Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

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

Direct tel: 412-374-4728
Direct fax: 412-374-5005
e-mail: vijukrp@westinghouse.com

Your ref: Docket No. 52-006
Our ref: DCP/NRC1690

March 26, 2004

SUBJECT: Transmittal of Revised Responses to AP1000 DSER Open Items

This letter transmits Westinghouse revised responses for Open Items in the AP1000 Design Safety Evaluation Report (DSER). A list of the revised DSER Open Item responses transmitted with this letter is Attachment 1. The non-proprietary responses are transmitted as Attachment 2.

Please contact me at 412-374-4728 if you have any questions concerning this submittal.

Very truly yours,

for.

R. P. Vijuk, Manager
Passive Plant Engineering
AP600 & AP1000 Projects

/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1690
2. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated March 26, 2004

D063

March 26, 2004

Attachment 1

List of

Non-Proprietary Responses

Table 1 **"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1690"**	
3.6.3.4-2 Addendum 2, Revision 1 6.4-1, Revision 1 15.3-1, Revision 2 15.3.6-1, Revision 1 19.1.10.2-1, Revision 2	

DCP/NRC1690
Docket No. 52-006

March 26, 2004

Attachment 2

AP1000 Design Certification Review
Draft Safety Evaluation Report Open Item Non-Proprietary Responses

DSER Open Item Number: 3.6.3.4-2 Addendum 2 Revision 1

Original RAI Number(s): None

Summary of Issue:

The assessment of the feasibility of successfully qualifying the AP1000 leak-before-break (LBB) piping that has not yet been analyzed was performed by applying correction factors to the piping analysis results for the AP600 plant, by adjusting for changes in load and pipe geometry, using flow stress based on statistical evaluation of applicable material samples, and using leak detection capability for a 0.25 gpm leak for three subsystems, the pressurizer safety subsystem, the core makeup tank supply - east subsystem, and the main steam lines inside containment.

In writing the input to the final safety evaluation report, the NRC staff has the following questions related to Open Item 3.6.3.4-2 involving the leak detection system used to take credit for LBB:

1) The wording for Section 3.4.7, "RCS Operational Leakage," of Chapter 16, "Technical Specification," indicates that unidentified leakage must not exceed 0.5 gpm. This specification is relied upon in the LBB evaluation for reactor coolant pressure boundary piping and for lines attached to the reactor coolant pressure boundary. Based on the work performed by Westinghouse to resolve open item 3.6.3.4-2, Westinghouse indicated that it may be necessary to rely on a leak detection system with a capability of detecting 0.25 gpm for the pressurizer safety subsystem and the core makeup tank supply - east subsystem. A combined license (COL) commitment should be included in the AP1000 design control document to indicate that the unidentified leakage limit will be reduced to 0.25 gpm if it is determined necessary in order to qualify these two AP1000 candidate subsystems for LBB.

2) To qualify the LBB application for main steam lines L006A and L006B, please provide additional information regarding measures to be used and the capability of these measures to detect leakage of 0.25 gpm from these two lines and additional information on how these measures will provide redundancy. In addition, the technical specifications in Chapter 16 should be revised to include a 0.50 gpm limit on unidentified leakage from the main steam lines inside containment and a COL commitment should be provided to indicate that the technical specifications will be modified to include a 0.25 gpm limit on unidentified leakage from the main steam lines inside containment if it is determined necessary in order to qualify these two AP1000 candidate lines for LBB.

NRC Additional Comment (February 18, 2004 telecon):

The leak detection capability for steam lines qualified for LBB should include redundancy and diversity.

Westinghouse Response:

(Revision 1 response) This revised response provides additional discussion concerning redundancy and diversity of steam line leakage detection. The primary means of detecting steam line leakage inside containment is through use of the containment sump level instruments.

As discussed with the NRC staff (March 18, 2004), the AP1000 Technical Specification 3.7.9 concerning containment leak detection instruments will be modified to require two containment sump level instruments to be available, instead of only one. Because these instruments are not accessible for repair during plant power operation, a third instrument will be added to provide failure tolerance.

In addition to the containment sump level sensors, the AP1000 provides 3 containment water level sensors. These sensors provide additional redundancy and diversity to the containment sump level sensors. The DCD will be modified to include the following:

- Specify that these sensors will use a different level measurement process than the containment sump level sensors
- Specify that these instruments are 1E (already in DCD Table 7.5-1)
- Specify that these instruments provide main control room indication and alarm that will notify the operators of a 0.5 gpm leak within 3.5 days

Note that the probability that the plant will have to rely on these containment water level sensors instead of the sump level sensors is extremely small because Technical Specification have been changed to require two sump level sensors to be operable. If they are not operable, then the plant has 3 days to fix them or shut down. The containment water level sensors are only important if all three sump level sensors have failed AND the operators are not aware that they have all failed. This situation is very unlikely because:

- The containment sump level instruments have continuous internal instrument fault monitoring.
- The Data Display System (plant computer) provides continuous monitoring of redundant sensors, including the sump level monitors. Failures of two or more redundant sensors that continuously give the same incorrect value are very unlikely.
- The failure of these instruments will cause the periodic transfer of water from the containment sump to the waste holdup tanks outside containment to stop. The sump pumps would have been operating frequently prior to the failure of the sump level sensors, on the order of once every 6 hours assuming 0.5 gpm leakage (or 12 hours at 0.25 gpm). As discussed in DCD section 5.2.5.3.1, the plant monitors and integrates this flow and makes it available for display in the main control room. The operators would notice a sudden change in the operation of these pumps.

In addition, the containment water level sensors provide an effective backup to the redundant containment sump level sensors by limiting the time over which the plant could operate to less than 3.5 days with a leak rate of 0.5 gpm (or 7 days at 0.25 gpm). During this time the leak rate would not change unless an SSE occurred and the chance of an SSE during such a short time is insignificant.

(Original response) This response supplements previous responses to DSER OI-3.6.3.4-2.

1) Westinghouse will add a combined license (COL) commitment in the AP1000 design control document (DCD) to indicate that the unidentified leakage limit will be reduced to 0.25 gpm if it is determined to be necessary in order to qualify LBB lines. Please refer to the "Design Control Document (DCD) Revision:" portion of this DSER Open Item Response for specific DCD changes.

2) DCD 3.6.3.3, second paragraph under the subheading "Leakage Flaw" states, "The method used to detect leakage from the main steam line inside containment is the containment sump level." Thus to qualify the main steam lines (L006A and L006B) for LBB application, Westinghouse is utilizing the redundant containment sump level sensors described in DCD 5.2.5.3.1. DCD 5.2.5.3.1 also identifies that the sump is able to detect a minimum leak of 0.03 gpm, which is below the 0.25 gpm that may be required to qualify the main steam lines for LBB.

