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Energy Systems

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Docket No.: STN-52-003

September 19, 1994

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

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of April 29, 1994, May 23, 1994, May 24, 1994, June 15, 1994, and August 15, 1994. This completes the responses associated with the April 29, May 24, June 15 and August 15 letters. In addition, revisions of responses previously submitted are provided. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A. Attachment B is a listing of the questions associated with your letters of April 29, 1994, May 24, 1994, June 15, 1994, and August 15, 1994 and the date of the Westinghouse letters that transmitted the responses.

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

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

Brian A. McIntyre / for

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

/nja

Enclosure

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

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NTD-NRC-94-4305
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED SEPTEMBER 19, 1994

RAI No.	Issue
220.063R01	Air baffle structural design
260.030	System/component not included in test descriptions
260.031	System/component not included in test descriptions
410.206	Hydrogen concentration, safety related SSCs
440.012R01	ADS testing
480.036R01	Diffusion flames above IRWST
480.077	Isolation valves >10 ft from cont wall
952.090	PRHR test data
952.092	OSU Test Facility Information

ATTACHMENT B
CROSS REFERENCE OF WESTINGHOUSE RAI RESPONSE TRANSMITTALS TO NRC LETTERS OF APRIL 29, 1994, MAY 24, 1994, JUNE 15, 1994, AUGUST 15, 1994

Question No.	Issue	NRC Letter	Westinghouse Transmittal Date
220.092	Containment structural calculations	06/16/94	07/27/94
220.093	Containment severe accident loading	06/16/94	08/03/94
260.023	Startup and/or preoperational testing	05/24/94	07/08/94
260.024	Scoping document for startup & pre-op testing	05/24/94	07/22/94
260.025	Startup administrative manual	05/24/94	07/22/94
260.026	Pre-op tests for first plant only	05/24/94	08/08/94
260.027	Test program schedule as COL item	05/24/94	07/22/94
260.028	Individual test descriptions	05/24/94	07/22/94
260.029	Basis for determining acceptable performance	05/24/94	07/22/94
260.030	System/component not included in test descriptions	05/24/94	09/19/94
260.031	System/component not included in test descriptions	05/24/94	09/19/94
260.032	SSAR Section 14.2.9	05/24/94	07/22/94
480.049	Provisions for Type C testing, Table 6.2.3-1	04/29/94	07/25/94
480.050	Type C testing of service air	04/29/94	07/25/94
480.051	Component cooling system isolation signals	04/29/94	06/16/94
480.052	SSAR Table 6.2.3-1 & Figure 9.2.4-1	04/29/94	07/25/94
480.053	Cont pressure instrument line penetration RG 1.11	04/29/94	07/08/94
480.054	Type C testing of RHR suction isolation valves	04/29/94	07/08/94
480.055	Inidentified NRHR penetrations	04/29/94	07/29/94
480.056	Relief valves as containment isolation barriers	04/29/94	07/25/94
480.057	LTC signal	04/29/94	07/01/94
480.058	Airlock seal testing as reduced pressure	04/29/94	07/27/94
480.059	Method of testing spare penetrations	04/29/94	07/01/94
480.060	manual vs remote manual	04/29/94	07/25/94
480.061	Chilled water return isolation valve size	04/29/94	07/27/94
480.062	Steam generator isolation valve closure time	04/29/94	06/27/94
480.063	Nonsafety power supply for hydrogen recombiners	04/29/94	07/01/94
480.064	Rate of hydrogen generation due to radiolysis	04/29/94	08/03/94
480.065	Potential for drain clogging from coatings	04/29/94	07/08/94
480.066	Margin between max calculated & design cont press	04/29/94	07/25/94
480.067	HT coefficient sensitivity to node size near wall	04/29/94	07/22/94
480.068	Postulated break size for subcompartment analyses	04/29/94	07/25/94
480.069	Use of TMD code for M&E releases	04/29/94	07/25/94
480.070	Containment pressure analyses for ECCS performance	04/29/94	07/27/94
480.071	Testing of containment heat transfer	04/29/94	07/01/94
480.072	Credit taken for secondary containment during DBA	04/29/94	06/27/94
480.073	Closure time for containment isolation valves	04/29/94	07/08/94
480.074	Recombiner power supply; post-LOCA cont purging	04/29/94	07/22/94
480.075	Containment leakage testing	04/29/94	07/15/94
480.076	Containment penetrations beyond "state of art"	04/29/94	07/29/94
480.077	Isolation valves >10 ft from cont wall	04/29/94	09/19/94
480.078	Max cont. P,T for severe accident conditions	04/29/94	06/30/94
480.079	Fuel-coolant interaction parameters	06/16/94	09/02/94
952.090	PRHR test data	06/16/94	09/19/94
952.091	CMT Test Facility Drawings	08/15/94	08/26/94
952.092	OSU Test Facility Information	08/15/94	09/19/94

Records printed: 46

Question 220.63

For the air baffle,

a. pertaining to the fatigue aspects of the containment shell design, provide information on the magnitude, distribution and number of cycles of the stresses induced by the wind,

b. consider the potential of tornado missiles generated by the air baffle and discuss whether or not the air deflector is protected against tornado missiles, and

c. provide detailed information of the flexible seal between the air baffle and the shield building roof.

Response: (Revision 1)

a. SSAR Section 3.3 describes the design wind conditions and resulting loads for the AP600. Wind tunnel tests are reported in WCAP-13323-~~P~~, **WCAP-14068 and WCAP-14169** (References 220.63-1, **2 and 3**) . The data from these tests was used to determine wind pressure loads on the containment vessel and air baffle for the design wind and the tornado.

- Wind conditions result in a pressure reduction in the annulus between the shield building and the containment vessel as well as above the containment dome. The maximum reduction is **0.87** ~~0.65~~ psi for the 110 mph design wind. This reduced pressure is equivalent to an increase in containment internal pressure, and is within the normal operating range for containment pressure (-0.2 to 1.0 psig). Stresses resulting from this pressure are small and will not contribute to fatigue.

