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Your ref: Docket No. 52-006
Our ref: DCP/NRC1592

May 21, 2003

SUBJECT: Transmittal of Westinghouse Responses to US NRC Requests for Additional Information on the AP1000 Application for Design Certification

This letter transmits the Westinghouse responses to NRC Requests for Additional Information (RAI) regarding our application for Design Certification of the AP1000 Standard Plant. A list of the RAI responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the RAI responses.

Please contact me if you have questions regarding this submittal.

Very truly yours,

M. M. Corletti

M. M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects

/Attachments

1. Table 1, "List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1592"

2. Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission Requests for Additional Information dated May 2003

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Attachment 1

"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1592"

Docket No. 52-006

DCP/NRC1592

May 21, 2003

Attachment 1

Table 1 **"List of Westinghouse's Responses to RAIs Transmitted in DCP/NRC1592"**
220.007, Rev. 1 230.022, Rev. 0 252.010, Rev. 0 252.011, Rev. 0 261.015, Rev. 0 261.018, Rev. 0 410.007, Rev. 3

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DCP/NRC1592

May 21, 2003

Attachment 2

Westinghouse Non-Proprietary Response to US Nuclear Regulatory Commission
Requests for Additional Information dated May 2003

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 220.007 Revision 1

Question:

AP1000 DCD Subsection 3.8.3.3.1, "Passive Core Cooling System Loads," describes the pressure and thermal transients associated with operation of the passive core cooling system which are used to evaluate structures inside containment. Some of the water temperature transients have changed from the AP600 design and it is not clear how these have affected the analysis and design of the structural modules. Therefore, please address the following issues:

A. The transient temperature was revised from 240°F reached in 5.5 hours (AP600) to 250°F reached in 3.5 hours (AP1000). Provide a discussion to explain the change and how this change was considered in the analysis and design of the AP1000 modules?

B. The extreme transient starting temperature used for the structural design was revised from 50°F (AP600) to 70°F (AP1000). This would seem to be less extreme than the 50°F case in the AP600 design. Provide the basis for this change and explain how this change was considered in the analysis and design of the AP1000 modules?

NRC Additional Comment:

This RAI response was discussed at the NRC Structural Meetings held at the Westinghouse offices on April 2-5, 2003. At the meeting, it was requested that additional clarification and justification of the analysis assumptions be provided in a revised RAI response.

Westinghouse Revision 1 Response:

The thermal transient associated with operation of the passive core cooling system that was provided in the Revision 0 response has been revised. The previous transient was calculated with a simplified bounding model that resulted in overly conservative thermal gradients across the IRWST wall. The calculation has been revised using the Westinghouse WGOTHIC containment model that is used to perform the Chapter 6 design basis containment analysis.

This design transient is determined by a calculation of the temperature transient of the IRWST water following operation of the passive RHR heat exchanger. Upon actuation of the PRHR heat exchanger, natural circulation flow through the heat exchanger removes core decay heat and reactor coolant system sensible heat. This heat transfer process causes the temperature of the water in the IRWST to increase until it begins to boil. As the IRWST boils, the steam that is produced causes the containment pressure and temperature to increase. Once the maximum containment pressure is achieved, then the IRWST water temperature remains constant.

The calculation also determines the corresponding containment air temperature associated with the design transient such that the maximum temperature gradient across the IRWST wall for purposes of structural design of the IRWST wall.

The WGOTHIC containment model was used to determine the IRWST water temperature as a function of time. Decay heat is added to the IRWST using a heater component to represent the PRHR heat exchanger. The following assumptions are used in this model:

- ANSI-79 decay heat (best estimate)
- IRWST water volume = 73,900 ft^3
- IRWST initial temperature = 50°F
- Complete mixing in the IRWST (no stratification)
- No heatup of IRWST walls or other structures (all decay heat deposited in IRWST)
- Full reactor power of 3400 MWt prior to simultaneous scram and PRHR actuation.
- Containment pressure conditions as calculated by WGOTHIC.
- RCS structure heat capacity of 1.2E6 Btu/°F released to IRWST over 5 hour period

The IRWST temperature transient is shown in the attached revised DCD Figure 3.8.3-7. The temperature transient has been modified based on a more conservative assumption regarding the cooldown time of the reactor coolant system.

The IRWST begins steaming approximately 4 hours after the event is initiated, and reaches a maximum temperature of 260°F in approximately 11 hours. The containment atmosphere heats up once the IRWST steams. This transient will be used to determine the thermal loads on the IRWST walls. Also shown in Figure 3.8.3-7 is the CMT room atmosphere temperature which serves as the boundary condition on the opposite side of the wall. There are no changes in the analysis or design criteria for the structural modules for thermal effects. The critical loading for the concrete filled steel plate modules above the containment internal structure basemat is the thermal gradient across the wall. Close to the containment internal structure basemat the critical loading is the maximum temperature difference between the steel plate and the basemat concrete. The design of the structural modules for the AP1000 IRWST transient will be available for review during the NRC audit of critical sections.

Design Control Document (DCD) Revision:

DCD Section 3.8.3.3.1 will be revised as follows:

- ADS_2 – This automatic depressurization system transient considers heatup of the water in the in-containment refueling water storage tank. This may be due to prolonged operation of the passive residual heat removal heat exchanger or due to an automatic depressurization system discharge. For structural design, an extreme transient is defined starting at 50°F since this maximizes the temperature gradient across the concrete-filled structural module walls. Prolonged operation of the passive residual heat removal heat

