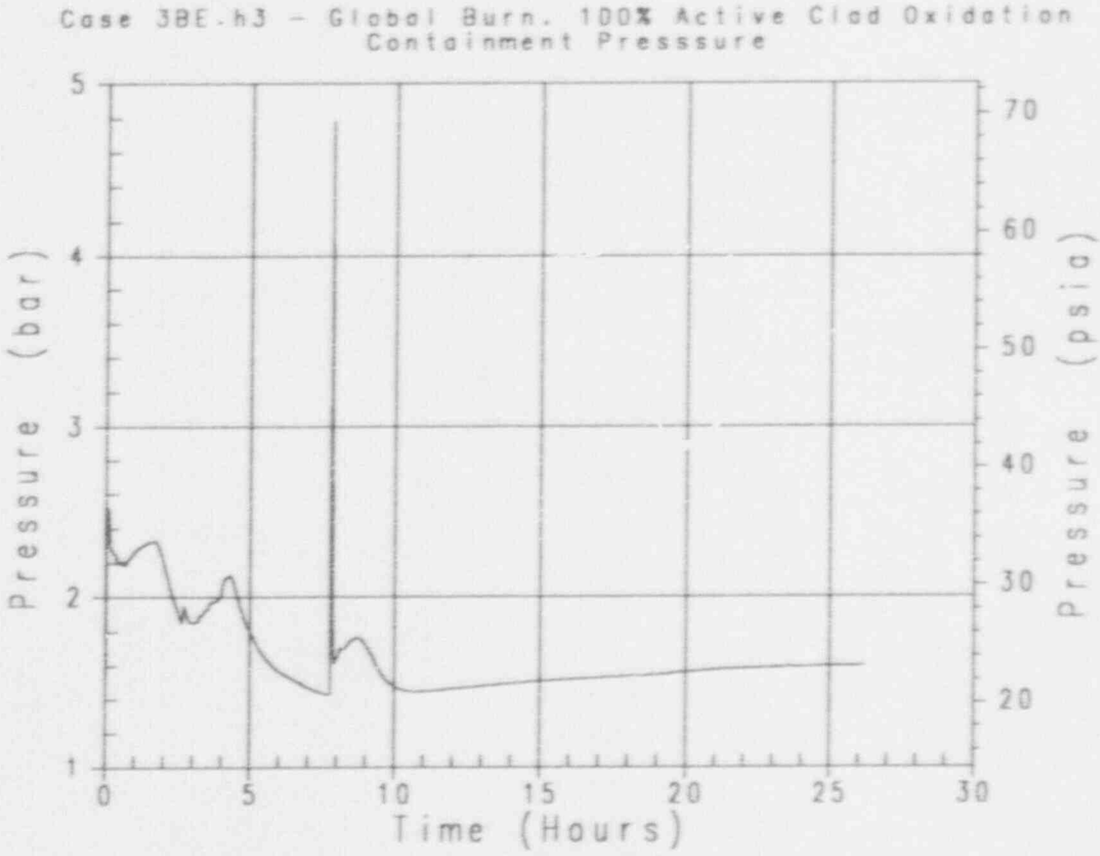

Figure 480.78-1

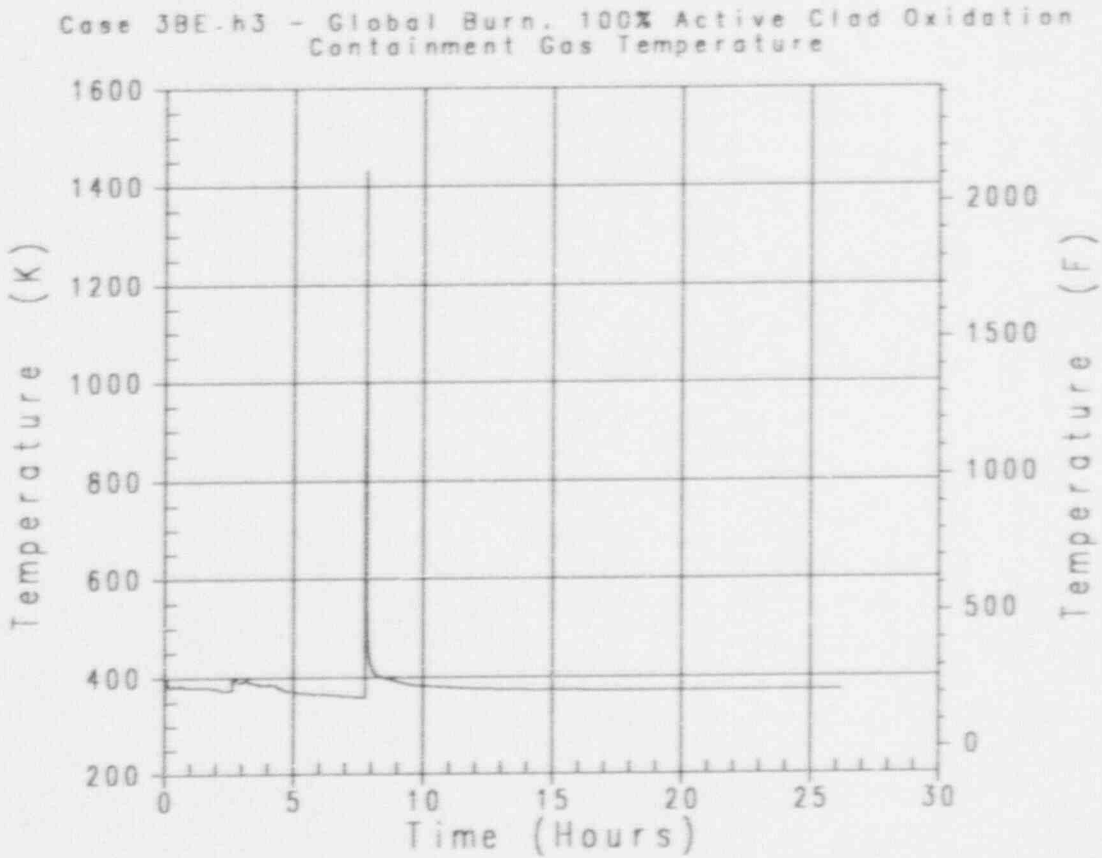

Figure 480.78-2

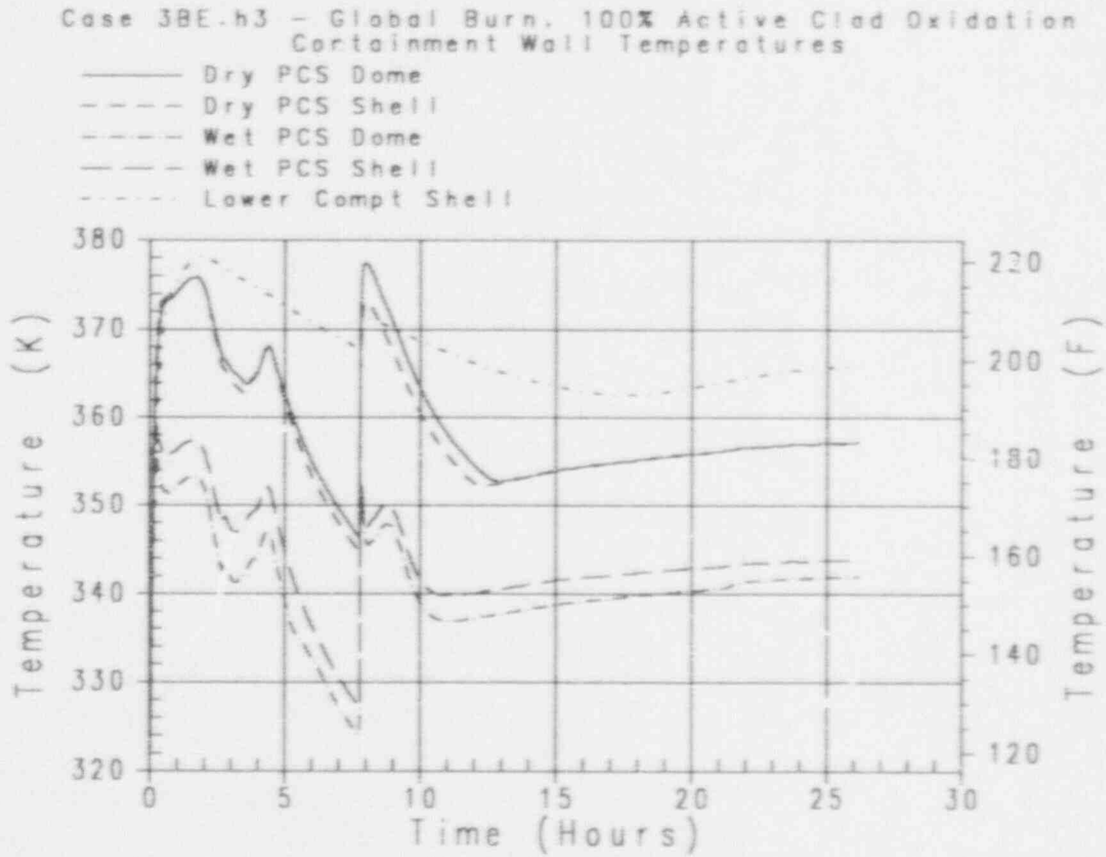

Figure 480.78-3

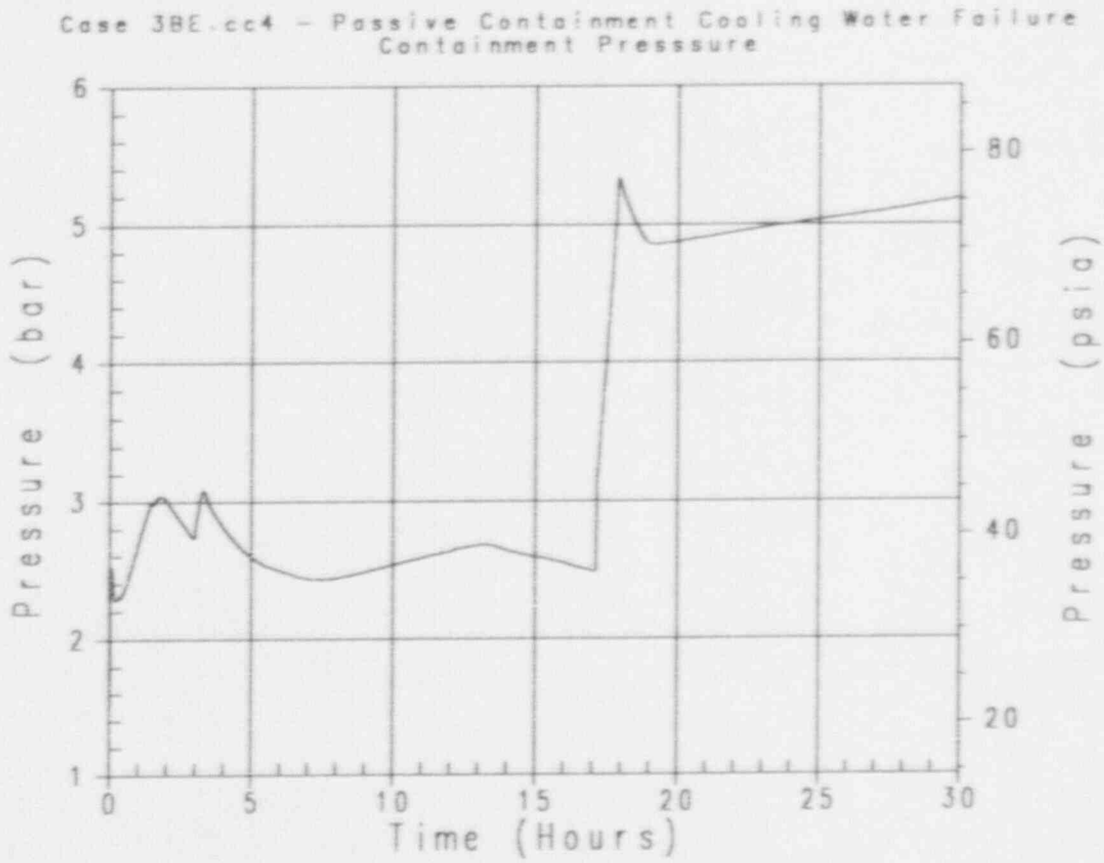

Figure 480.78-4

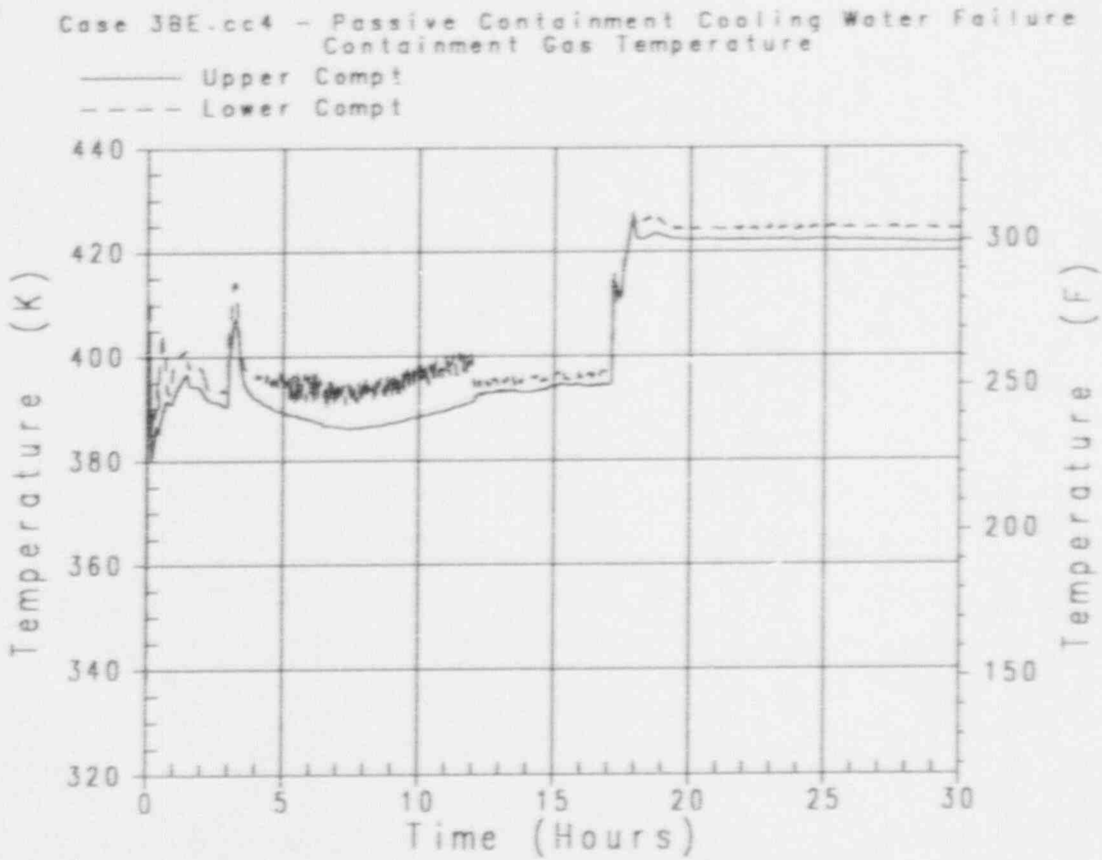

Figure 480.78-5

AP600

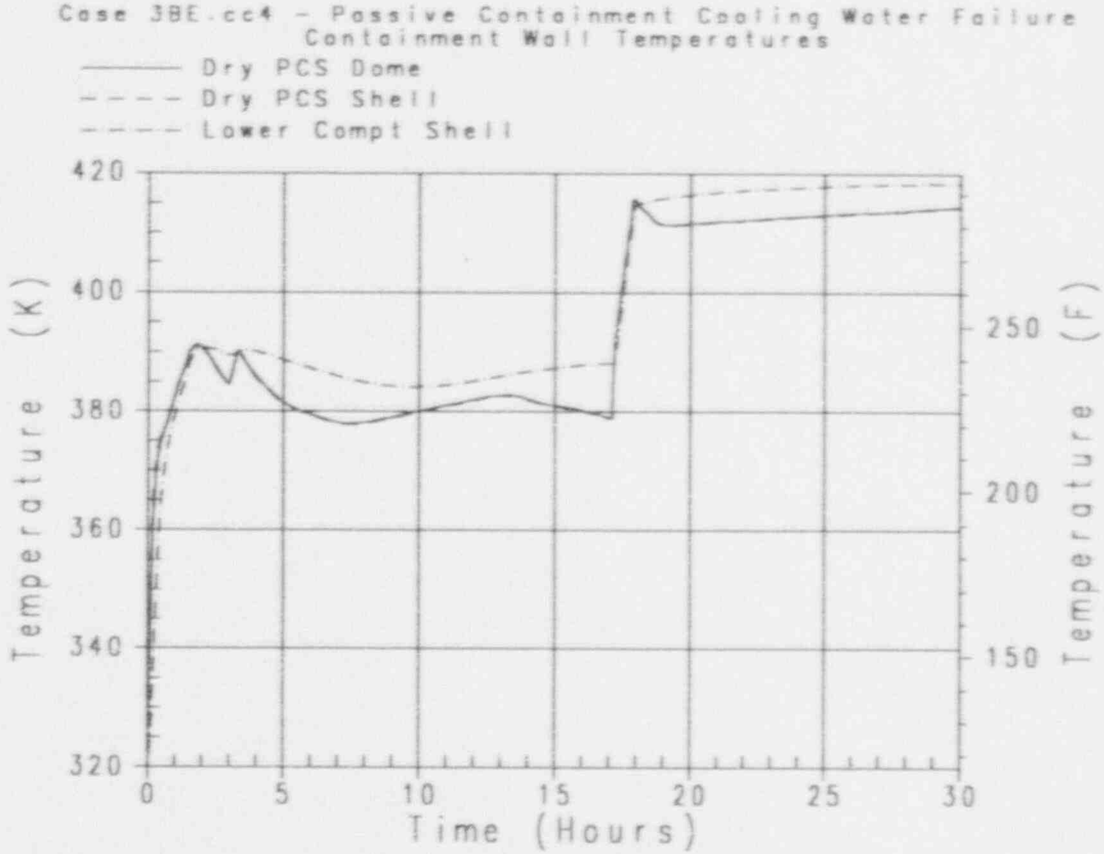

Figure 480.78-6

Question 490.1

Section 4.1 of the SSAR states that the AP600 fuel is **similar** to the 17x17 VANTAGE-5H fuel assemblies. Does "**similar**" mean that the