- Wind conditions result in a small wind load across the containment vessel. This is maximum opposite the air intakes where positive pressures occur on the windward side and negative pressures occur on the leeward side. **Lateral loads on the containment vessel are developed in Reference 220.63-3. This reference uses the results of the Phase IVA tests reported in Reference 220.63-2 and calculates the resultant lateral loads on the vessel for each level of taps. Figures 4 - 7 of Reference 220.63-3 show the distribution of the pressure around the circumference at the instant in time corresponding to the maximum lateral load. These figures show that the pressure is fairly uniform around the circumference and that the differential loads on the vessel are small. However, for completeness the loads are included in the containment vessel design specification.** Resulting stresses are small and do not contribute to fatigue.

b. As described in SSAR Subsection 3.8.4.1.3, the air baffle is designed for the wind and pressure loads from the tornado and hence it will not fail and generate missiles. The air baffle is protected from tornado missiles by the shield building. The upper portion of the air baffle (designated as the air deflector in the RAI) may be subjected to missile impact by missiles that could pass through the air inlets. This portion of the air baffle is constructed from one quarter inch thick plate, which would stop small missiles but would experience local damage from the large tornado missiles. Such damage would not prevent function of the air baffle.

c. Information on the flexible seal was provided in the response to RAI 220.28

References:

220.63-1. WCAP-13323-~~P~~, Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor, June, 1992
220.63-2. WCAP-14068, Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, May, 1994
220.63-3. WCAP-14169, Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report, September, 1994

SSAR Revision: NONE

Question 260.30

The preoperational and startup test phase descriptions in Section 14.2.8, "Individual Test Descriptions," of the SSAR do not provide assurance that the operability of several of the systems and components listed in Appendix A of Regulatory Guide 1.68 (Revision 2) will be demonstrated. The test abstracts of Section 14.2.8 should be expanded to address the following items identified in Appendix A to RG 1.68, or Section 1A of the SSAR should be revised to provide technical justification for any exceptions taken.

RG Paragraph	System/Component
1.	Preoperational Testing
1.a.(2)(i)	Pressurizer safety valves.
1.b.(1)	Control rod withdrawal inhibit and rod runback functions.
1.c	Diverse actuation system that provides protection of facility for anticipated transients without a scram (ATWS).
1.e.(4)	Steam generator pressure safety valves.
1.e.(10)	Feedwater heater and drains.
1.f.(2)	Cooling towers and associated auxiliaries.
1.j.(7)	Leak detection systems used to detect failures in ECCS and containment recirculation systems located outside containment. For example, potential leakage in normal RHR system or the post accident sampling systems that could be used to recirculate reactor coolant outside containment after an accident.
1.j.(8)	Automatic reactor power control system and primary T-average control system.
1.j.(13)	Excore neutron instrumentation.
1.j.(17)	Feedwater heater temperature, level, and bypass controls.
1.j.(20)	Instrumentation used to detect external and internal flooding conditions.
1.j.(22)	Instrumentation used to track the course of postulated accidents such as: containment wide-range pressure indicators, reactor vessel water level monitors, containment sump level monitors, high radiation detectors, and humidity monitors.

1.j.(23)	Post-accident hydrogen monitors.
1.j.(24)	Annunciators for reactor control and engineered safety features.
1.k.(2)	Personnel monitors and radiation survey instruments. As the calibration program applied to these devices will be site specific, it would be appropriate to identify this as a COL action item.
1.k.(3)	Laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.
1.l.(5)	Isolation features for condenser offgas systems.
1.m.(4)	Static load testing at 125-percent rated load of cranes, hoists, and associated lifting and rigging equipment.
1.n.(5)	Secondary sampling systems.
1.n.(9)	Drain systems and pumping systems serving essential areas.
1.n.(12)	Boron recovery system.
1.n.(13)	Communications systems relating to offsite emergency notification.
1.n.(14)(c)	Class 1E electrical room heating, ventilating, and air conditioning.
1.n.(14)(f)	Main Control Room: Proper operation of smoke and toxic chemical detection systems and ventilation shutdown devices, including leaktightness of ducts.
1.n.(15)	Shield cooling systems.
1.o.(1)	Dynamic and static load tests of reactor components handling system cranes, hoists, and associated lifting and rigging equipment.
1.o.(2)	Protective devices and interlocks of reactor components handling system equipment.
1.o.(3)	Safety devices for reactor components handling systems equipment.
2.	<u>Initial Fuel Loading and Precritical Tests</u>
2.f	Reactor core and other major components differential pressure and vibration testing after fuel loading.
4.	<u>Low Power Testing</u>

4.c Pseudo rod ejection test.

4.i Control rod block and inhibit functions.

5. Power Ascension Tests

5.e Pseudo rod ejection test.

5.m Reactor core and major reactor coolant system components differential pressure.

5.r Process computer and control room computer.

5.t Pressurizer safety valves and secondary system safety valves.

5.c.c Include a test description for power ascension tests to demonstrate that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design.

5.g.g Design features to prevent or mitigate anticipated transients without scram (ATWS).

5.k.k Dynamic response of the plant for loss of feedwater heaters or bypassing feedwater heaters.

Response:

The specific response to each paragraph question follows:

RG Paragraph **System/Component**

1. Preoperational Testing

1.a.(2)(i) Pressurizer safety valves.

The test is specified in Section 3.1.2 of the AP600 ITAAC. This testing will be conducted as required by ASME Code Section III, subsection NB.

No test abstract needs to be added or amended.

1.b.(1) Control rod withdrawal inhibit and rod runback functions.

The control rod withdrawal inhibit and runback functions are tested as part of reactor control, rod control and digital RPI Abstract 14.2.8.1.63. This test requires inputs from other systems such as the NIS. Objective identified in the Abstract "Demonstrate Operation of the Rod Control System in Response to Interlock Signals" provides for the testing of these functions.