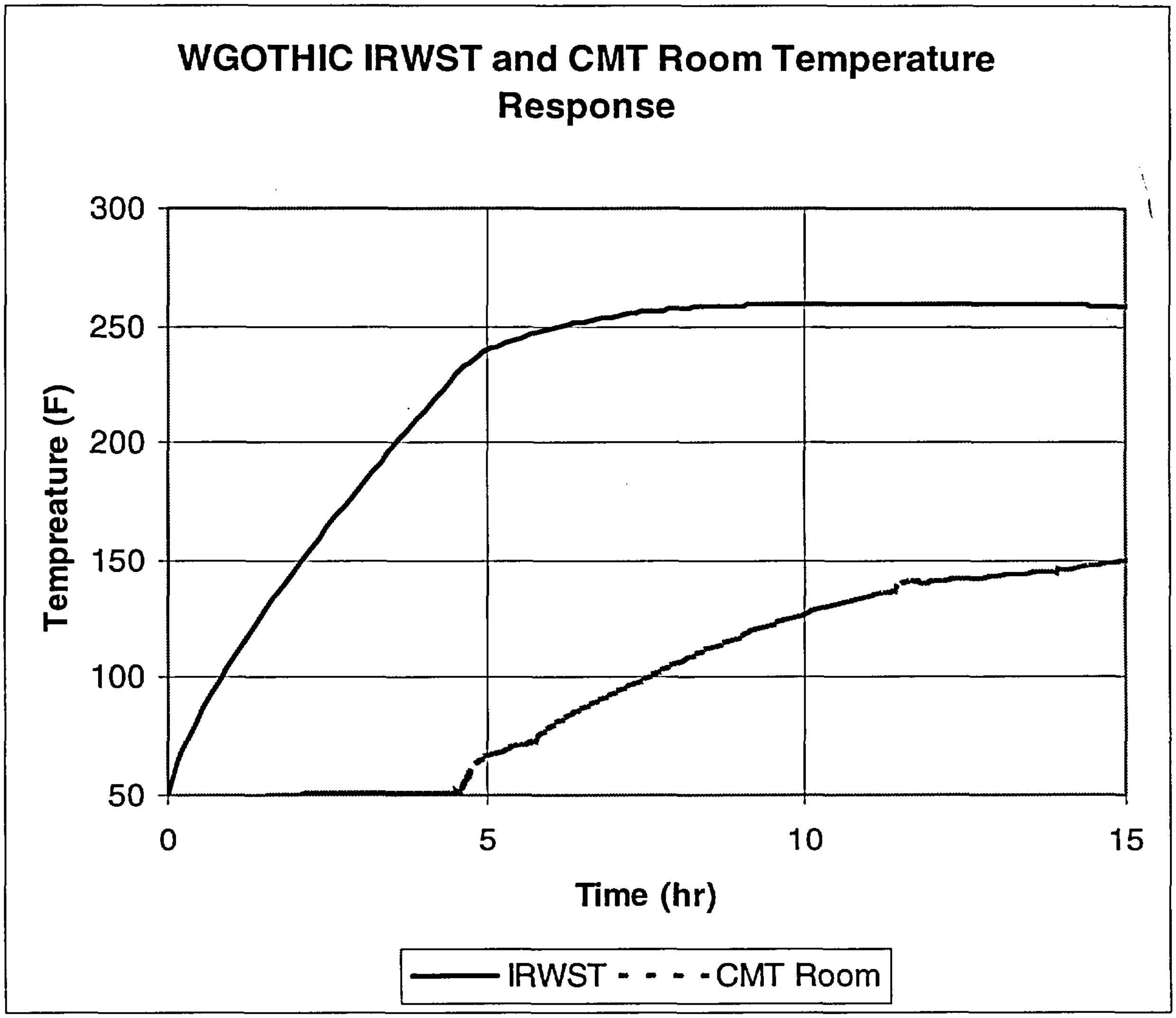
exchanger raises the water temperature from an ambient temperature of 50°F to saturation in about ~~5~~**4** hours, increasing to about 260°F within about ~~11~~**10** hours. Steaming to the containment atmosphere initiates once the water reaches its saturation temperature. For structural design an extreme transient is defined starting at 70°F since this maximizes the temperature gradient across the concrete filled structural module walls. The temperature transient is shown in Figure 3.8.3-7. Blowdown of the primary system through the spargers may occur during this transient and occurs prior to 24 hours after the initiation of the event. Since the flow through the sparger cannot fully condense in the saturated conditions, the pressure increases in the in-containment refueling water storage tank and steam is vented through the in-containment refueling water storage tank roof. The in-containment refueling water storage tank is designed for an equivalent static internal pressure of 5 psi in addition to the hydrostatic pressure occurring at any time up to 24 hours after the initiation of the event.

The attached figure replaces DCD Figure 3.8.3-7

PRA Revision:

None

Revised Figure 3.8.3-7
IRWST Temperature Transient



Westinghouse Original Response

The thermal transients associated with operation of the passive core cooling system have been revised. The extreme transient starting temperature used for the structural design has been revised to 50°F and is now the same as was used for the AP600.

The AP1000 containment model was used to determine the IRWST water temperature as a function of time. Decay heat is added to the IRWST using a heater component to represent the PRHR heat exchanger. The following assumptions are used in this model:

- ANSI-79 decay heat with 2σ uncertainty
- IRWST water volume = 75,300 ft^3
- IRWST initial temperature = 50°F
- Complete mixing in the IRWST (no stratification)
- No heatup of IRWST walls or other structures (all decay heat deposited in IRWST)
- Full reactor power of 3400 MWt prior to simultaneous scram and PRHR actuation.
- Containment pressure conditions as calculated by WGOTHIC.

The IRWST temperature transient is shown in the attached Figure 3.8.3-7, which will be added in the DCD. The IRWST begins steaming approximately 5 hours after the event is initiated, and reaches a maximum temperature of 259°F in approximately 11 hours. The containment atmosphere heats up once the IRWST steams. This transient will be used to determine the thermal loads on the IRWST walls. There are no changes in the analysis or design criteria for the structural modules for thermal effects. The critical loading for the concrete filled steel plate modules above the containment internal structure basemat is the thermal gradient across the wall. Close to the containment internal structure basemat the critical loading is the maximum temperature difference between the steel plate and the basemat concrete. The design of the structural modules for the AP1000 IRWST transient will be available for review during the NRC audit of critical sections.

Design Control Document (DCD) Revision:

Revise second bullet in subsection 3.8.3.3.1:

- ADS_2 – This automatic depressurization system transient considers heatup of the water in the in-containment refueling water storage tank. This may be due to prolonged operation of the passive residual heat removal heat exchanger or due to an automatic depressurization system discharge. **For structural design an extreme transient is defined starting at 50°F since this maximizes the temperature gradient across the concrete filled structural module walls.** Prolonged operation of the passive residual heat removal heat exchanger raises the water temperature from an ambient temperature of ~~120~~**50**°F to saturation in about ~~2~~**5** hours, increasing to about 2~~5~~60°F within about **11**~~3.5~~ hours. Steaming to the containment atmosphere initiates once the water reaches its saturation

temperature. The temperature transient is shown in Figure ~~19.E.4.10-4~~**3.8.3-7.** Blowdown of the primary system through the spargers may occur during this transient and occurs prior to 24 hours after the initiation of the event. Since the flow through the sparger cannot fully condense in the saturated conditions, the pressure increases in the in-containment refueling water storage tank and steam is vented through the in-containment refueling water storage tank roof. The in-containment refueling water storage tank is designed for an equivalent static internal pressure of 5 psi in addition to the hydrostatic pressure occurring at any time up to 24 hours after the initiation of the event.

PRA Revision:

None

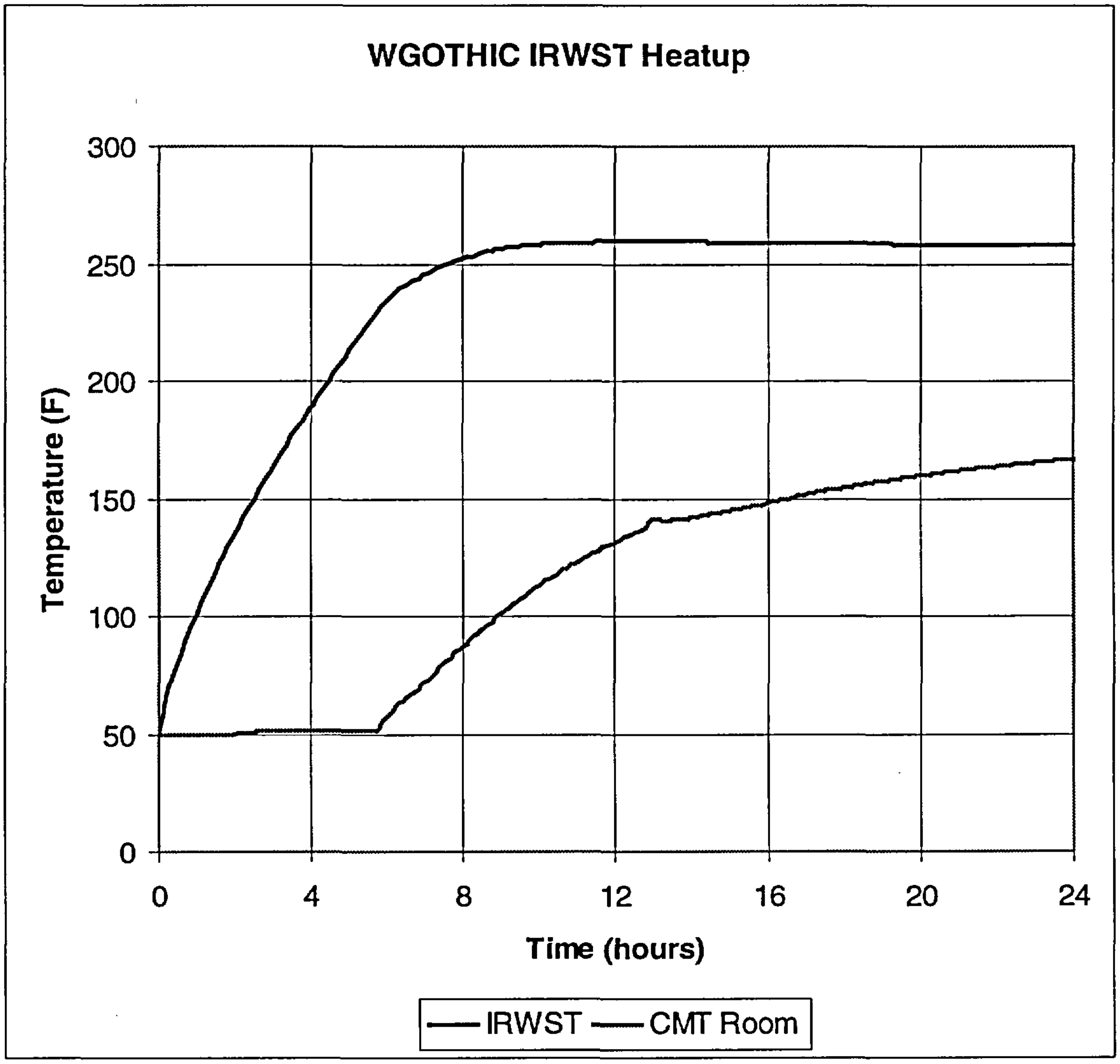


Figure 3.8.3-7
IRWST Temperature Transient

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 230.022

Question:

Analyses for Loads during Operation:

The basemat is represented by a three-dimensional finite element model with the computer program ANSYS (Design Control Document (DCD) Reference 21). The model considers the interaction of the basemat with the overlying structures and with the soil. Provisions are made in the model for two possible uplifts. One is the uplift of the containment internal structures from the lower basemat. The other is the uplift of the basemat from the soil. The staff notes that uplift and slapping back of the containment internal structures on the basemat would affect both the seismic design loads and in-structure response spectra for all SSCs associated with the containment internal structure, and would also affect the seismic response of the steel containment shell. Please clarify how these effects have been addressed in the seismic analyses used for design of the containment and containment internal structures.

Westinghouse Response:

The bottom head of the containment vessel rests on the nuclear island basemat. The containment internal structures basemat rest within the bottom head. There are no shear studs or anchors designed to transfer loads tangential to the vessel surface. The interface is designed to transfer load in compression and friction. The configuration is identical to the AP600 and the stability evaluation shown in Figure 230.022-1 follows the AP600 methodology described in the AP600 response to RAI 230.47.

The provisions in the nuclear island basemat model are included for use in the equivalent static analyses to develop design loads for basemat design. The uplift capability assures that the reaction between the two basemats is correctly transferred as compression loads only. The stability evaluation shows a factor of safety against overturning of about 2.5. Since the deadweight has not been overcome no "liftoff or slapping" is expected to occur. However, allowing for a small separation of the containment internal structures from the basemat, there would be no significant effect to the seismic design loads or the in-structure response spectra. A small separation (slapping) might cause small localized changes in seismic response loads similar to those for the lift off observed between the nuclear island basemat and the rock addressed in the response to RAI 230.021. Any change in high frequency response due to "slapping back" would not propagate through the building structures to affect the seismic response. This is because of energy loss, damping, and filtering effects due to gaps and cracking. Therefore, it is not necessary to modify the analysis methods from those that were accepted by the NRC for the AP600 plant.

Design Control Document (DCD) Revision: None

PRA Revision: None

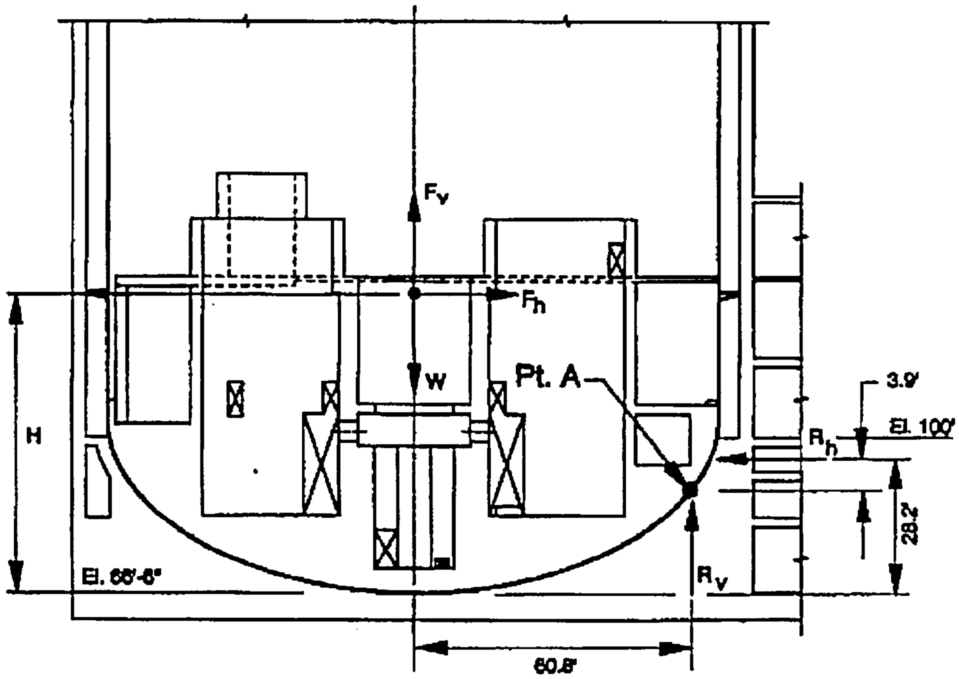


Figure 230.022-1
Free-body Diagram for Containment Internal Structures Overturning Evaluation

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 252.010

Question:

In RAI 252.001, the staff requested information related to the geometry, fabrication, materials, accessibility for inspection, and operating conditions for control rod drive system penetrations, as motivated by recent operating experience, NRC Bulletins 2001-01, 2002-01, and 2002-02. Since the RAI was issued, the staff has issued Orders to operating license holders related to inspection for cracks in these penetrations. The staff subsequently issued follow-up questions to Westinghouse related to changes in design and fabrication to reduce residual stresses, ability to visually inspection 360 degrees around each nozzle, pre-service volumetric inspection, and determination of the operating head temperature. Westinghouse responded to the follow-up questions in a letter dated April 7, 2003.

The NRC staff believes that new inspection, test, analyses, and acceptance criteria (ITAACs) should be included in the DCD to address the issues discussed in your RAI responses. Please provide proposed ITAACs related to the issues noted above.

Westinghouse Response:

Westinghouse has committed to the materials, accessibility for inspection, and pre-service inspections for the AP1000 reactor vessel head penetrations in DCD sections 5.2.3.1 and 5.3.4.7. Similarly, DCD section 5.2.4 includes commitments to design the AP1000 ASME Code Class 1components so that access is provided for inservice examinations specified by the ASME Code Section XI. There is no AP1000 ITAAC for this design commitment, just as no access and inspectability commitments have been included in the ITAACs of recently licensed new plant designs. Therefore, the design commitments for the reactor head penetrations included in the AP1000 DCD are sufficient to ensure the design meets the NRC requirements as given in NRC Order EA-03-009 (February 11, 2003) without the addition of new ITAACs.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 252.011

Question:

Operating experience continues to show cracking of Alloy 600 components. Recent experience appears to indicate that cracking has even occurred in welds or components not previously expected to crack based on the temperature of the weld or component and the time in service. The staff believes that the use of Alloy 690 materials in contact with the reactor coolant is a substantial improvement over the use of other materials currently in wide use in the industry. However, data is not presently available to demonstrate that cracking in these welds and components will not occur over the projected 40-year COL period of an AP1000 plant. The staff also believes that bare metal visual inspection of these locations is highly effective in identifying locations where cracking occurs.

a. Please provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure (RCP) boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions.

b. Please provide proposed ITAACs to verify that all Alloy 600/690 components and welds in the RCP boundary are identified and are readily accessible for bare metal visual inspection.