   Westinghouse will revise COL commitment 3.6.4.2 in the AP1000 DCD to also indicate that the unidentified leakage limit will be reduced to 0.25 gpm if it is determined to be necessary in order to qualify any of the LBB lines. Please see the "Design Control Document (DCD) Revision:" portion of this DSER Open Item Response for specific DCD changes.

   Regarding the technical specifications and main steam line leakage, Westinghouse notes that Technical Specification 3.7.8 already includes a 0.50 gpm limit on leakage from the main steam lines inside containment. If the limit needs to be reduced to 0.25 gpm for LBB qualification purposes, the COL will revise the technical specifications. This will be document in the DCD as shown in the "Design Control Document (DCD) Revision:" portion of this DSER Open Item Response.

Design Control Document (DCD) Revision:

Revise DCD5.2.5.3 as follows:

5.2.5.3 Collection and Monitoring of Unidentified Leakage

Position 3 of Regulator Guide 1.45 identifies three diverse methods of detecting unidentified leakage. AP1000 use two of these three and adds a third method. To detect unidentified leakage inside containment, the following diverse methods may be utilized to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Other methods that can be employed to supplement the above methods include:

- Containment Atmosphere Pressure, Temperature, and Humidity
- Containment Water Level
- Visual Inspection

The reactor coolant system is an all-welded system, except for the connections on the pressurizer safety valves, reactor vessel head, explosively actuated fourth stage automatic depressurization system valves, pressurizer and steam generator manways, and reactor vessel head vent, which are flanged. During normal operation, variations in airborne radioactivity, containment pressure, temperature, or specific humidity above the normal level signify a possible increase in unidentified leakage rates and alert the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage. The following sections outline the methods used to collect and monitor unidentified leakage.

These methods also allow for identification of main steam line leakage inside containment. The primary method of identifying steam line leakage is redundant containment sump level monitoring. A diverse backup method is provided by containment water level monitoring. The safety-related class 1E containment water level sensors use a different measuring process than the containment sump level sensors.

5.2.5.3.1 Containment Sump Level Monitor

In conformance with position 2 of Regulatory Guide 1.45, leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump.

A leak in the primary system would result in reactor coolant flowing into the containment sump. Leakage is indicated by an increase in the sump level. The containment sump level is monitored by three~~two~~ seismic Category I level sensors~~, in accordance with p~~Position 6 of Regulatory Guide 1.45 requires two sensors. The third sensor is provided for redundancy in detecting main steam line leakage. The level sensors are powered from a safety-related Class 1E electrical source. These sensors remain functional when subjected to a safe shutdown earthquake in conformance with the guidance in Regulatory Guide 1.45. The containment sump level and sump total flow sensors located on the discharge of the sump pump are part of the liquid radwaste system.

Failure of two~~one~~ of the level sensors will still allow the calculation of a 0.5 gpm in-leakage rate within 1 hour. The data display and processing system (DDS) computes the leakage rate and the plant control system (PLS) provides an alarm in the main control room if the average change in leak rate for any given measurement period exceeds 0.5 gpm for unidentified leakage. The minimum detectable leak is 0.03 gpm. Unidentified leakage is the total leakage minus the identified leakage. The leakage rate algorithm subtracts the identified leakage directed to the sump.

To satisfy positions 2 and 5 of Regulatory Guide 1.45, the measurement interval must be long enough to permit the measurement loop to adequately detect the increase in level that would correspond to 0.5 gpm leak rate, and yet short enough to ensure that such a leak rate is detected within an hour. The measurement interval is less than or equal to 1 hour.

When the sump level increases to the high level setpoint, one of the sump pumps automatically starts to pump the accumulated liquid to the waste holdup tanks in the liquid radwaste system. The sump discharge flow is integrated and available for display in the control room, in accordance with position 7 of Regulatory Guide 1.45.

Procedures to identify the leakage source upon a change in the unidentified leakage rate into the sump include the following:

- Check for changes in containment atmosphere radiation monitor indications,
- Check for changes in containment humidity, pressure, and temperature,
- Check makeup rate to the reactor coolant system for abnormal increases,
- Perform an RCS inventory balance.
- Check for changes in water levels and other parameters in systems which could leak water into the containment, and
- Review records for maintenance operations which may have discharged water into the containment.

This procedure allows identification of main steam line leakage as well as RCS leakage.

Revise DCD 5.2.5.6 and 5.2.5.7 as follows:

5.2.5.6 Instrumentation Applications

The parameters tabulated below satisfy position 7 of Regulatory Guide 1.45 and are provided in the main control room to allow operating personnel to monitor for indications of reactor coolant pressure boundary leakage. The containment sump level, containment atmosphere radioactivity, reactor coolant system inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

Parameter	System(s)	Alarm or Indication
Containment sump level and sump total flow	WLS	Both
Reactor coolant drain tank level and drain tank total flow	WLS	Both
Containment atmosphere radioactivity	PSS	Both
Reactor coolant system inventory balance parameters	PCS, PXS, RCS, VCS, WLS	Both
Containment humidity	VUS	Indication
Containment atmospheric pressure	PCS	Both
Containment atmosphere temperature	VCS	Both

Containment water level	PXS	Both [1]
Reactor vessel head seal leak temperature	WLS	Both
Pressurizer safety relief valve leakage temperature	RCS	Both
Steam generator blowdown radiation	BDS	Both
Turbine island vent discharge radiation	TDS	Both
Component cooling water radiation	CCS	Both
Main steam line radiation	SGS	Both
Component cooling water surge tank level	CCS	Both

Note (1) The containment water level instruments provide indication and alarm for identification of a 0.5 gpm leak within 3.5 days.

5.2.5.7 Technical Specification

Limits which satisfy position 9 of Regulatory Guide 1.45 for identified and unidentified reactor coolant leakage are identified in the technical specifications, Chapter 16. LCO 3.4.7~~8~~ addresses RCS leakage limits. LCO 3.4.10 addresses main steam line leakage limits. LCO 3.4.9~~10~~ addresses leak detection instrument requirements.

Revise DCD 16.1, LCO 3.4.9 as follows:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Leakage Detection Instrumentation

LCO 3.4.9 The following RCS leakage detection instrumentation shall be OPERABLE:

a. <u>Two</u>~~One~~ containment sump level channel<u>s</u>;

b. One containment atmosphere radioactivity monitor (gaseous N13/F18).

- NOTE -

The N13/F18 containment atmosphere radioactivity monitor is only required to be OPERABLE in MODE 1 with RTP > 20%.

APPLICABILITY: MODES 1, 2, 3, and 4.

- NOTES -

1. Containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump such as by redirection to the IRWST by the containment shell gutter drains.