No test abstract needs to be added or amended.

1.c Diverse actuation system that provides protection of facility for anticipated transients without scram (ATWS).

Diverse actuation system testing is provided in the Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC) Section 3.5.1, Diverse Actuation System.

No test abstract needs to be added or amended.

1.e.(4) Steam generator pressure safety valves.

Steam generator pressure safety valves are identified in ITAAC, Section 3.2.5 under safety related functions. Steam generator safety valves will be tested according to ASME Section III, Sub-Section NC.

No test abstract needs to be added or amended.

1.e.(10) Feedwater heater and drains.

Feedwater heaters are tested under Abstract 14.2.8.1.30, Feedwater Control System. Feedwater drains are tested as part of another series of non-safety related plant systems which are not included in the SSAR.

No test abstract needs to be added or amended.

1.f.(2) Cooling towers and associated auxiliaries.

Separate cooling towers and associated auxiliaries are provided for the circulating water system (CWS) and service water system (SWS). The circulating water system is not safety related. CWS is site specific and is the responsibility of the Combined Licensing Applicant. The cooling towers associated with the service water system provide a non-safety related shutdown decay heat removal path. The SWS is tested as part of ITAAC 3.3.9.

No test abstract needs to be added or amended.

1.j.(7) Leak detection systems used to detect failures in ECCS and containment recirculation systems located outside containment. For example, potential leakage in normal RHR system or the post accident sampling systems that could be used to recirculate reactor coolant outside containment after an accident.

The AP600 ECCS including recirculation capability is fully contained within the containment. No leak detection system is required or provided for the post accident sampling system.

No test abstract needs to be added or amended.

1.j.(8) Automatic reactor power control system and primary T-average control system.

Automatic reactor power control system and primary T average control system are tested per a program of start-up tests that include the following Abstracts.

14.2.8.2.12	Rapid Power Reduction System
14.2.8.2.24	Process Installation Alignment
14.2.8.2.37	Power Ascension Test Sequence
14.2.8.2.42	Plant Performance
14.2.8.2.43	Thermal Power Measurement and State Point Data Collection
14.2.8.2.45	Start-Up Adjustment of Reactor Control System
14.2.8.2.46	Plant Control System
14.2.8.2.49	Load Swing Test
14.2.8.2.50	50 Percent Load Rejection
14.2.8.2.51	100 Percent Load Rejection
14.2.8.2.52	Load Follow Demonstration
14.2.8.2.54	Nuclear Steam Supply System Performance Test

14.2.8.2.55 Plant Trip From 100 Percent Power

No test abstract needs to be added or amended.

1.j.(13) Excore neutron instrumentation.

Excore neutron instrumentation is tested per Abstract 14.2.8.2.44.

No test abstract needs to be added or amended.

1.j.(17) Feedwater heater temperature, level, and bypass controls.

See above response to RAI 260.30, Paragraph 1.e.(10).

No test abstract needs to be added or amended..

1.j.(20) Instrumentation used to detect external and internal flooding conditions.

The non-safety related sump instrumentation is tested as part of the construction tests and it is not necessary to include as part of Chapter 14. The containment sump level instrumentation is tested under 1.j(22) below.

No test abstract needs to be added or amended.

1.j.(22) Instrumentation used to track the course of postulated accidents such as: containment wide-range pressure indicators, reactor vessel water level monitors, containment sump level monitors, high radiation detectors, and humidity monitors.

Instrumentation used for tracking the course of postulated accidents is tested per test Abstract 14.2.8.1.60, post-accident monitoring and sampling functions.

1.j.(23) Post-accident hydrogen monitors.

See above response to RAI Question 260.30, Paragraph 1.J.(22)

No tests abstract needs to be added or amended.

1.j.(24) Annunciators for reactor control and engineered safety features.

Annunciator testing is identified in Abstract 14.2.8.1.72, Protection and Safety Monitoring System.

No tests abstract needs to be added or amended.

1.k.(2) Personnel monitors and radiation survey instruments.

The calibration program applied to these devices will be site specific, and is not part of the standard AP600 design.

No tests abstract needs to be added or amended.

1.k.(3) Laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.

AP600 standard plant radiation effluent monitoring survey instrument testing is identified in Abstract 14.2.8.1.20, Radiation and Effluent Monitoring Systems. The calibration program applied to these devices will be site specific and is not part of the standard AP600 design.

No tests abstract needs to be added or amended.

1.l.(5) Isolation features for condenser offgas systems.

The condenser air removal system is tested under Abstract 14.2.8.1.44.

No test abstract needs to be added or amended.

1.m.(4) Static load testing at 125-percent rated load of cranes, hoists, and associated lifting and rigging equipment.

This testing is required per ASME code, which in turn references ANSL standards. Sections 3.1 and 3.3.5 of the AP600 ITAACs require this to be done for fuel handling equipment and containment polar crane.

No test abstract needs to be added or amended.

1.n.(5) Secondary sampling systems.

The secondary sampling system is tested as part of Abstract 14.2.8.2.19 "Primary and Secondary Chemistry."

No test abstract needs to be added or amended.

1.n.(9) Drain systems and pumping systems serving essential areas.

Function will be tested per abstract 14.2.8.1.37, Radioactive Waste Drain System

No tests abstract needs to be added or amended.

1.n.(12) Boron recovery system.

The AP600 does not have a boron recovery system.

No test abstract needs to be added or amended.

1.n.(13) Communications systems relating to offsite emergency notification.

This system is a Combined License Applicant responsibility.

1.n.(14)(c) Class 1E electrical room heating, ventilating, and air conditioning.

This is identified in ITAAC Section 3.7.1, Nuclear Island Non-Radioactive Ventilation System and also in Abstract 14.2.8.1.101 Nuclear Island Non-Radioactive Ventilation System.

No test abstract needs to be added or amended.