AP600 core will be analyzed as if it were a full core VANTAGE-5H? What are the bases for the validity of the VANTAGE-5H analysis for AP600 applications? (Sections 4.1 and 4.2)

Response:

The AP600 fuel assembly is a version of the VANTAGE-5H fuel assembly with intermediate flow mixing grids. The word "similar" is used to emphasize the AP600 mechanical design features common with the VANTAGE-5H fuel assembly. The word "similar" is not intended to mean that the VANTAGE-5H results are directly applicable to AP600. Analyses and evaluations are performed to qualify each AP600 fuel assembly component to assure that the design criteria are satisfied. Because most AP600 components are the same as their counterparts on the VANTAGE-5H design, for example, low pressure drop intermediate grids, the results from existing VANTAGE-5H evaluations may be used for AP600. AP600 design basis loads are evaluated to verify that the VANTAGE-5H design basis loads are applicable and enveloping (or bounding) for the AP600 application.

SSAR Revision: NONE

Question 490.3

The various Westinghouse fuel designs currently utilized in operating plants use approved calculational procedures. Were these same calculational procedures applied to the AP600 fuel design, or were new calculational procedures developed for this purpose? If so, were these new procedures approved by the NRC? Provide references for these approvals, if any.

Response:

Only NRC approved procedures and methodology are used to assess the AP600 fuel design. The use of gray rods and the MSHIM power distribution control strategy (see the response to RAI 491.7) does not require changes to approved procedures and methodology. Also the use of fixed incore detectors in the incore instrumentation system (see the response to RAI 492.5) does not require changes to approved procedures and methodology. The use of the W-3 correlation for departure from nucleate boiling in the 300 to 500 psia range (see the response to RAI 492.4) represents an extension of previously approved procedures and methodology.