2. LCO 3.0.4 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<u>A.</u> <u>One required containment sump channel inoperable</u>	<u>A.1</u>	<u>Verify that the integrated sump discharge flow does not increase or decrease more than 20%</u>	<u>Once per 24 hours</u>
	<u>AND</u>		
	<u>A.2</u>	<u>Restore two containment sump channels to OPERABLE status</u>	<u>14 days</u>
<u>B.</u> Two <u>required</u>	<u>B</u>~~A~~.1	- NOTE -	

~~A.~~ containment sump channels inoperable.	Not required until 12 hours after establishment of steady state operation. ------------------------------------ Perform SR 3.4.7.1 (RCS inventory balance).	 Once per 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	AND BA.2 Restore one containment sump channel to OPERABLE status.	 72 hours
C ~~B.~~ Required containment atmosphere radioactivity monitor inoperable.	C~~B~~.1.1 Analyze grab samples of containment atmosphere. OR C~~B~~.1.2 ---------------------------------- - NOTE - Not required until 12 hours after establishment of steady state operation. ---------------------------------- Perform SR 3.4.7.1. AND C~~B~~.2 Restore containment atmosphere radioactivity monitor to OPERABLE status.	Once per 24 hours Once per 24 hours 30 days
D ~~C.~~ Required Action and associated Completion Time not met.	~~C~~C.1 Be in MODE 3. AND D~~C~~.2 Be in MODE 5.	6 hours 36 hours
E ~~D.~~ All required monitors inoperable.	E~~D~~.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.9.1	Perform a CHANNEL CHECK of required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.9.2	Perform a COT of required containment atmosphere radioactivity monitor.	92 days
SR 3.4.9.3	Perform a CHANNEL CALIBRATION of required containment sump monitor.	24 months
SR 3.4.9.4	Perform a CHANNEL CALIBRATION of required containment atmosphere radioactivity monitor.	24 months

Revise DCD 16.1, LCO 3.4.9 Basis as follows:

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Leakage Detection Instrumentation

BASES

BACKGROUND	GDC 30 of Appendix A to 10CFR50 (Ref. 1) requires means for detecting, and, to the extent practical, identifying the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting LEAKAGE detection systems. LEAKAGE detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. Industry practice has shown that water flow changes of 0.5 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE, is instrumented to alarm for increases of 0.5 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE. Note that the containment sump level instruments are also used to identify leakage from the main steam lines inside containment. Since there is not another method to identify steam line leakage in a short time frame, two sump level sensors are required to be operable. The containment water level sensors (LCO 3.3.3) provide a diverse backup method that can detect a 0.5 gpm leak with-in 3.5 days. The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity used for leak detection is the decay of N13/F18. The production of N13 and F18 is proportional to the reactor power level. N13 has a short half life and comes to equilibrium quickly. F18 has a longer half life and is the dominant source used for leak detection. Instrument sensitivities for gaseous monitoring are practical for these LEAKAGE detection systems. The Radiation Monitoring System includes monitoring N13/F18 gaseous activities to provide leak detection.
APPLICABLE SAFETY ANALYSES	The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in Chapter 15 (Ref. 3). The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS LEAKAGE detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump level monitor, in combination with an N13/F18 gaseous activity monitor, provides an acceptable minimum. Containment sump level monitoring is performed by <u>three</u>~~two~~ redundant, seismically qualified level instruments. The LCO note clarifies that if LEAKAGE is prevented from draining to the sump, its level change measurements made by OPERABLE sump level instruments will not be valid for quantifying the LEAKAGE.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS LEAKAGE detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are lower than those for MODES 1, 2, 3, and 4, the likelihood of LEAKAGE and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

Containment sump level monitoring is a valid method for detecting LEAKAGE in MODES 1, 2, 3, and 4. The containment atmosphere N13/F18 radioactivity LEAKAGE measurement during MODE 1 is valid only for reactor power > 20% RTP. RCS inventory monitoring via the pressurizer level changes is valid in MODES 1, 2, 3, and 4 only when RCS conditions are stable, i.e., temperature is constant, pressure is constant, no makeup and no letdown.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment atmosphere N13/F18 radioactivity LEAKAGE measurement during MODE 1 is not valid while containment purge occurs or within 2 hours after the end of containment purge.

The containment sump level change method of detecting leaks during MODES 1, 2, 3, and 4 is not valid during extremely cold outside ambient conditions when frost is forming on the interior of the containment vessel.

<u>ACTIONS</u>

<u>A.1 and A.2</u>

<u>With one of the two required containment sump level channel inoperable, the one remaining operable channel is sufficient for RCS leakage monitoring since the containment radiation provides a method to monitor RCS leakage. However, that is not the case for the steam line leakage monitoring. The remaining operable sump level monitor is adequate as long as it continues to operate properly. Continuing plant operation is expected to result in periodic operation of the containment sump pump. Therefore, proper operation of the one remaining sump level sensor is verified by the operators checking the integrated sump discharge flow to determine that it does not change significantly (more than +/-20%). The containment water level sensors also</u>

provide a diverse backup that can detect a 0.5 gpm leak within 3.5 days.

Restoration of two sump channels to OPERABLE status is required to regain the function in a Completion Time of 14 days after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the monitoring of the change in integrated sump discharge required by Action A.1.

~~ACTIONS~~

~~B~~A.1 and ~~A~~B.2

With two of the two required containment sump level channels inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere N13/F18 radioactivity monitor will provide indications of changes in LEAKAGE. Together with the atmosphere monitor, the periodic surveillance for RCS inventory balance, SR 3.4.7.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect LEAKAGE. A Note is added allowing that SR 3.4.7.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.
Restoration of one sump channel to OPERABLE status is required to regain the function in a Completion Time of 72 hours after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the RCS inventory balance required by Action A.1.

C~~B~~.1.1, ~~B~~C.1.2, and C~~B~~.2

With one gaseous N13/F18 containment atmosphere radioactivity-monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or RCS inventory balanced, in accordance with SR 3.4.7.1, to provide alternate periodic information.
With a sample obtained and analyzed or an RCS inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the radioactivity monitor.
The 24 hours interval for grab samples or RCS inventory balance provides periodic information that is adequate to detect LEAKAGE. A Note is added allowing that SR 3.4.7.1 is not required to be performed
until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, and makeup and letdown). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.
Required Action C~~B~~.1 and Required Action ~~B~~C.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous N13/F18 containment atmosphere radioactivity monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

D~~C~~.1 and D~~C~~.2

If a Required Action of Condition A, B or C~~B~~ cannot be met within the required Completion Time, the reactor must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, BASES

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

SR 3.4.9.1 requires the performance of a CHANNEL CHECK of the containment atmosphere N13/F18 radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and risk and is reasonable for detecting off normal conditions.