1.n.(14)(f) Main Control Room: Proper operation of smoke and toxic chemical detection systems and ventilation shutdown devices, including leaktightness of ducts.

This is identified in Abstracts 14.2.8.1.99 Main Control Room Ventilation System and 14.2.8.1.100 Main Control Room Habitability System. No toxic chemical detector system is required for the AP600 since toxic chemicals are not used.

No test abstract needs to be added or amended.

1.n.(15) Shield cooling systems.

This system is not applicable to the AP600.

No test abstract needs to be added or amended.

1.o.(1) Dynamic and static load tests of reactor components handling system cranes, hoists, and associated lifting and rigging equipment.

Tests are identified in ITAAC Table 3.1.1-1.

No test abstract needs to be added or amended.

1.o.(2) Protective devices and interlocks of reactor components handling system equipment.

Tests are identified in ITAAC Table 3.1.1-1.

No test abstracts needs to be added or amended.

1.o.(3) Safety devices for reactor components handling systems equipment.

Tests are identified in ITAAC Table 3.1.1-1.

No test abstract needs to be added or amended.

2. Initial Fuel Loading and Precritical Tests

2.f Reactor core and other major components differential pressure and vibration testing after fuel loading.

Reactor coolant system flow measurements are performed both prior to criticality and at full power. Performance criteria are that the flow is less than the mechanical design limits.

This test Abstract 14.2.8.2.13 "Reactor Coolant System Flow Measurement" will be revised to include a Maximum Allowable Flow criteria.

4. Low Power Testing

4.c Pseudo rod ejection test.

This test is not performed in the "Low Power" condition. See Item 5.e below.

4.i Control rod block and inhibit functions.

These functions are tested as part of Abstract 14.2.8.1.63 Reactor Control, Rod Control and Digital Rod Position Indication and Abstract 14.2.8.1.71 Control Rod Drive Mechanisms. Just prior to criticality, functions are again checked in Abstract 14.2.8.2.8 Rod Control System and under plant hot & cold conditions tested per 14.2.8.2.10 Control Rod Drive Mechanisms.

No test abstract needs to be added or amended.

5. Power Ascension Tests

5.e Pseudo rod ejection test.

This test is described in Abstract 14.2.8.2.47.

5.m Reactor core and major reactor coolant system components differential pressure.

See response to item above.

5.r Process computer and control room computer.

The process and control room computers must be operable prior to core loading. This is tested as described in Abstract 14.2.8.1.17. As a part of the program of statepoint data collection, comparisons of the measured parameters are made with the process computer information. Abstract 14.2.8.2.43 will be modified to reflect this. The performance criteria is that the process computer measured parameters agree with the primary measurements within the specific prescribed tolerances.

5.t Pressurizer safety valves and secondary system safety valves.

See Items 1.a.(2)i and 1.e(4) above.

5.c.c Include a test description for power ascension tests to demonstrate that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design.

The gaseous and liquid waste systems are tested in the preoperational test per Abstracts 14.2.8.1.24 and 14.2.8.1.35.

No test abstract needs to be added or amended.

5.g.g Design features to prevent or mitigate anticipated transients without scram (ATWS).

Designed feature testing is identified in ITAAC Section 3.5.1 Diverse Actuation System.

No test abstract needs to be added or amended.

5.k.k Dynamic response of the plant for loss of feedwater heaters or bypassing feedwater heaters.

A test abstract addressing the dynamic response of the plant for loss of feedwater heaters will be added to Chapter 14. The proposed test abstract is provided below.

SSAR Revision:

Revise SSAR subsection 14.2.8.2.13, "Reactor Coolant System Flow Measurement" as follows:

Performance Criteria

- The reactor coolant system flow determined from the measurements at approximately 100 percent rated thermal power equals or exceeds the minimum value required by the plant technical specifications and is less than or equal to Mechanical Design Flow. See SSAR subsection 5.1.4.4.

Add the following test abstract to SSAR Chapter 14:

14.2.8.2.x "Dynamic Response for Loss of Feedwater Heater"

Objective

- To demonstrate feedwater heater bypass and/or condensate recirculation capability due to a feedwater heater or feedwater heater control malfunction.

Prerequisites

- Instrumentation monitoring the feedwater system and the feedwater heater operating parameters has been calibrated and is functioning normally.
- Feedwater system and extraction heating steam have been placed in normal operation.
- Portable test instrumentation capable of injecting control loop test signals is available and within calibration due date.

Test Method

- Check for proper setting of feedwater heater level controls.
- Check for proper setting of feedwater heater bypass controls.
- Verify correct operation of extraction steam non-return valves and isolation valves to prevent turbine water induction.
- Verify correct operation of condensate recirculation and bypass isolation valves.
- Verify correct operation of annunciator and indicating lights in Main and Auxiliary Control Rooms.

Acceptance Criteria

- Feedwater heater controls perform the design function of bypassing feedwater heaters or returning condensate to the main condenser, isolation of extraction steam, and verify correct response of remote and local alarms, indicating lights, and interlocks.

Modify SSAR subsection 14.2.8.2.43 as follows:

14.2.8.2.43 Thermal Power Measurement and Statepoint Data Collection

Objectives

- Obtain thermal power measurement and statepoint data at selected power levels during the power ascension testing program, typically at 25, 50, 75, and 100% of rated thermal power.
- Compare measured data from primary instrumentation with process and control room computer indications.

Prerequisites

- The following equipment is installed and is checked out and operational: sensors for measuring steam generator feedwater temperature, differential pressure measuring devices for determining feedwater flow to each steam generator, and pressure gauges to measure steam pressure at steam generator outlets.
- The following control systems are in automatic: pressurizer pressure and level, and steam generator level.
- The plant process computer is available for logging supplemental plant data.
- Reactor power is stable at the required level.

Test Method

- The required data are obtained using installed plant equipment, special test equipment, and the plant process computer. These data are subsequently used to determine reactor thermal power and assess the performance of the plant.