Westinghouse Response:

AP1000 DCD Section 5.2.4 contains the commitment that inservice inspection of pressure-retaining components in the reactor coolant pressure boundary are performed in accordance with Section XI of the ASME Code including addenda according to 10 CFR 50.55a(g). Also Westinghouse commits to a design for inspectability program for the AP1000. This program provides for the inspectability access and conformance of component design with available inspection equipment and techniques. The insulation for the AP1000 reactor coolant system pressure boundary is designed to be a stand-off type and/or removable to enable inspection of pressure boundary welds where required by ASME Code Section XI.

There are no Alloy 600 components in the AP1000 which are in contact with the primary reactor coolant. Alloy 690 components in the AP1000 reactor coolant pressure boundary include:

- the control rod drive, in-core instrumentation tube, and head vent penetrations through the reactor vessel upper closure head;
- passive residual heat removal (PRHR) heat exchanger piping nozzle in the steam generator channel head;
- nozzle safe ends; and
- steam generator and PRHR heat exchanger tubes.

All of these components are accessible for inspection. For example, the insulation can be removed from the PRHR nozzle for inspection and the inside of the nozzle can also be accessed from inside the steam general channel head. Also, the insulation on the top of the reactor vessel closure head is a stand-off design (minimum 3-inch clearance) which allows bare metal visual inspection of the head penetrations.

Westinghouse has included in AP1000 DCD section 5.3.4.7 design commitments relative to access to the to reactor vessel upper closure head which ensures access for visual inspection of the entire head surface. This is similar to other commitments to provide access and inspectability for reactor coolant pressure boundary welds and components as required by ASME Section XI. These access and inspectability commitments have not been included in the ITAACs of recently licensed new plant designs. Accordingly, the commitments included in the DCD are sufficient to ensure the required design features are incorporated into the AP1000 design without additional ITAACs.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.015

Question:

In the exception to RG 1.68, Appendix A, Item 4.t, Westinghouse states:

For the AP1000, natural circulation heat removal to cold conditions using the steam generators is not safety-related, as in current plants. This safety function is performed by the PRHR [passive residual heat removal]. Natural circulation heat removal via the PRHR is tested for every plant during hot functional testing. Therefore, Westinghouse has met the intent of the previous licensing commitments for natural circulation testing.

The NRC found that the exception to RG 1.68, Appendix A, Item 4.t, contradicts the low power tests in DCD test abstracts 14.2.10.3.6, "Natural Circulation (First Plant Only)" and 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger (First Plant Only)." The exception to RG 1.68 states, in part, that "the PRHR is tested for every plant during hot functional testing." The lower power test abstracts are performed on the first plant only. DCD Section 14.2.10.3.7, also states, in part, "Also note that this test is not required to be performed if a large scale test of the AP600 or AP1000 type passive residual heat removal heat exchanger has been conducted, and has provide[d] data confirming adequate heat removal capability." The NRC staff requests the following information:

(a) Westinghouse should clarify and justify the inconsistent natural circulation testing requirements in the exception to RG 1.68 and in test abstracts 14.2.10.3.6 and 14.2.10.3.7.

(b) Should natural circulation testing be performed on every plant, the first plant only, or is it not required if a large scale test facility provides data confirming the adequacy of heat removal capability?

Westinghouse Response:

(a) Westinghouse meets the intent of Regulatory Guide 1.68 Appendix A, Item 4.t with the provisions to perform the pre-operational tests of the passive RHR heat exchanger, as well as the low power tests described in DCD test abstracts 14.2.10.3.6, "Natural Circulation (First Plant Only)" and 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger (First Plant Only)." This is the same approach that was licensed for the AP600, and this approach is directly applicable to the AP1000. The exception to Regulatory Guide 1.68, Appendix A, Item 4.t will be removed in Appendix 1A as shown below.

(b) Similar to the approach licensed for the AP600, natural circulation testing need only be conducted on a First-Plant only basis. The AP1000 is a standard plant, and the design of systems, structures and components that operate to provide natural circulation cooling are

the same for all AP1000 plants. The statement that the natural circulation testing was not required if a large scale test facility provides data confirming the adequacy of heat removal capability was the approach that was licensed for AP600. This approach is also applicable to the AP1000, because the design of the AP1000 natural circulation heat removal system (i.e. the PRHR) is the same basic design as AP600.

Design Control Document (DCD) Revision:

Appendix 1A will be revised as shown:

1. Introduction and General Description of PlantAP1000 Design Control Document

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
App. A.4 ~~(except A.4.t)~~		Conforms	As applicable for pressurized water reactor.
	~~App. A.4.t~~	~~Exception~~	**Compliance with A.4.t is met** ~~F~~for the AP1000 **with the provisions to perform the pre-operational tests of the passive RHR heat exchanger, as well as the low power tests described in DCD test abstracts 14.2.10.3.6, "Natural Circulation (First Plant Only)" and 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger (First Plant Only)."** ~~; natural circulation heat removal to cold conditions using the steam generators is not safety-related, as in current plants. This safety-related function is performed by the PRHR. Natural circulation heat removal via the PRHR is tested for every plant during hot functional testing. Therefore, Westinghouse has met the intent of the previous licensing commitments for natural circulation testing.~~

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.018

Question:

In the response to RAI 261.007b, Item 5, Westinghouse states that the dynamic response of the plant to close all main steam isolation valves (MSIVs) is bounded by a plant trip from 100 percent power, which is performed in test abstract 14.2.10.4.24. MSIV testing is performed during preoperational testing to avoid an at-power transient. This is consistent with the valve in-service test requirements (DCD Table 3.9-16). Please see note 20 of DCD Table 3.9-16.

The NRC staff lacks sufficient information to conclude that the plant trip from100 percent power is bounded for the MSIV closure transient. The NRC staff requests additional information to provide the basis for the statement that the MSIV closure transient is bounded by a plant trip from 100 percent power.

Westinghouse Response:

As stated in Reg Guide 1.68, the purpose of the MSIV closure test is to demonstrate the dynamic response of the plant. Closing the MSIVs at 100% power will result in an undesirable plant transient (opening pressurizer safety valves for example), and will not test for some features of the turbine control system (steam dump, turbine overspeed) that are desired to be tested dynamically. The AP1000 Plant Trip from 100% Power test characterizes the dynamic response of the plant without inducing a severe transient, and satisfactorily characterizes the dynamic response of the plant, such as the turbine control systems.

The plant trip from 100% power test as described in Section 14.2.10.4.24 is initiated by opening the main generator breakers. Opening of the main generator breakers will cause the turbine generator to over speed. The over speed of the turbine generator results in a turbine trip and a loss of steam flow transient. The turbine stop, control and intercept valves have closing times on the order of 0.3 seconds (see DCD Table 10.2-4). The closing times of the MSIVs are approximately 5 seconds.

During the initial phase of the Plant Trip from 100% Power Test, the closing times of the turbine valves are much more rapid than those of the MSIVs. Therefore this test will result in a more rapid steam flow reduction, and the test as described in Section 14.2.10.4.24 will produce a loss of steam flow transient which bounds the steam flow transient caused by the closure of the MSIVs.

Note that Test Abstract 14.2.10.4.24 will be modified to clarify the test method.