SR 3.4.9.2

SR 3.4.9.2 requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the atmosphere N13/F18 radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers risks and instrument reliability, and operating experience has shown that it is proper for detecting degradation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.9.3 and SR 3.4.9.4

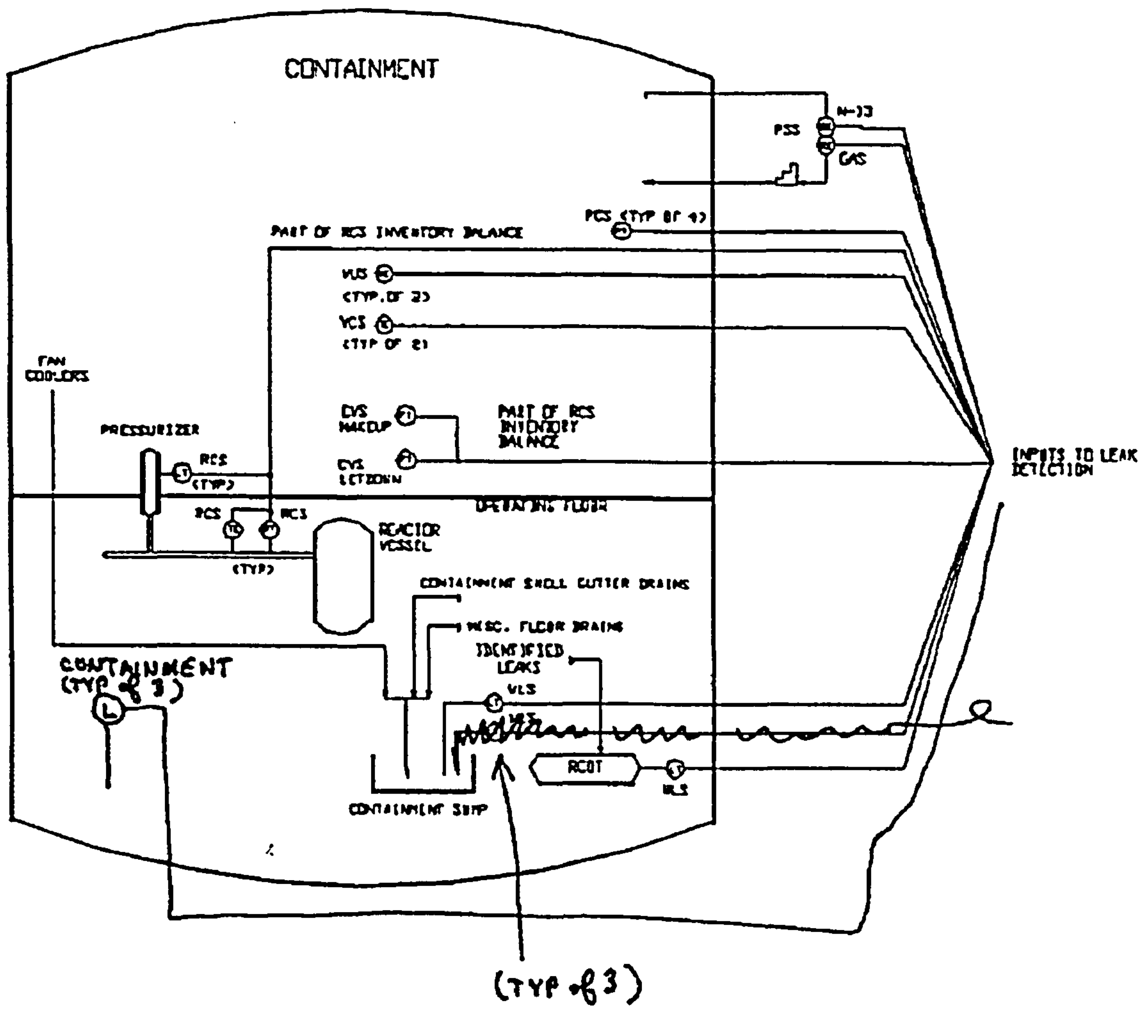
These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS Leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary LEAKAGE Detection Systems," U.S. Nuclear Regulatory Commission.
3. Chapter 15, "Accident Analysis."

DCD Figure 5.2-1, Leak Detection Approach, will be revised as follows:

<u>Add the third containment sump level sensor. Also add the 3 containment water level instruments.</u>