Performance Criterion

- The process computer measured parameters agree with the primary measurements within prescribed tolerances.~~None. This test is only for the collection of data.~~

Question 260.31

The preoperational and startup test phase descriptions in Section 14.2.8, "Individual Test Descriptions," of the SSAR do not provide assurance that the operability of several of the systems and components listed in the following regulatory guides will be demonstrated. The test abstracts of Section 14.2.8 of the SSAR should be expanded to address the following items, or Section 1A of the SSAR should be revised to provide technical justification for any exceptions taken.

a. Regulatory Guide 1.68.2, "Initial Startup Test Program To Demonstrate Remote Shutdown Capability For Water-Cooled Nuclear Power Plants" - Preoperational test abstract 14.2.8.1.94, "Remote Shutdown" does not provide sufficient detail to verify conformance with the following Regulatory Positions (RPs) of RG 1.68.2.

1. Hot Standby Demonstration (RP C.3), including:

   A. With initial conditions of the reactor at a moderate power level (10 to 25 percent) sufficiently high that plant systems are in the normal configuration with the turbine generator in operation and with the minimum shift crew;

   B. Demonstrate using only credited remote shutdown equipment the capability to achieve hot standby status and maintain stable hot standby conditions for at least 30 minutes.

2. Cold Shutdown Demonstration (RP C.4), including:

   A. With the plant at hot standby conditions;

   B. With the procedurally designated crew positions;

   C. Demonstrate using only credited remote shutdown equipment the capability to perform a partial cooldown by performing the following actions:

      (1) Lower reactor coolant pressure and temperature sufficiently to permit operation of the RHR system;

      (2) Initiate and control operation of the RHR system;

      (3) Establish a heat transfer path to the ultimate heat sink,

      (4) Reduce reactor coolant temperature approximately 50°F using the DHR system.

b. Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems" - Preoperational test abstract 14.2.8.1.6, "Compressed and Instrument Air Systems" does not provide sufficient detail to verify conformance with the following RPs of RG 1.68.3:

1. After coolers, oil separators, air receivers, and pressure-reducing stations (RP C.2);

2. Flow, temperature, and pressure meet design specifications (RP C.4);

3. Total air demand with leakage meets design (RP C.5);

4. Single failure criterion (RP C.7);

5. Sudden and gradual loss of system pressure and appropriate response of air-powered equipment (RP C.8);

6. Functional test for increase in the air supply system pressure does not cause loss of operability (RP C.11).

c. Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria For Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" - Preoperational test abstracts 14.2.8.1.28, "Containment Air Filtration System," 14.2.8.1.29, "Radiologically Controlled Area Ventilation Test," and 14.2.8.1.88, "High-Efficiency Particulate Air Filters and Charcoal Absorbers" do not provide sufficient detail to verify conformance with the following RR of RG 1.140:

1. Heaters (RP C.3.a);

2. Prefilters (RP C.3.m);

3. HEPA filters DOP tests (RPs C.3.b and C.5.c);

4. Ductwork (RP C.3.f);

5. Fans and motors mounting and ductwork (RP C.3.i);

6. Dampers (RP C.3.l);

7. Adsorber sections/cells and activated charcoal (RPs C.3.h and C.5.d).

Response:

a.1 Requirements of RG 1.68.2 will be met by Test Abstract 14.2.8.1.103, "Remote Shutdown Using Passive Systems (first plant only)," which will be added as a new test abstract to include Passive Core Cooling System controls at the remote location. The actual plant procedure prepared by the COL Applicant will contain the Level of detail shown in RG 1.68.2.

a.2 Requirements of RG 1.68.2 regarding Cold Shutdown Demonstration (RP C.4) are met by Test Abstract 14.2.8.1.94, "Remote Shutdown (first plant only)".

No test abstract needs to be added or amended.

b. Regulatory Guide 1.68.3 "Preoperational Testing of Instrument and Control Air Systems" applies to safety application of instrument air systems. Preoperational Test Abstract 14.2.8.1.6, "Compressed and Instrument Air Systems", applies to the non-safety related AP600 instrument air system. No safety related function is prevented by malfunction of this system.

No test abstract needs to be added or amended.

c. Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" applies to safety ventilation systems. Abstracts 14.2.8.1.28, and 14.2.8.1.88 apply for non-safety related systems. These systems are not necessary for safe shutdown of the plant. Nonradioactive Ventilation Systems are also tested per AP600 Inspection Test Analysis and Acceptance Criteria Sections 3.7.1, 3.7.3 and 3.7.4. Abstract 14.2.8.1.29 Radiologically Controlled Area is also non-safety related. The plant procedures for the nonradioactive Ventilation Systems will contain detail as required by RG 1.140 and is the responsibility of the Combined License applicant.

No test abstracts needs to be added or amended.

SSAR Revision:

Revise SSAR subsection 14.2.8.1.103 to include the following test abstract:

14.2.8.1.103 Remote Shutdown from 10 to 25 Percent Power (First Plant Only)

Objectives

- With the reactor at 10 to 25 percent power and the turbine generator in operation, demonstrate the ability to cooldown the plant using controls and instrumentation located outside the control room.

Prerequisites

- Initial conditions of the reactor at a moderate power level (10 to 25 percent), sufficiently high that plant systems are in the normal configuration with the turbine generator in operation.

Test Method

- With a minimum crew and using only credited remote shutdown equipment, achieve hot standby status and maintain stable hot standby conditions for at least 30 minutes.

Performance Criteria

- The ability to reach and maintain hot standby conditions using remote controls and instrumentation from outside the control room has been demonstrated.

PRA Revision: NONE

Question 410.206

The February 25, 1993, response to Q410.60 states that hydrogen is supplied to the CVS inside containment from one 550 scf H_2 bottle located in the plant gases storage tank area. The maximum concentration within the CVS compartment was found to be 4.3 percent, less than the detonation limits in NUREG/CR-2017. Areas other than the CVS compartment were also considered with the maximum concentration being ~4.4 percent in the valve/piping penetration room at the 100' elevation of the Auxiliary Building (12420 ft^3). However, this apparently assumes uniform mixing within the containment. How is this assured?