Design Control Document (DCD) Revision:

Section 14.2.10.4.24 will be revised as follows:

Test Method

- Trip the ~~reactor~~**plant** by opening the main generator breaker.
- Monitor and record selected plant parameters.
- If necessary, adjust the control systems setpoints to obtain optimal response.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.007 (Response Revision 3)

Question:

(DCD, Tier 2, Section 6.4, 9.4. through 9.4.3 and 9.4.6 through 9.4.11) The required aspects of a control room for nuclear power reactors are documented in 10 CFR Part 50, Appendix A,"General Design Criteria for Nuclear Power Plants." GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Section 6.4.5.4 states that "[t]esting for main control room in-leakage during VES [main control room emergency habitability system] operation will be conducted once every 10 years. This testing will be conducted in accordance with ASTM [American Society for Testing and Materials] E741, 'Standard Test Method for Determining Leakage Rate by Tracer Dilution'." The staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. The staff expects that testing requirements for the AP1000 design will be consistent with the resolution of the control room habitability issues currently pursued by the industry and the staff. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry. Please provide such a commitment and revise Section 6.4.8 to add the ASTM E741 standard.

In addition, consistent with the SRP, Westinghouse should commit to complying with the guidance contained in the latest versions of RG 1.52, "Design, Testing, and Maintenance for Post-Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Westinghouse Response:

Westinghouse recognizes that the NRC staff and the industry are working on in-leakage testing, however it is not reasonable to commit to a standard that does not currently exist. Westinghouse therefore is not providing a commitment to have the Main Control Room Emergency Habitability System (VES) meet the anticipated requirements currently being pursued. The VES design addresses in-leakage and meets the codes and standards that were in effect six months prior to the date of the AP1000 design certification application (March 28, 2002).

Westinghouse is revising the DCD, subsection 6.4 to include ASTM E741.

Westinghouse is including Regulatory Guide (RG) 1.140 (Rev. 2, 06/2001) in the DCD in subsections 1A, 3.2 and 9.4. Please see the corresponding DCD revisions below.

RG 1.52, is not applicable to the AP1000 as the AP1000 has no safety-related air filtration systems.

Design Control Document (DCD) Revision:

- ***Changes to DCD 6.4:***

6.4.5.1 Preoperational Inspection and Testing

Preoperational testing of the main control room emergency habitability system is performed to verify that the air flow rate of 65 ±5 scfm is sufficient to maintain pressurization of the main control room envelope of at least 1/8-inch water gauge with respect to the adjacent areas. The positive pressure within the main control room is confirmed via the differential pressure transmitters within the control room. The installed flow meters are utilized to verify the system flow rates. The pressurization of the control room limits the ingress of radioactivity to maintain operator dose limits below regulatory limits. Air quality within the MCR environment is confirmed to be within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1 by analyzing air samples taken during the pressurization test.

The storage capacity of the compressed air storage tanks is verified to be in excess of 314,132 scf of compressed air at a minimum pressure of 3400 psig. This amount of compressed air will assure 72 hours of air supply to the main control room.

An inspection will verify that the heat loads within the rooms identified in Table 6.4-3 are less than the specified values.

Preoperational testing of the main control room isolation valves in the nuclear island nonradioactive ventilation system is performed to verify the leaktightness of the valves.

Preoperational testing for main control room inleakage during VES operation will be conducted in accordance with ASTM E741, ~~"Standard Test Method for Determining Air Leakage Rate by Tracer Dilution."~~**(Reference 4).**

Testing and inspection of the radiation monitors is discussed in Section 11.5. The other tests noted above are discussed in Chapter 14.

6.4.5.4 Air Inleakage Testing

Testing for main control room inleakage during VES operation will be conducted once every ten years. This testing will be conducted in accordance with ASTM E741, ~~"Standard Test Method for Determining Leakage Rate by Tracer Dilution."~~(Reference 4).

6.4.8 References

1. "Ventilation for Acceptable Indoor Air Quality," ASHRAE Standard 62 - 1989.
2. "Human Engineering Design Guidelines," MIL-HDBK-759C, 31 July 1995.
3. "Human Engineering," MIL-STD-1472E, 31 October 1996.
4. **"Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," ASTM E741, 2000**

- ***Changes to DCD compliance table for RG 1.140 in DCD Appendix 1A. The following replaces the existing compliance:***

APPENDIX 1A

CONFORMANCE WITH REGULATORY GUIDES

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
DIVISION 1 – Power Reactors			
Reg. Guide 1.140, Rev. 2, 06/01 - Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup System in Light-Water-Cooled Nuclear Power Plants			
C.1		Conforms	Regulatory Guide 1.140 endorses ASME Standard N509-1989 (Reference 39), ASME Standard N510-1989 (Reference 40) and ASME AG-1-1997 (Reference 38). The AP1000 uses the latest version of the industry standards (as of 3/2002).
C.2.1-2.4		Conforms	

Criteria Section Exceptions	Referenced Criteria	AP1000 Position	Clarification/Summary Description of
C.3.1-3.2		Conforms	
C.3.3	ERDA 76-21, Section 5.6; ASME N509-1989 Section 4.9	Conforms	
C.3.4	Regulatory Guide 8.8	Conforms	
C.3.5		Conforms	
C.3.6	ASME AG-1-1997 Article SA-4500	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are designed to meet at least SMACNA design standards.
	ASME AG-1-1997, Section TA	Conforms	
C.4.1	ASME AG-1-1997, Section FB	Conforms	
C.4.2	ASME AG-1-1997, Section CA	Conforms	
C.4.3	ASME AG-1-1997, Section FC, and Section TA	Conforms	
C.4.4	ASME AG-1-1997, Section FG	Conforms	
C.4.5	ERDA 76-21, Section 4.4; ASME AG-1a-2000, Section HA	Conforms	

Criteria Section Exceptions	Referenced Criteria	AP1000 Position	Clarification/Summary Description of
C.4.6	ASME N509-1989, Section 5.6; ASME AG-1a-2000, Section HA	Conforms	
C.4.7	ASME AG-1-1997, Section CA	Conforms	
C.4.8	ASME AG-1-1997, Section FD or FE	Conforms	
C.4.9	ASME AG-1-1997, Section FD and FE or, Section FF	Conforms	
C.4.10	ASME AG-1-1997 Section SA	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are designed to meet at least SMACNA design standards.
C.4.11		Conforms	
C.4.12	ASME AG-1-1997 Section DA	Conforms	
C.4.13	ASME AG-1-1997, Section BA and SA	Conforms	
C.5.1	ERDA 76-21, Section 2.3.8; ASME AG-1a-2000, Section HA	Conforms	
C.5.2		Conforms	
C.6	ASME N510-1989	Conforms	
C.7	ANSI N509-1989	Conforms	

- ***Add new reference to DCD Appendix 1A:***

1A.1 References

38. ASME AG-1-1997, "Code on Nuclear Air and Gas Treatment" 1997

- ***Changes to DCD 3.2:***

1. Changes to Subsection 3.2.6 References

18. ASME/ANSI ~~N509-89~~**AG-1-1997**, "**Code on Nuclear Air and Gas Treatment**~~Nuclear Power Plant Air Cleaning Units and Components~~."