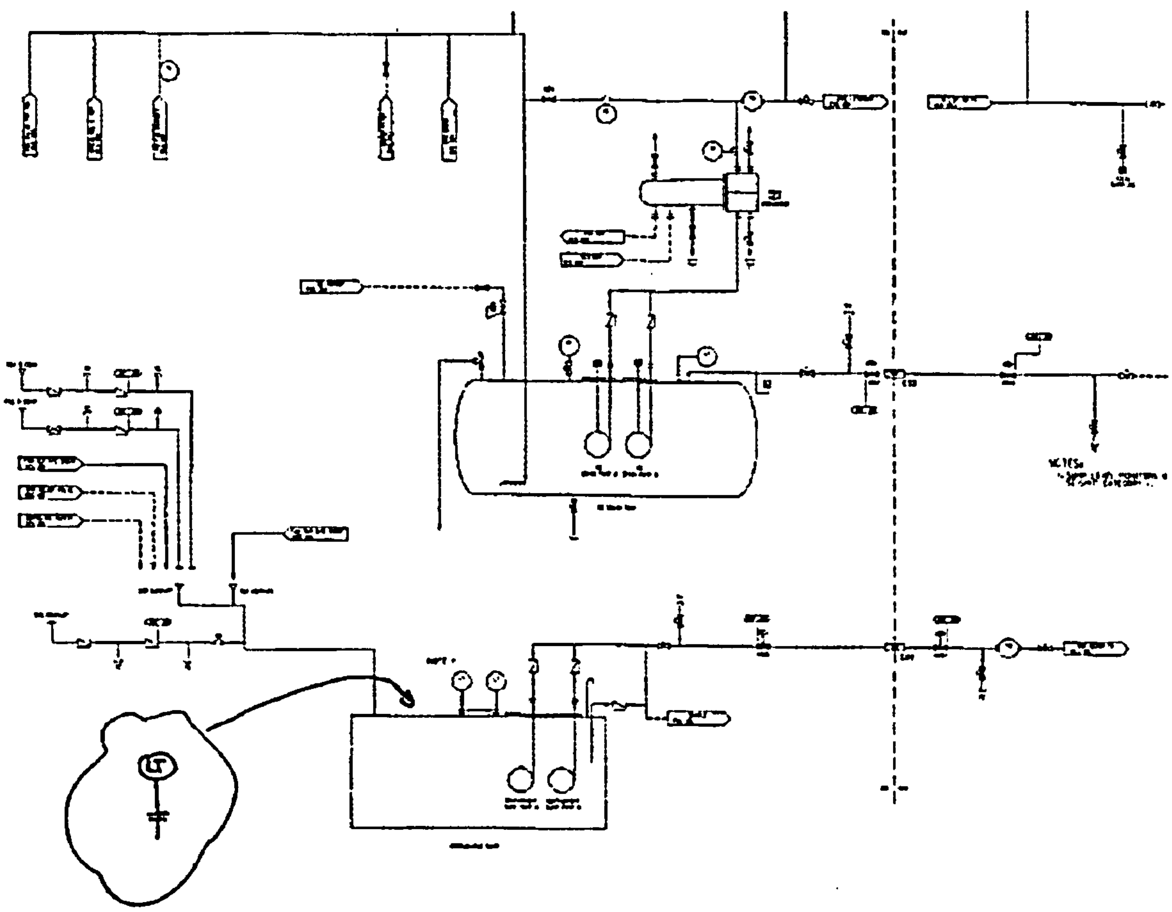


AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DCD Figure 11.2-2, Liquid Radwaste System P&ID, sheet 1 will be revised as follows:

<u>A third containment sump level instrument will be added.</u>



Revise ITAAC Table 2.3.10-1 as follows:

Add third containment sump level sensor.

Table 2.3.10-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Active Function
WLS Containment Sump Level Sensor	WLS-LT-034	No	Yes	No	No/No	No	-
WLS Containment Sump Level Sensor	WLS-LT-035	No	Yes	No	No/No	No	-
WLS Containment Sump Level Sensor	WLS-LT-036	No	Yes	No	No/No	No	-
WLS Drain from Passive Core Cooling System (PXS) Compartment A (Room 11206) Check Valve	WLS-PL-V071B	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment A (Room 11206) Check Valve	WLS-PL-V072B	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment B (Room 11207) Check Valve	WLS-PL-V071C	Yes	Yes	No	-/-	No	Transfer Closed

Revise DCD 3.6.4.2 as follows: <u><Note – this change has been incorporated in DCD Revision 9.></u>

3.6.4.2 Leak-before-Break Evaluation of as-Designed Piping

Combined License applicants referencing the AP1000 certified design will complete the leak-before-break evaluation by comparing the results of the as-designed piping stress analysis with the bounding analysis curves documented in Appendix 3B. The Combined License applicant may perform leak-before-break evaluation for a specific location and loading for cases not covered by the bounding analysis curves. <u>Successfully satisfying the bounding analysis curve limits in Appendix 3B may necessitate lowering the detection limit for unidentified leakage in containment from 0.5 gpm to 0.25 gpm. If so, the Combined License applicant shall provide a leak detection system capable of detecting a 0.25 gpm leak within one-hour and shall modify appropriate portions of the DCD including subsections 5.2.5, 3.6.3.3, 11.2.4.1, Technical Specification 3.4.7 (and Bases), Technical Specification Bases B3.4.9 and Technical Specification 3.7.8 (and Bases).</u> The leak-before-break evaluation will be documented in a leak-before-break evaluation report.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **6.4-1 Revision 1**

Original RAI Number(s): **451.006, 451.006 Rev. 1**

Summary of Issue:

The staff has not completed its review of the applicant's control room atmospheric dispersion factors (see Section 2.3.4 of this report). These factors are an input to the radiological analyses. Pending resolution of the staff's concerns with the hypothetical reference control room χ/Q values, review of the control room habitability radiological consequences analyses for design basis accidents is also incomplete as discussed in DSER Open Item 15.3-2. Therefore, the resolution of issues associated with the analysis of the dose to MCR personnel during design-basis accidents is DSER Open Item 6.4-1.

Westinghouse Response:

This item will be resolved through the resolution of DSER Open Item 2.3.4-1.

Westinghouse Response (Revision 1):

Open Item 15.3-1 Response Revision 2 includes revision to the allowable control room χ/Q values resulting from revision of the containment aerosol removal analysis. The methodology for determining control room χ/Q as discussed in Open Item 2.3.4-1 Response Revision 3 and DCD Appendix 15A remains applicable for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.3-1 Response Revision 2

Original RAI Number(s): 470.009, 470.011

Summary of Issue:

The staff has not completed its evaluation of the applicability of the AP600 aerosol removal coefficients to the AP1000 design. The staff will evaluate the impact of the differences in the AP1000 design as compared to the AP600 on the modeling of aerosol removal and will perform independent analyses of the estimated aerosol removal rates. Upon resolution of issues with the determination of aerosol removal rates in containment, as discussed in RAIs 470.009 and 470.011, the staff will complete its evaluation of the bounding accident sequence and the aerosol behavior and removal rates corresponding to the selected bounding accident sequence in the containment following a DBA. This is Open Item 15.3-1.

Westinghouse Response:

The Westinghouse responses to RAI 470.009 transmitted by Westinghouse letter DCP/NRC1535, November 26, 2002 and RAI 470.011 Rev. 1 transmitted by Westinghouse letter DCP/NRC1571, April 11, 2003 address previous NRC comments related to this issue.

NRC Additional Comments (Nov 6, 2003 telecon):

a) Clarify the use of shape factor described in section 15B.2.1.1 of the DCD.

b) Discuss the sensitivity of aerosol removal to aerosol void fraction identified in section 15B.4.2.3.

Westinghouse Response to NRC Additional Comments (Nov 6, 2003 telecon):

a) Section 15B.2.1.1 and 15B.3 of the DCD will be revised as shown below.

b) Section 15B.2.4.3 of the DCD will be revised as shown below.

NRC Additional Comments (March 10, 15 and 16, 2004 telecons):

a) Provide additional justification for aerosol removal by thermophoresis. In particular, address how the heat transfer rate from the air to the containment wall is calculated and applied in the determination of aerosol removal by thermophoresis.

b) The particle density fraction (0.8) and void content (water) used by Westinghouse are not sufficiently conservative; NRC would agree with particle density fraction of 0.6 and void content of air.

c) The core inventory of Iodine and Cesium used in the AP1000 analysis seems high compared to these values for cores that NRC has reviewed for operating plants.

Westinghouse Response to NRC Additional Comments (*March 10, 15 and 16, 2004 telecons*):

a) The previous aerosol removal analysis used an incorrect input for the heat transfer from the containment air to the containment shell. The STARNAUA analysis has now been performed using correct input for this heat transfer that drives the thermophoresis removal mechanism. The following discussion provides the results of the new STARNAUA analysis and the results of new dose calculations using the aerosol removal rates from the new STARNAUA analysis.