Additionally, the CVS has high-energy (HE) portions in the auxiliary building that are not designed to Code requirements. Specifically, this includes the portion of CVS from the makeup pumps to the CIVs. Are these HE portions separated from safety-related equipment in the auxiliary building? If so, what is the nature of the separation? Is it by physical spacing, by separate enclosures, or by the use of barriers? How will safety-related SSCs be protected from missiles generated during a postulated failure of this portion of the CVS?

Response:

The maximum hydrogen concentration of the CVS compartment inside containment was calculated assuming that a failure of the hydrogen supply line releases the entire contents of a single hydrogen bottle into that compartment. In calculating the maximum hydrogen concentration, uniform mixing within the CVS compartment was assumed. No credit was taken for hydrogen escaping from the CVS compartment into other areas of the containment, and thus mixing with other parts of the containment is not a factor.

Similarly, the maximum hydrogen concentration in the valve/piping room in the auxiliary building was calculated assuming that the contents of a single hydrogen bottle is released and uniform mixing occurs. No credit was taken for hydrogen escaping into other areas of the auxiliary building.

The CVS high energy lines located in the auxiliary building are routed from the CVS makeup pumps to the containment penetration via a pipe chase except for a small segment of the lines located in the room containing the makeup pumps. These lines are not completely separated from safety-related lines. Containment penetrations and containment isolation valves are located in the pipe chase. The ability to achieve safe shutdown given a rupture is maintained. Complete separation is not necessary.

The ability to achieve safe shutdown given a rupture in one of the high energy lines from the CVS makeup pumps is maintained by the following features:

1. The CVS makeup line is routed through a pipe chase with lines that are not required for safe shutdown of the plant.

2. Containment isolation valves located in the pipe chase with the CVS makeup line are located sufficiently far away from the CVS pipe break locations.

Pipe breaks are postulated to occur at the terminal ends of a piping run for those lines which are not designed using mechanistic pipe break criteria. The terminal ends are defined as either:

A. Connection points to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.

B. Branch intersection points for the branch line unless the following are met: the branch and the main piping systems are modeled in the same static, dynamic and thermal analyses and the branch and main run are of comparable size and fixity.

The terminal ends for the CVS piping located outside containment are at the discharge flanges of the CVS makeup pumps, at an anchor located approximately midway between the pumps and the containment penetration and at the containment penetration. Pipe breaks are not postulated at intermediate points and the total pipe stress is controlled. Due to the small nominal diameter of the CVS lines, only circumferential breaks are postulated at these terminal ends. Containment isolation valves for other lines are not located near the three postulated pipe break locations.

3. Safety related equipment located in the auxiliary building is separated from the CVS makeup line by the reinforced concrete walls.

4. The safety related lines in the vicinity of the containment penetration for the CVS charging line are not required for safe shutdown.

5. The room containing the CVS makeup pumps does not contain equipment required for safe shutdown.

6. Since the CVS supply line always contains cold water there are no environmental or pressurization affects from a CVS line break. Water spray affects can not damage process piping in the pipe chase. There are no safety related electrical connections or instrument lines in the vicinity of the pipe break locations.

There are no missiles postulated which could impact safety related structures, systems and components (SSCs) along this portion of the CVS high energy line.

SSAR Revision: NONE

Question 440.12

Since Revision 0 to WCAP-13342 was issued, the 4th stage of the ADS has been redesigned to incorporate larger valves (12" vs. 8") and a different configuration with respect both to number of valves and connection to the RCS hot leg piping. Westinghouse has previously committed to test these valves as part of the ADS test program; however, no information has been provided to the staff specifying the test loop configuration or the test matrix envisioned for these tests. This information should be provided for staff review. The level of detail in this information should be commensurate with that requested for the Phase A and Phase B tests.

Response: (Revision 1)

~~The size of the fourth stage ADS valve has increased from an 8-inch valve to a 12-inch valve. A specific test will be run on the fourth stage valve. The fourth stage ADS valve test will examine not only the flow effects of the valve geometry, but the mechanical design aspects of the valve design such as opening torques, loads, sealing capabilities and sizing of the valve operator. The component testing of the valve will be performed as a component verification test following design certification, when the detailed valve design is complete.~~

~~In the interim, if there are questions on the flow through the fourth stage valve, they can be addressed with sensitivity studies. The resulting flow behavior will be confirmed by the fourth stage tests performed on the specific valve design which will be used in the plant.~~

The fourth stage ADS valves will be tested for equipment qualification purposes. Design certification testing is not required. The fourth stage of ADS is simpler than the first three stages of ADS because it has fewer parallel paths, the downstream piping is completely separate and the discharge paths are straight pipes that discharge into the containment atmosphere and not under water through a sparger. As discussed with NRC staff on April 7, 1994, the following steps will be performed to verify the performance of these valves:

1) A sensitivity study has been performed under design certification, where the capacity of the fourth stage valves is reduced. This study was provided to the NRC via Westinghouse letter NTD-NRC-94-4298.

2) The prototype ADS fourth stage valves will be tested during their equipment qualification program. This testing will show that the valves can pass the minimum amount of steam at the limiting RCS pressures. This pressure is anticipated to be in the range of 100 psia. A more detailed discussion on the valve testing will be provided in response to RAI 952.96.

SSAR Revision: NONE

Question 480.36

Hydrogen Control - Diffusion Flames Above IRWST

Discuss the potential for and impact on the steel containment of diffusion flames above the in-containment refueling water storage tank (WCAP 13388).