2. Changes to Table 3.2-3 sheet 54

Table 3.2-3 (Sheet 54 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code
Nuclear Island Nonradioactive Ventilation System (VBS) (Continued)				
n/a	MCR/TSC Supplemental Air Filtration Units	Note 2	NS	ASME ~~N509~~**AG-1, Note 4**

3. *Changes to DCD Table 3.2-3 sheet 60*

Table 3.2-3 (Sheet 60 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code
Containment Air Filtration System (Continued)				
n/a	Air Exhaust Filtration Units	R	NS	ASME ~~N509~~**AG-1, Note 4**
n/a	Fans, Ductwork	L or R	NS	SMACNA or ASME ~~N509~~**AG-1, Note 4**

4. *Changes to Notes as the end of DCD Table 3.2-3*

Notes:

1. Component performs a safety-related function equivalent to AP1000 equipment Class C. The component is constructed using the standards for Class R and a quality assurance program in conformance with 10 CFR Part 50 Appendix B.
2. Component performs an AP1000 equipment Class D function and is constructed using the standards for Class L or Class R.
3. Fire dampers are constructed to the requirements of UL-555 or UL-555S if they are fire and smoke dampers and are located in Class D, Class L, and Class R ducts.
4. **Construction is non-seismic and meets applicable portions of ASME AG-1 consistent with RG 1.140.**

- ***Changes to Section 9.4***

9.4.1.1.1 Safety Design Basis

The nuclear island nonradioactive ventilation system provides the following nuclear safety-related design basis functions:

- Monitors the main control room supply air for radioactive particulate and iodine concentrations

- Isolates the HVAC penetrations in the main control room boundary on high-high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4

Those portions of the nuclear island nonradioactive ventilation system which penetrate the main control room envelope are safety-related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. Other functions of the system are nonsafety-related. HVAC equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonsafety-related and nonseismic. The equipment is procured to meet the environmental qualifications used in standard building practice.

The nuclear island nonradioactive ventilation system is designed to control the radiological habitability in the main control room within the guidelines presented in Standard Review Plan (SRP) 6.4 and NUREG 0696 (Reference 1), if the system is operable and ac power is available.

Portions of the system that provide the defense-in-depth function of filtration of main control room/technical support center air during conditions of abnormal airborne radioactivity are designed, constructed, and tested to conform with Generic Issue B-36, as described in Section 1.9 and Regulatory Guide 1.140 (Reference 30), as described in Appendix 1A, **and the applicable portions of ASME AG-1 (Reference 36)**, ASME N509 (Reference 2) and ASME N510 (Reference 3).

9.4.1.2.2 Component Description

The nuclear island nonradioactive ventilation system is comprised of the following major components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Tables 9.4.1-1, 9.4.1-2 and 9.4.1-3 provide design parameters for major components in each subsystem.

Supply Air Handling Units

Each air handling unit consists of a mixing box section, a low efficiency filter bank, high efficiency filter bank, an electric heating coil, a chilled water cooling coil bank, and supply and return/exhaust air fans.

Supply and Return/Exhaust Air Fans

The supply and return/exhaust air fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with high efficiency wheels and backward inclined

blades to produce non-overloading horsepower characteristics. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Ancillary Fans

The ancillary fans are centrifugal type with non-overloading horsepower characteristics. Each can provide a minimum of 1,530 cfm. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5), and ANSI/AMCA 300 (Reference 6).

Supplemental Air Filtration Units

Each supplemental air filtration unit includes a high efficiency filter bank, an electric heating coil, a charcoal adsorber with upstream HEPA filter bank, a downstream postfilter bank and a fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans and controls, and the location of the fans on the filtered side of units are designed, constructed, and tested to meet the **applicable** performance requirements of **ASME AG-1,** ASME N509 and ASME N510 (References **36,** 2 and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52 and 126 (References 7 and 35). Filter minimum average dust spot efficiency is shown in Table 9.4.1-1 and 9.4.1-2. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME ~~N509~~-**AG-1** (Reference ~~2~~**36**), Section ~~5.3~~**FB**. Postfilters downstream of the charcoal filters have a minimum DOP efficiency of 95 percent. The filters meet UL 900 (Reference 8) Class I construction criteria.

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with UL-586 (Reference 9) and ASME ~~N509~~-**AG-1** (Reference ~~2~~**36**), Section ~~5.1~~**FC**. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-µm aerosol **in accordance with ASME AG-1 (Reference 36), Section TA.**

Charcoal Adsorbers

Each charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME ~~N509~~-**AG-1 (Reference 36)**, Section ~~5.2~~**FE**~~; ASME 510, Sections 11, 12, and 16~~; and Regulatory Guide 1.40. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with IE Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are multi-stage fin tubular type. The electric heating coils meet the requirements of UL-1995 (Reference 10). The coils for the supplemental air filtration subsystem are constructed, qualified, and tested in accordance with ASME ~~N509~~ **AG-1** (Reference ~~23~~ 36), Section ~~5.5~~ CA.

Electric Unit Heaters

The electric unit heaters are single-stage or two-stage fin tubular type. The electric unit heaters are UL-listed and meet the requirements of UL-1996 (Reference 26) and the National Electrical Code NFPA 70 (Reference 28).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifiers

The humidifiers are packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifiers are designed and rated in accordance with ARI 620 (Reference 13).

Isolation Dampers and Valves

Nonsafety-related isolation dampers are bubble tight, single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power. The isolation dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 (Reference 14) or ASME ~~N509~~ **AG-1** (Reference ~~23~~ 36), Section ~~5.9~~ **DA**.

The main control room pressure boundary penetrations include isolation valves, interconnecting piping, and vent and test connection with manual test valves. The isolation valves are classified as Safety Class C (see subsection 3.2.2.5 and Table 3.2-3) and seismic Category I. Their boundary isolation function will be tested in accordance with ASME N510 (Reference 3).

The main control room pressure boundary isolation valves have electro-hydraulic operators. The valves are designed to fail closed in the event of loss of electrical power. The valves are qualified to shut tight against control room pressure.

Tornado Protection Dampers

The tornado protection dampers are split-wing type and designed to close automatically. The tornado protection dampers are designed against the effect of 300 mph wind.

Shutoff, Balancing and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Backdraft dampers are of the counterbalanced type and are provided to delay smoke migration through ductwork in case of fire. The backdraft dampers meet the Leakage Class II requirements of ASME N509 (Reference 2). Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements in accordance with ANSI/AMCA 500 (Reference 14). The supplemental air filtration subsystem dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 or ASME ~~N509~~ **AG-1** (Reference ~~2~~36), Section ~~5.9~~ **DA**.

Combination Fire/Smoke Dampers

Combination fire/smoke dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The combination fire/smoke dampers meet the design, leakage testing, and installation requirements of UL-555S (Reference 25).

Ductwork and Accessories

Ductwork, duct supports, and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. Ductwork, supports, and accessories meet the design and construction requirements of SMACNA Industrial Rectangular and Round Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards – Metal and Flexible (Reference 17). The supplemental air filtration and main control room/technical support center HVAC subsystem's ductwork, including the air filtration units and the portion of the ductwork located outside of the main control room envelope, that maintains integrity of the main control room/technical support center pressure boundary during conditions of abnormal airborne radioactivity are designed in accordance with ASME ~~N509~~ **AG-1** (Reference ~~2~~36), ~~Section 5.10~~ **Article SA-4500** to provide low leakage components necessary to maintain main control room/technical support center habitability.

9.4.7.2.2 Component Description

The containment air filtration system is comprised of the following components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Table 9.4.7-1 provides design parameters for the major components of the system.

Supply Air Handling Units

Each supply air handling unit consists of a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil bank, a chilled water cooling coil bank and a supply fan.

Exhaust Air Filtration Units

Each exhaust air filtration unit consists of an electric heater, an upstream high efficiency filter bank, a HEPA filter bank, a charcoal adsorber with a downstream postfilter bank, and an exhaust fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans, and controls, are designed, constructed, and tested to meet the **applicable** performance requirements of ASME **AG-1,** N509 and ASME N510 (References **3**6, 2 and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30) except as noted in Appendix 1A. The filtration unit housings maximum leakage rates do not exceed one percent of the design flow in accordance with ASME ~~N509~~**AG-1**. Refer to Table 9.4-1 for a summary of the containment air filtration system filtration efficiencies and Appendix 1A for a comparison of the containment air filtration system exhaust air filtration units with Regulatory Guide 1.140 (Reference 30).