SSAR Revision: NONE

Question 490.2

Section 4.2.2.1 of the SSAR states that the AP600 fuel rod may include an axial fuel blanket, an integral fuel burnable absorber, and Zircaloy-4 or ZIRLO cladding.

Under what condition(s) will these be used (incorporated) into the fuel management program? Also, does the ultimate decision to do so rest with Westinghouse, the combined operating licensee, or both? Are the licensing analyses based on inclusion of these features or not? If not, how do you justify the application of one analysis to support the actual design not analyzed?

Response:

Since the AP600 has the capability of achieving up to and including a 24 month refueling cycle, integral fuel burnable absorbers, wet annular burnable absorbers (refer to SSAR Section 4.2.2.3.3), or a combination of these may be employed to provide that the moderator temperature coefficient and power distribution requirements are met. Axial blankets could be utilized in the designs should the combined license applicant choose the economic benefits associated with reduced axial neutron leakage. The AP600 fuel mechanical design has the capability of assembly-average burnups of at least 60,000 MWD/MTU. ZIRLO cladding is an ideal candidate which can be used to support extended fuel burnup applications such as these.

The ultimate decision to incorporate specific fuel and fuel assembly features resides with the combined license applicant.

The licensing analyses for the AP600 are not based on inclusion of the design features identified above. Each of these features has been incorporated into operating Westinghouse plant designs. Impacts (from the employment of these design features) with regard to core design characteristics, operational characteristics, and safety analyses, are well known. Inclusion of any of the above design features individually or in combination into the AP600 could be included in the AP600 fuel design.

SSAR Revision: NONE

Question 490.5

Vibrational problems have recently been associated with VANTAGE-5H fuel at existing Westinghouse plants. Provide a detailed design and testing program indicating actions taken by Westinghouse to ensure that these problems will not affect the AP600 design proposed VANTAGE-5H fuel and future AP600 mixed core designs.

Response:

Westinghouse has been engaged in a testing program of VANTAGE-5H fuel designs. This program has determined a design change which essentially eliminates excitation of fuel assembly natural frequencies. This change will be incorporated in the AP600 fuel assembly design.

SSAR Revision: NONE

Question 491.7

Section 4.3.2.4.13 of the SSAR does not indicate whether an approved detailed review of the use of gray rods has ever been made. Is there such a review currently waiting approval for utilizing gray rods in the AP600 or in any of the existing Westinghouse plants?

Response:

No application of a core design utilizing gray rods is waiting NRC approval. Gray rods are low-worth control rods use for reactivity and power distribution control. Gray rods are employed in the AP600 to provide a load follow capability with no boron change. Gray rods are comprised of approved materials currently in use in operating reactors and are functionally similar to conventional rod cluster control assembly designs.

Gray rods are utilized in conjunction with the MSHIM (Mechanical SHIM) power distribution control strategy. A topical report detailing gray rod operation in conjunction with MSHIM is scheduled and will be submitted to the NRC for approval in support of the final safety evaluation report.

SSAR Revision: NONE

Question 492.2

Section 4.4.1.3 of the SSAR provides the value of 91.0 percent as the minimum flow rate passing though the fuel region of the AP600 core. The remaining 9.0 percent is allotted to bypass flow. How do these values compare to other Westinghouse designs? Provide justification for the 9% bypass flow used in the AP600 design.

Response:

The value of 91 percent cited in Subsection 4.4.1.3.1 is a minimum design value used for fuel thermal analysis and is lower than the best estimate value. The bypass flow includes flow from the thimbles in the core, the reflector cooling flow, outlet nozzles, reflector to fuel assembly gap and head cooling. The bypass flow for the AP600 is somewhat higher than for typical Westinghouse designs. This is primarily due to the absence of thimble plugs and the presence of cooling flow for the reflector. The bypass flow was calculated based on the AP600 fuel assembly and reactor internals designs.

SSAR Revision: NONE

Question 952.74

Is the leakage power of 1.3% from the heater rod power cable which runs through the lower plenum of the SPES-2 facility a portion of the total power of 4.9 MW or additive to the total power?

Response:

The response to the RAI is provided in the response to RAI 952.32.

SSAR Revision: NONE

PRA Revision: NONE