Response: (Revision 1)

In order to meet the conditions for a diffusion flame to form above the in-containment refueling water storage tank (IRWST), a steady stream of hydrogen must be released into the IRWST water from the reactor coolant system (RCS) through the automatic depressurization system (ADS) sparger. ~~A diffusion flame sequence must be a small LOCA or non-LOCA initiated event in which one of the first three ADS stage operates, and both fourth stage valves fail. If ADS were to fail completely, the hydrogen flow to the containment would be through the RCS safety valves which are directed to the containment upper compartment atmosphere, bypassing the IRWST. If the event were initiated by a medium or large LOCA, or the fourth stage ADS were to operate, the RCS hydrogen would be vented preferentially through the break or the fourth stage valve, bypassing the IRWST. A review of the cutsets shows that a limited number of sequences fit the accident profile required to create a diffusion flame in the IRWST. All of the sequences which could form a diffusion flame are passive residual heat removal heat exchanger tube rupture sequences with failure of fourth stage ADS. The total frequency of these sequences is less than 2.0 x 10^{-9} per reactor year. This frequency is less than 1% of the core damage frequency presented in the AP600 PRA report.~~

~~The impact of diffusion flames from the IRWST vents is not accounted for in the current AP600 PRA. Further consideration of this scenario will be provided in the PRA revision in February 1994.~~

Standing diffusion flames on the IRWST pool or at the IRWST vents can be postulated early into an accident following core uncovery for sequences where the igniters provide an ignition source and the ADS provides a pathway for the hydrogen to collect in the IRWST. Hydrogen which collects in the pressurizer as the cladding is damaged can be forced out through the ADS sparger into the IRWST during a reactor coolant system pressure transient, such as fuel relocation into the water in the lower head or core reflood. The degree of mixing of hydrogen and oxygen from the containment air that can occur in the gas space of the IRWST determines whether the burning can occur in the tank or at the vent exit. A standing diffusion flame at the vent could present a significant thermal load to the containment steel shell which is in close proximity to some of the vents. However, a stable, standing flame at the vent exit is considered to be highly unlikely. Hydrogen released to the IRWST is expected to mix with containment air in the gas space above the IRWST water such that, when ignited by an igniter at the vent exit, the flame front flashes back and consumes the bulk of the hydrogen in the tank. Depending on the amount of hydrogen released and the duration, this could result in periodic deflagrations in the IRWST. This type of burning does not result in a significant thermal loading on the containment shell since the burn only lasts for a short time (on the order of seconds), and the thermal inertia of the containment wall would prevent the temperature from increasing more than several degrees.

Hydrogen mixing analyses and, if needed, adjustment to the vent design will be used to assure that the diffusion flame induced containment failure mode makes an insignificant contribution to the large release frequency in the AP600 Probabilistic Risk Assessment. Diffusion flames at the IRWST vent exits are not considered as a containment failure mode in the PRA, revision 1 analysis.

SSAR Revision: NONE

PRA Revision: NONE

Question 480.77

Containment isolation valves should be as close to the containment wall as is practical. From the staff's review of the AP600 design, it appears that several lines have considerable runs (greater than 10 feet) inside the containment before the interior containment isolation valve is encountered. An example is the service modules or islands that are incorporated into the AP600 design. Provide a list of lines that have runs greater than 10 feet, and justify placing the containment isolation valve so far from the containment boundary in each case.

Response:

The use of service modules or equipment modules on the AP600 design is not used as a criteria for the location of containment isolation valves. An AP600 layout criteria specifies that "Valves used for containment isolation, which are Class 2 components, should be located as close as practical to the containment. However, sufficient space for in-service inspection must be provided." This layout criteria has been incorporated into the AP600 design process and the subject valves are located at or very near the associated penetration unless overriding considerations must be factored into the design process. Overriding considerations are typically due to providing sufficient space (including access) to the valve to assure proper inspection, maintenance and testing. Improper location without adequate consideration of these factors potentially results in a less reliable valve and would defeat any perceived benefit derived from proximity to the containment. Less obvious, but nonetheless critical layout considerations include the following.

- Flooding: As reported in response RAI 410.5: "The containment isolation valves subject to flooding are normally closed isolation valves and are not required to stroke under flooded conditions." Therefore containment isolation valves that are not normally closed may need to be raised above the flood level and depending on the containment penetration location may be located to assure more reliable containment isolation under potential environments rather than a specific distance from the containment.

- Shielding: Some penetrations, piping and valves warrant special considerations due to shielding for personnel protection and consequently, the line routing and the valve location limits personnel exposure during maintenance, inspection and testing of the subject valve as well as other equipment in the vicinity.

- Platform/Floor Elevation: Containment isolation valves are located at elevations where maintenance platforms or floor levels readily permit valve maintenance and inspection without the need for construction of temporary platforms resulting in delays, increased costs, and exposure.

- Penetration Area: For piping penetrations in the middle annulus (elevation 100') it is undesirable to locate valves and additional pipe lengths in the electrical penetration area so the lines either drop to the lower annulus (elevation 92'6") or extend into the valve piping and penetration rooms in the auxiliary building. The containment isolation valves are then located as near as practical to the penetration consistent with other criteria.

- NPSH: When NPSH or void formation or thermal fatigue (due to isolation valve leakage) are a concern, for example within the RNS, the pipe routing slopes to the pump suction and the containment isolation valve is located as near containment as possible consistent with other layout criteria.

- Electrical Separation: When there be a need to limit introducing an electrical division in a particular area either the valve is relocated in an alternate area or the power division is reassigned to achieve adequate redundancy and fire protection.

In summary a specific distance such as 10 feet is not the critical parameter, but the implementation of a layout criteria that focuses appropriate attention on a balance of criteria is essential and has been implemented on the AP600. As piping analysis is completed and criteria are met the containment isolation valves will be located as close to the containment consistent with those criteria.