Isolation Dampers

Isolation dampers are bubble tight, single-blade or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power or instrument air. The design and construction of the isolation dampers is in accordance with ANSI/AMCA 500 or ASME ~~N509~~-**AG-1** (References 14 and ~~2~~36).

Pressure Differential Control Dampers

Pressure differential control dampers utilize opposed-blade type construction and meet the performance requirements of ANSI/AMCA 500 (Reference 14) or ASME ~~N509~~-**AG-1** (Reference ~~2~~36), Section ~~5.9~~**DA**. The dampers maintain a slight negative pressure within the fuel handling building area, with respect to the environment and adjacent non-radiologically controlled plant areas.

Supply and Exhaust Fans

The supply and exhaust air fans are centrifugal type, single width single inlet (SWSI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. Fan performance is rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Containment Penetrations

The containment penetrations include containment isolation valves, interconnecting piping, and vent and test connections with manual test valves. The containment isolation components that maintain the integrity of the containment pressure boundary after a LOCA are classified

as Safety Class B and seismic Category I. Seismic Category I debris screens are mounted on Safety Class C, seismic Category I pipe to prevent entrainment of debris through the supply and exhaust openings that may prevent tight valve shutoff. The screens are designed to withstand post-LOCA pressures.

The containment isolation valves inside and outside the containment have air operators. The valves are designed to fail closed in the event of loss of electrical power or air pressure. The valves are controlled by the protection and plant safety monitoring system as discussed in subsection 7.1.1. The valves shut tight against the containment pressure following a design basis accident.

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. The system air ductwork inside containment meets seismic Category II criteria so that it will not fall and damage any safety-related equipment following a safe shutdown earthquake. Ductwork, supports and accessories meet the design and construction requirements of SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standard - Metal and Flexible (Reference 17). The exhaust air ductwork and supports meet the design and construction requirements of ASME ~~N509~~**AG-1** (Reference ~~23~~36), ~~Section 5.10~~**Article SA-4500**.

Shutoff and Balancing Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements of ANSI/AMCA 500 (Reference 14). The containment exhaust air dampers meet the design and construction criteria of ASME ~~N509~~**AG-1** (Reference ~~23~~36), Section ~~5.9~~**DA**.

Fire Dampers

Fire dampers are provided where the ductwork penetrates a fire barrier to maintain the fire resistance rating of the fire barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

Low and high efficiency filters are rated in accordance with ASHRAE Standard 52 and 126 (References 7 and 35). The minimum average dust spot efficiencies of the filters are shown in Table 9.4.7-1. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME ~~N509~~**AG-1** (Reference ~~23~~36), Section ~~5.3~~**FB**. Postfilters located downstream of the charcoal adsorbers have a minimum DOP efficiency of 95 percent. The filters meet UL 900 Class I construction criteria (Reference 8).

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with ASME ~~N509-~~**AG-1** (Reference ~~2~~36), Section ~~5.1~~FC. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol **in accordance with ASME AG-1, Section TA.**

Charcoal Adsorbers

Each charcoal adsorber is designed constructed, qualified, and tested in accordance with ASME ~~N509~~**AG-1 (Reference 36)**, Section ~~5.2~~FE ~~(Reference 2); ASME 510, Sections 11, 12, and 16 (Reference 3)~~; and Regulatory Guide 1.40. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with 1E Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are fin tubular type. The electric heating coils meet the requirements of UL-1995 (Reference 10). The coils are constructed, qualified and tested in accordance with ASME~~-N509~~ **AG-1** (Reference ~~2~~36), Section ~~5.5~~CA.

Heating Coils

The heating coils are hot water, finned tubular type. The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. Coils are performance rated in accordance with ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

9.4.12 Combined License Information

The Combined License applicants referencing the AP1000 certified design will implement a program to maintain compliance with **ASME AG-1 (Reference 36)**, ASME N509 (Reference 2), ASME N510 (Reference 3) and Regulatory Guide 1.140 (Reference 30) for portions of the nuclear island nonradioactive ventilation system and the containment air filtration system identified in subsection 9.4.1 and 9.4.7. The Combined License applicant will also provide a description of the MCR/TSC HVAC subsystem's recirculation mode during toxic emergencies, and how the subsystem equipment isolates and operates, as applicable, consistent with the toxic issues to be addressed by the Combined License applicant as discussed in DCD subsection 6.4.7.

- ***Add new reference 36 to DCD 9.4.13:***

9.4.13 References

36. "Code on Nuclear Air and Gas Treatment," ASME/ANSI AG-1-1997

PRA Revision:

None

NRC Additional Comments: (Revision 1)

The staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. The staff expects that testing requirements for the AP1000 design will be consistent with the resolution of the control room habitability issues currently pursued by the industry and the staff. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry.

AP600 Design Certification was based upon the ASTM E741 tracer gas dilution testing every 10 years interval after its initial testing for the control room envelope (MCRE) to determine its unfiltered inleakages. During the AP600 design Certification period, ASTM E741 tracer gas dilution testing was a first of a kind testing for the MCRE. During the period following the AP600 design Certification, the NRC staff and industry learned more about tracer gas testing and the staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. Therefore, the AP1000 design should include a commitment to resolving the inleakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry.

Westinghouse Additional Response: (Response Revision 1)

Westinghouse did not interpret that the original staff comment was limited to only the testing frequency of the control room leakage test. Since this has been clarified, and it is understood that testing will remain in accordance with ASTM E741, Westinghouse will revise the DCD as follows:

Design Control Document (DCD) Revision: (Response Revision 1)

6.4.5.4 Air Inleakage Testing

Testing for main control room inleakage during VES operation will be conducted ~~once every ten years. This testing will be conducted~~ in accordance with ASTM E741, (Reference 4).

6.4.7 Combined License Information

At the end of DCD section 6.4.7, add the following new paragraph...

The Combined License applicant will provide the testing frequency for the main control room inleakage test discussed in section 6.4.5.4.

Add the following to DCD Table 1.8-2:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
6.4-3	Main Control Room Inleakage Test Frequency	6.4.7

PRA Revision: (Response Revision 1)

None

NRC Additional Comments: (Revision 2)

Westinghouse needs to:

(a) verify that chemicals listed in SSAR Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release," to conclude that these chemicals do not represent a toxic hazard to control room operators;

(b) verify that combined license applicants are responsible for the amount and location of possible sources of toxic chemicals (as shown in SSAR Table 6.4-1, and their locations, as shown in SSAR Figure 1.2-2) in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required and assess control room protection for toxic chemicals, and for evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with RG 1.78-December 2001, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" to meet the requirements of TMI Action Plan Item IIID.3.4 and GDC 19 ;

(c) add RG 1.78-December 2001, Revision 1 reference to SSAR Section 6.4.8, "References" Since "Regulatory Guide 1.78-December 2001, Revision 1" replaces the both "Regulatory Guide 1.78-June 1974, Revision 0" and "Regulatory Guide 1.95-January 1977, Revision 1";

(d) delete reference of "Regulatory Guide 1.95" from SSAR Section 6.4.7;

(e) revise Appendix 1A to assess the conformance with RG 1.78-December 2001, Revision 1, and revise DCD Tier 2 Sections 2.2, 6.4, 9.4.1, 9.5.1, and Table 1.9-1 (Sheet 7 of 15) to correctly state the reference as "RG 1.78-December 2001, Revision 1"; and

(f) revise references list in Technical Specifications Bases B.3.7.6 to add a reference of ASHRAE Standard 62-1989.

Westinghouse Additional Response: (Response Revision 2)

The following responses correspond to the items in the "NRC Additional Comments: (Revision 2)".

(a) Westinghouse confirms that the chemicals listed in SSAR Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release."