Definition of thermophoresis

If a temperature gradient exists in an air volume, a particle in that volume tends to migrate towards the cooler region. The motion is the result of gas molecules on the warm side striking the particle with a greater average momentum than those on the cooler side. This phenomena is defined as thermophoresis. Thermophoresis will exist when there is a temperature gradient in the gas regardless of whether the gradient is caused by conduction or by natural and/or forced convection.

Determination of temperature gradient

Boundary layer theory for convective flow and heat transfer in a gas at a solid surface indicates that the heat transfer rate is ultimately determined by the thermal conduction in the sublayer at the heat transfer surface, which is given by:

$$q = -k_{air} \cdot A \cdot \frac{dT}{dy}$$

where q is the heat transfer rate, k_{air} is the thermal conductivity of air and dT/dy is the temperature gradient at the heat transfer surface (i.e., y=0). However, in engineering applications, convective heat transfer problems are solved not by the equation above, but by:

$$q = h \cdot A \cdot (T_a - T_s)$$

where h is the heat transfer coefficient, T_a is the ambient temperature and T_s is the temperature of heat transfer surface. The reason is that dT/dy at the surface is unknown and is hard to determine in tests. On the other hand, the unknown h can be calculated easily by many empirical or semi-empirical correlations. Once q is calculated using second equation above, it can be substituted into the first equation to calculate the temperature gradient at the wall, i.e.,

$$\left|\frac{dT}{dy}\right| = \frac{q}{k_{air} \cdot A}$$

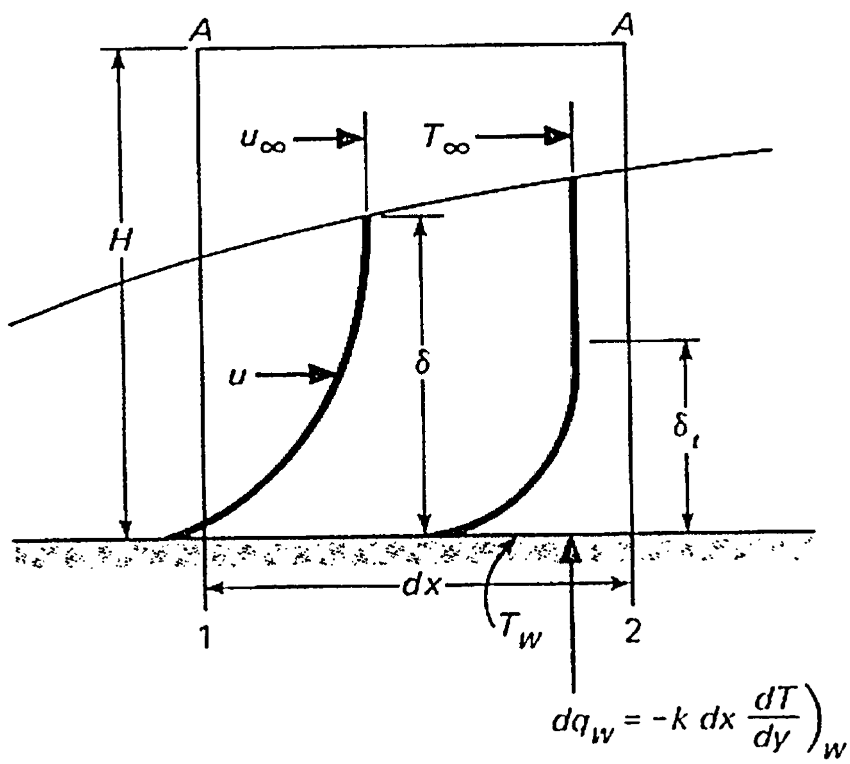
Detailed development of this relationship can be found in heat transfer textbooks (e.g., Heat Transfer, J.P. Holman, 4th edition, 1976, McGraw Hill).

Value for a realistic temperature gradient

A high temperature gradient at the wall is not unreasonable even for natural convection situations; on pages 248-250 of the Holman textbook cited above is an example for natural convection of air on a heated vertical surface. The heat flux in this example is 800 W/m^2 and the conductivity of air is given as 0.032 W/m°C. This results in a temperature gradient at the wall of 25 °C/mm.

As shown below the natural convection and radiative heat transfer for the AP1000 severe accident scenario is this same order of magnitude (few hundred W/ m^2), so the temperature gradient is also the same order of magnitude as in the textbook example.

Th figure below from the Holman textbook illustrates the thermal and momentum boundary layers.



Calculation of heat transfer from containment air to the wall

The convective and radiative heat transfer to the containment shell was determined using the results from the MAAP4 simulation of sequence 3BE-01. The containment gas temperature and the film temperature on the wall are shown in Figure 1 for this sequence. Figures 2 and 3 shows the temperature difference between the gas and the film which is the driving mechanism for heat transfer. For the first 10 hours, the temperature difference is approximately 60C, and for times after 10 hours, the temperature difference is approximately 50C.To determine the gas properties, the following temperatures are used:

Tg = Bulk Containment Gas Temperature = 110 C

Tw = Average film temperature containment wall = 55 C

For AP1000, the heat transfer area of the containment shell is

Aw = Containment shell surface area = 5847 m^2

Assuming these temperatures the properties for air are used:

kg = Gas thermal conductivity = 0.0305 W/m-K

vg = Gas kinematic viscosity = 2.51E-5 m^2/s

Pr = Gas Prandtl Number = 0.8625

ε is the effective emissivity (assumed to be 0.8 for the liquid film)

The McAdams correlation is used to determine the free convection heat transfer coefficient (Ref. 1)

$$h_{conv} = 0.13 * kg * [\, g * (Tg - Tw)\, Pr / vg^2\,]^{1/3}$$

Heat is also transferred between the particles suspended in the containment atmosphere and the film on the inside of the containment shell. The particles are assumed to be in thermal equilibrium with the containment atmosphere at Tg, and form an opaque, isothermal hemisphere as viewed from the wall. Thus, the radiation form factor between the shell and the particles is assumed to be 1.0, and the emissivity of the hemisphere of particles is assumed to be 1.0. The effective heat transfer coefficient for radiation is determined by linearizing the equation for radiation heat transfer

$$Qrad = \varepsilon * \sigma * Aw * (Tg^4 - Tw^4)$$

$$= \varepsilon * \sigma * Aw * (Tg^2 - Tw^2) * (Tg^2 + Tw^2)$$

$= \varepsilon * \sigma * Aw * (Tg^2 + Tw^2) * (Tg + Tw) * (Tg - Tw)$

$= h_{rad} * Aw * (Tg + Tw)$

where h_{rad} is the effective heat transfer coefficient for radiation

$h_{rad} = \varepsilon * \sigma * (Tg^2 + Tw^2) * (Tg + Tw)$

where

σ is the Stefan-Boltzmann constant for radiation heat transfer

ε is the effective emissivity (assumed to be 0.8 for the liquid film)

and the temperatures are absolute (K)

Using the values above,

$h_{conv} = 4.39\ W/m^2\text{-}K$

and

$h_{rad} = 8.75\ W/m^2\text{-}K$

These heat transfer coefficients are used to determine the heat transfer due to free convection and radiation for two time frames; 0-10 hours and 10+ hours.