SSAR Revision: NONE

Question 952.90

Provide the following information on the PRHR tests:

a. The detailed test procedures report (test conditions) for both Phase I and II tests. The staff has noted that in the Westinghouse procedures, the document is usually a checklist that refers to other documents that actually describe the different steps that were followed. Provide the initial and boundary conditions, configurations, and operator actions that caused the experiment to go the way it went. Define what is meant by steady state, what was the exact configuration of each test (i.e., water level), and when was the test complete. Do the test procedures address these questions?

b. A list of acronyms defining the data for tests SS-3B and TR-4, and for any new test data that the staff will receive.

c. Provide an evaluation of the measurement uncertainty for all parameters recorded. This should be included as part of the data analysis report.

d. Provide data in electronic format from the Phase II tests, including:

1. Configuration Tests C-01 and C-02
2. Plume Tests P-01, P-03, and P-05
3. Transient Tests T-01 through T-03
4. Steady State Tests S-01 through S-05
5. Uncovery Tests U-01 through U-05

Response:

a. Test procedures are summarized in section 4.5 of the PRHR test final report (reference 952.90-1). The detailed test procedures used by the test organization were provided to the NRC via Westinghouse letter NTD-NRC-94-4288. The test procedures address the questions posed.

b. The acronyms used in the test report are described as they are used in the data reduction and anlaysis methodology. This is contained in section 6.0 of the PRHR test final report (reference 952.90-1).

c. An evaluation of the measurement uncertainty was performed. This information is being incorporated into a revision to the PRHR final test report (i.e., revision 2) which will be provided by December, 1994.

d. The data from the PRHR tests was transmitted to the NRC via Westinghouse letter NTD-NRC-94-4288 in electronic format with the exception of the plume tests. The test data for these tests was not recorded electronically but was recorded on strip charts. This data is presented in figures 7.2-1 through 7.2-7 of reference 952.90-1. The strip charts are maintained in the Westinghouse files and are available for review at the Westinghouse offices in Monroeville, PA.

References:

952.90-1 "AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report", WCAP-12980 Revision 1 dated December, 1992

SSAR Revision: NONE
PRA Revision: NONE

Question 952.92

Provide the following information on the Oregon State University facility:

a. The Set Point and Valves document.

b. The OSU Test Matrix Table (initial boundary conditions)

c. The pre-operational test procedures and test reports

d. Updated P&IDs

e. Detailed pump data for RCPs (flow curve, homologous quadrant data if available)

f. Valve & instrumentation description (e.g., table relating number, name or description, drawing number, range)

g. Louis K. Lau letter on orifice and nozzle requirements

Response:

a. Copies of the requested OSU setpoint document and OSU valve list were submitted to the NRC via reference 952.92-12.

b. A copy of the OSU Test Matrix table is contained in each of the Quick Look Data Reports (QLR) submitted to the NRC (References 952.92-7 through 952.92-11). Reference 952.92-12 provides additional details on the facility setup for each of the OSU matrix tests. Each QLR includes a comparison of specified and measured initial conditions.

c. Test procedures are maintained at OSU and in the Westinghouse files and are available for review by the NRC at the test facility or at the Westinghouse Energy Center. Quick Look Reports for the cold and hot pre-operational tests have been completed and submitted to the NRC (References 952.92-2 through 952.92-6). These reports contain a summary description of the procedures used during the tests as well as a tabulation of the test results.

d. Up to date, piping and instrumentation diagrams are provided in reference 952.92-1, WCAP-14124, Volume II, Appendix B, which was transmitted to the NRC via Westinghouse letter NTD-NRC-94-4245, dated July 29, 1994. Revision 4 of OSU drawing OSU600901, "OSU BAMS System" was provided to the NRC via reference 952.92-12.

e. Design information on the OSU test facility coolant pumps is provided in Section 3.10.3 of reference 952.92-1. Figure 3.10-1 of this reference provides RCP performance head vs. flow. Homologous quadrant data for the pumps is not available.

f. OSU test facility instrumentation database information is provided in Appendix D of WCAP-14124 (Reference 952.92-1). OSU test facility valve database information is provided in reference 952.92-12.

g. Orifice and nozzle details are provided in Appendix E of WCAP-14124 (Reference 952.92-1).

Reference:

952.92-1 WCAP-14124, "AP600 Low Pressure 1/4 Height Integral Systems Tests - Facility Description Report"

952.92-2 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: OSU Component Volume Determinations, Enclosure to Westinghouse letter NTD-NRC-94-4219, dated July 29, 1994.

952.92-3 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: OSU Flow Test/Line Resistance Determinations, Enclosure to Westinghouse letter NTD-NRC-94-4220, dated July 29, 1994.

952.92-4 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Hot Preoperational Test HS01, Enclosure to Westinghouse letter NTD-NRC-94-4221, dated July 29, 1994.

952.92-5 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Hot Preoperational Test HS02, Enclosure to Westinghouse letter NTD-NRC-94-4222, dated July 29, 1994.

952.92-6 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Hot Preoperational Test HS03, Enclosure to Westinghouse letter NTD-NRC-94-4223, dated July 29, 1994.

952.92-7 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Matrix Test SB01, Enclosure to Westinghouse letter NTD-NRC-94-4224, dated July 29, 1994.

952.92-8 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Matrix Test SB04, Enclosure to Westinghouse letter NTD-NRC-94-4225, dated July 29, 1994.

952.92-9 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Matrix Test SB13, Enclosure to Westinghouse letter NTD-NRC-94-4226, dated July 29, 1994.

952.92-10 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Matrix Test SB10, Enclosure to Westinghouse letter NTD-NRC-94-4227, dated July 29, 1994.

952.92-11 Quick Look Data Report for AP600 Long Term Cooling Tests at Oregon State University: Matrix Tests SB03, SB05, SB12, Enclosure to Westinghouse letter NTD-NRC-94-4268, dated August 22, 1994.

952.92-12 Westinghouse letter NTD-NRC-94-4285, "Additional Information in Support of Westinghouse Response to RAI 952.92," dated August 31, 1994

SSAR Revision: NONE
PRA Revision: NONE