(b) Westinghouse confirms that the combined license applicants are responsible for the amount and location of possible sources of toxic chemicals (as shown in SSAR Table 6.4-1, and their locations, as shown in SSAR Figure 1.2-2) in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required and assess control room protection for toxic chemicals, and for evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with RG 1.78-December 2001, Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" to meet the requirements of TMI Action Plan Item IIID.3.4 and GDC 19. See DCD change below. (In particular the changes to subsection 6.4.7.)

(c) Westinghouse will add RG 1.78-December 2001, Revision 1 reference to SSAR Section 6.4.8, "References." See DCD change below.

(d) Westinghouse will delete the reference of "Regulatory Guide 1.95" from SSAR Section 6.4.7. See DCD change below.

(e) Westinghouse will revise Appendix 1A to assess the conformance with RG 1.78-December 2001, Revision 1, and revise other DCD Tier 2 Sections to correctly state the reference as "RG 1.78-December 2001, Revision 1." See DCD changes below.

(f) Westinghouse will revise the references list in Technical Specifications Bases B.3.7.6 to add a reference of ASHRAE Standard 62-1989. See DCD changes below.

Design Control Document (DCD) Revision: (Response Revision 2)

Change DCD Appendix 1A as follows:

Reg. Guide 1.78, Rev. 1, 12/01~~Rev. 0, 6/74~~ - ~~Assumptions for~~ Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

C.1	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.2	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.3.1	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.3.2	**Conforms**	
C.3.3	**Exception**	**For AP1000 design certification the atmospheric dispersion factors are not calculated (since there are no specific site data) but are selected so as to bound the majority of existing sites. Section 2.3 provides additional information.**
C.3.4	**Conforms**	
C.4.1	**N/A**	**This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. It is the Combined License applicant's responsibility.**
C.4.2	**Conforms**	

C.4.3	Conforms	
C.5	N/A	Not applicable to AP1000 design certification. This is the Combined License applicant's responsibility.
~~C.1~~	~~N/A~~	~~This criterion is site specific. Therefore, this is not applicable to AP1000 design certification.~~
~~C.2~~	~~N/A~~	~~This criterion is site specific. Therefore, this is not applicable to AP1000 design certification.~~
~~C.3~~	~~Exception~~	~~In the event of a hazardous chemical spill occurring onsite during normal operation, the main control room emergency habitability system may be manually actuated from the main control room. In addition, the main control room is supplied with self-contained portable breathing equipment for operator protection.~~
		~~The Combined License applicant is responsible for the amount and location of possible sources of toxic chemicals near the plant, and toxic gas monitoring, as required. The Combined License applicant is also responsible for plant specific procedures and training in support of control room habitability.~~
~~C.4~~	~~N/A~~	~~Refer to discussion on item C.3~~
~~C.5.a~~	~~Conforms~~	
~~C.5.b~~	~~Exception~~	~~Refer to discussion on item C.6~~
~~C.6~~	~~Exception~~	~~For AP1000 design certification the atmospheric dispersion factors are not calculated (since there are no specific site data) but are selected so as to bound the majority of existing sites. Section 2.3 provides additional information.~~
~~C.7~~	~~Conforms~~	
~~C.8~~	~~Conforms~~	
~~C.9~~	~~Exception~~	~~Although the anticipated operating mode for the AP1000 in the event of a toxic gas release is for~~

~~100 percent recirculation, there is the potential for operation with a pressurized main control room using bottled air. The design pressurization is 1/8 in. water gauge.~~

~~C.10~~	~~N/A~~	
~~C.11~~	~~Conforms~~	
~~C.12~~	~~Conforms~~	
~~C.13~~	~~Conforms~~	~~Onsite toxic substances conform to these guidelines. Offsite toxic chemicals are site specific and are the Combined License applicant's responsibility.~~
~~C.14~~	~~Conforms~~	
~~C.15~~	~~N/A~~	~~Not applicable to AP1000 design certification. This is the Combined License applicant's responsibility.~~

Reg. Guide 1.95~~, Rev. 1, 1/77 – Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release~~ -Withdrawn

~~General~~	~~N/A~~	~~The AP1000 does not have onsite chlorine sources. Therefore, these guidelines are not applicable to the AP1000. Offsite chlorine sources are site specific and are the Combined License applicant's responsibility.~~

Change DCD Table 1.9-1 (Sheets 7 and 8 of 15) as follows:

	Division 1 Regulatory Guide	DCD Chapter, Section or Subsection
1.78	~~Assumptions for~~ Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Releases (~~Rev. 0, June 1974~~**Rev. 1 December 2001**)	2.2 6.4 9.4.1 9.5.1
1.95	~~Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Rev. 1, January 1977)~~**Withdrawn**	~~This regulatory guide is not applicable to AP1000 design certification.~~

Change DCD 6.4 as follows:

Next to last paragraph in 6.4.4

The protection of the operators in the main control room from offsite toxic gas releases is discussed in Section 2.2. The sources of onsite chemicals are described in Table 6.4-1 and their locations are shown on Figure 1.2-2. Analysis of these sources are in accordance with Regulatory Guide 1.78 (Reference 5) and shows that these sources do not represent a toxic hazard to control room personnel.

Revise paragraph as shown below. The revision incorporates the changes from Response Revision 1.

6.4.7 Combined License Information

Combined License applicants referencing the AP1000 certified design are responsible for the amount and location of possible sources of toxic chemicals in or near the plant and for seismic Category I Class 1E toxic gas monitoring, as required. Regulatory Guide~~s~~ 1.78 **(Reference 5)** ~~and 1.95~~ addresses control room protection for toxic chemicals, and ~~for evaluating~~ **evaluation** of offsite toxic releases (including the potential for toxic releases beyond 72 hours) ~~in accordance with the guidelines of Regulatory Guides 1.78 and 1.95~~ in order to meet the requirements of TMI Action Plan Item III.D.3.4 and GDC 19.

Combined License applicants referencing the AP1000 certified design are responsible for verifying that procedures and training for control room habitability are consistent with the intent of Generic Issue 83 (see Section 1.9).

The Combined License applicant will provide the testing frequency for the main control room inleakage test discussed in section 6.4.5.4.

Add new Reference 5 to 6.4.8:

5. **"Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release", Regulatory Guide 1.78, Revision 1, December 2001.**

Revise Technical Specifications Bases B3.7.6 as follows:

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island

Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air **(Ref. 4)** for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

REFERENCES

4. **ASHRAE Standard 62-1989, "Ventilation for Acceptable Indoor Air Quality"**

PRA Revision: (Response Revision 2)

None

Westinghouse Additional Response: (Response Revision 3)

The following is in response to Item (a) from the "NRC Additional Comments: (Revision 2)".

(a) Westinghouse commits to update DCD section 6.4.4 to state that the chemicals listed in DCD Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release."

DCD Revision: (Response Revision 3)

DCD Section 6.4.4 will be updated as shown:

The protection of the operators in the main control room from offsite toxic gas releases is discussed in Section 2.2. The sources of onsite chemicals are described in Table 6.4-1 and their locations are shown on Figure 1.2-2. Analysis of these sources are in accordance with Regulatory Guide 1.78 (Reference 5) **and the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release," (Reference 6)** and shows that these sources do not represent a toxic hazard to control room personnel.

DCD section 6.4.8

6.4.8 References

1. "Ventilation for Acceptable Indoor Air Quality," ASHRAE Standard 62 - 1989.

2. "Human Engineering Design Guidelines," MIL-HDBK-759C, 31 July 1995.

3. "Human Engineering," MIL-STD-1472E, 31 October 1996.

4. "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," ASTM E741, 2000.

5. "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Regulatory Guide 1.78, Revision 1, December 2001.

6. **"NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release," June 1979.**