<u>0-10 hours</u>

$Q_{conv} = h_{conv} * Aw * \Delta T = 4.39\ W/m^2\text{-}K * 5847\ m^2 * 60K = 1.54\ MW$

and

$Q_{rad} = h_{rad} * Aw * \Delta T = 8.75\ W/m^2\text{-}K * 5847\ m^2 * 60K = 3.07\ MW$

The heat fluxes for each mechanism are

$q''_{conv} = Q_{conv} / Aw = 263\ W/m^2$

and

$q''_{rad} = Q_{rad} / Aw = 525\ W/m^2$

<u>10+ hours</u>

$Q_{conv} = h_{conv} * Aw * \Delta T = 4.39\ W/m^2\text{-}K * 5847\ m^2 * 50K = 1.28\ MW$

and

$Q_{rad} = h_{rad} * Aw * \Delta T = 8.75\ W/m^2\text{-}K * 5847\ m^2 * 50K = 2.56\ MW$

The heat fluxes for each mechanism are

$q"_{conv} = Q_{conv} / Aw = 219\ W/m^2$

and

$q"_{rad} = Q_{rad} / Aw = 438\ W/m^2$

References

1. Holman, J.P., Heat Transfer, McGraw Hill, 4th Ed, 1976.

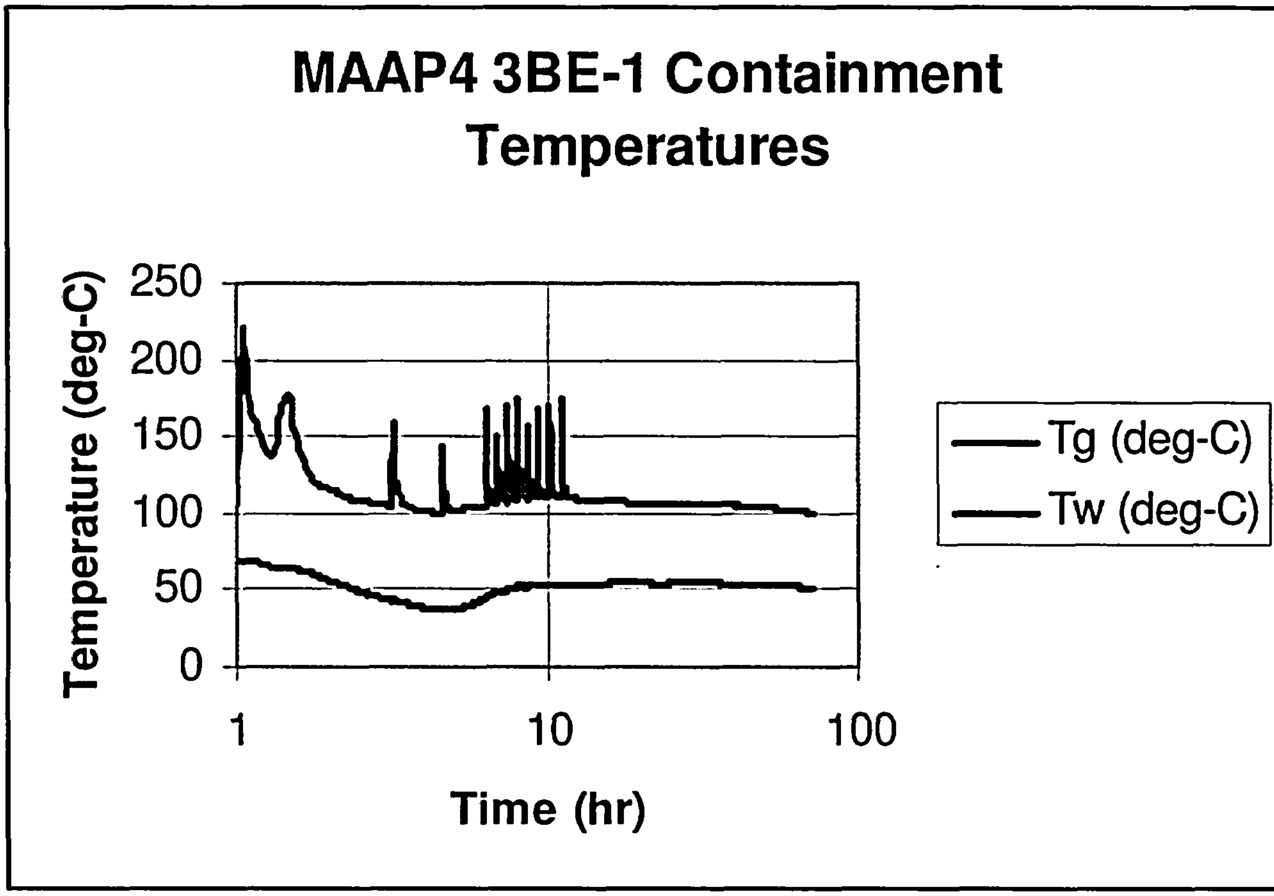


Figure 1: MAAP4 Containment Temperature for Sequence 3BE-1

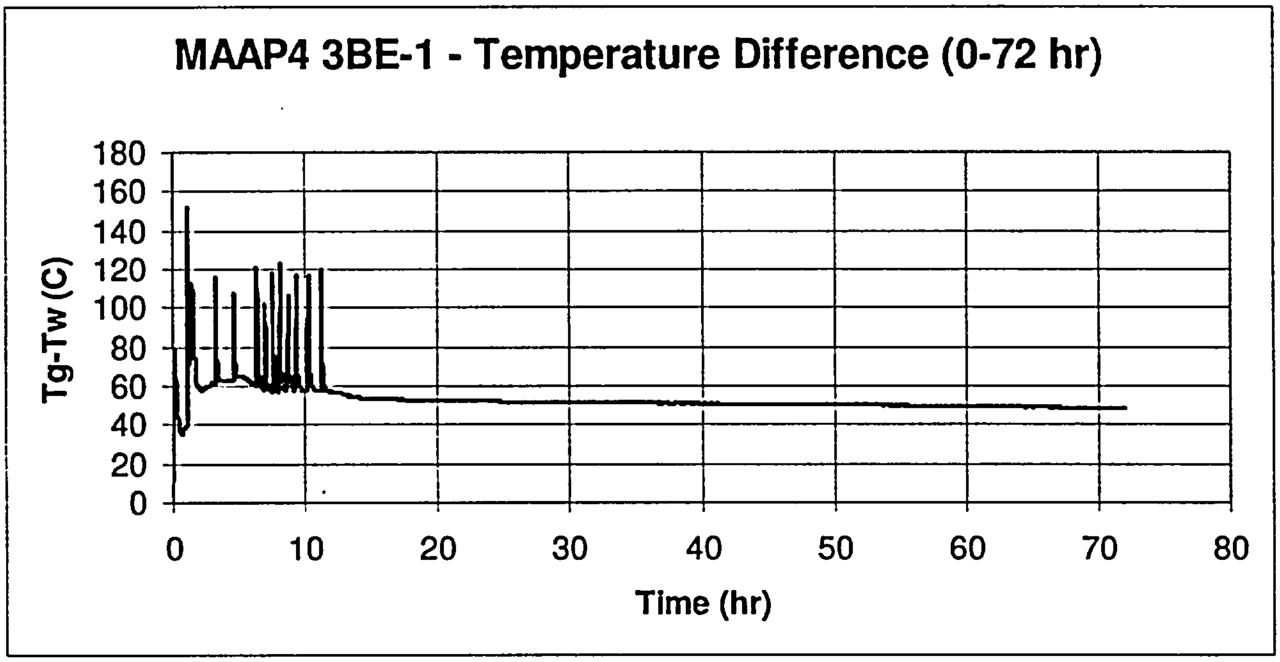


Figure 2: MAAP4 Temperature Difference Between Containment Atmosphere and Liquid Film at the Wall (0 – 72 hours)

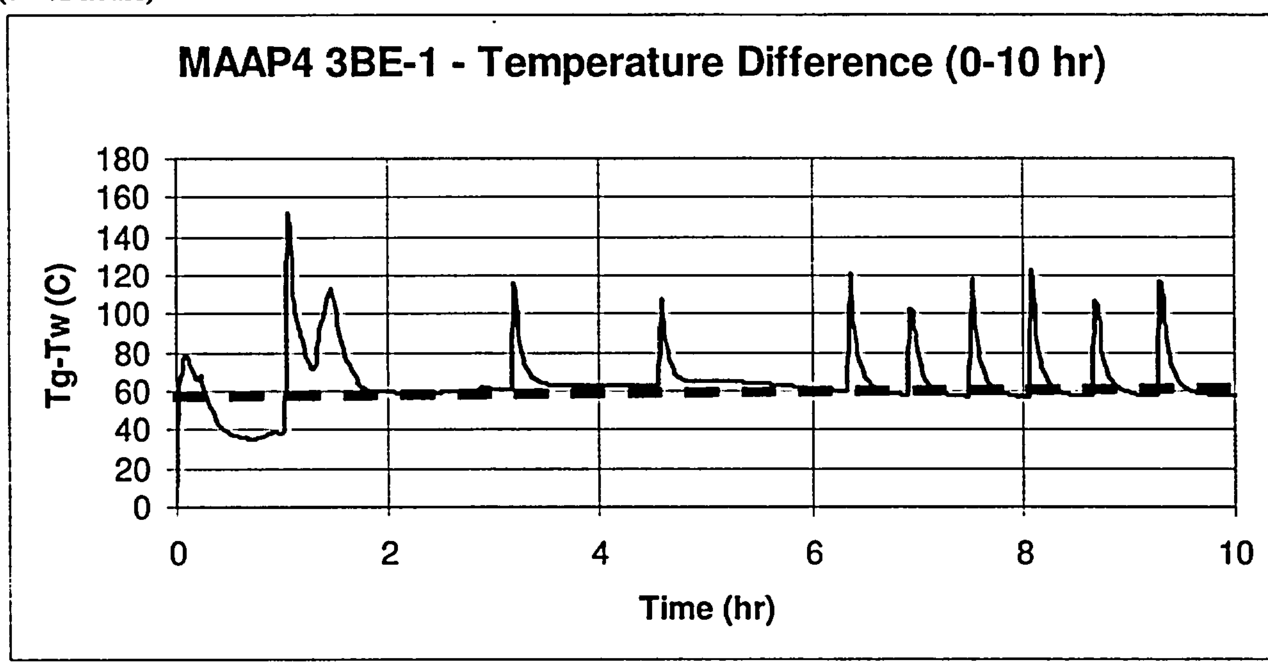


Figure 3: MAAP4 Temperature Difference Between Containment Atmosphere and Liquid Film at the Wall (0 – 10 hours)

The combined convective and radiative heat flux is used in the STARNAUA analysis determination of thermophoretic aerosol removal. The heat transfer rate is converted to heat flux by dividing by the containment wall area. The temperature gradient in the air near the containment shell is calculated by dividing the heat flux by conductivity of the air. This is a direct determination of the temperature gradient and can be quite large even for modest heat transfer coefficients and temperature differences, as shown above.

This temperature gradient is used in the Thermophoresis correlation to get thermophoretic deposition velocity. STARNAUA applies this deposition velocity to determine the rate at which aerosol particles are deposited on the water film surface at the wall by multiplying the thermophoretic deposition velocity by the particle concentration and wall surface area. The particle concentration used here is based on a well mixed containment atmosphere. Mixing in the bulk containment is driven by the convective forces resulting from steaming into containment and heat transfer at the boundaries of the containment atmosphere. Even with a high thermophoretic deposition of particles near the wall, the particles in the bulk space will travel fast enough to the region near the wall to maintain a continuous deposition. One can easily derive that the thermophoretic velocity of particles is on the order of a fraction of cm per second. It is very small when compared to particle movement in the bulk space, which is on the order of meters per second or higher. Therefore, thermophoresis will not cause a particle free zone near the surface.

Evaluation of Mixing and Stratification in the AP1000 Containment

As part of the AP600 Design Certification process, a test facility was constructed to characterize the passive containment cooling system. The Large Scale Test facility (Ref: M.D. Kennedy, et al, "Westinghouse-GOTHIC Comparisons with Passive Containment Cooling Tests Using a One-to-Ten Scale Test Facility", Nuclear Technology, Vol. 113, January 1996) was constructed to test a range of containment designs from the Heavy Water Reactor Facility which was a 1:10 scale, to the AP600 which was a 1:8 scale. The vessel was designed with a prototypic height to diameter ratio. The facility was equipped with a water film distribution system on the outside of the shell, and a steam injection system to simulate the mass and energy releases during a large pipe break inside containment. Several tests were performed including steady-state tests to determine the heat and mass transfer characteristics inside and outside the containment shell, transient simulations to determine the containment pressure response to simulated releases, and releases of non-condensable gas along with the steam to determine the degree of mixing and stratification inside the containment.

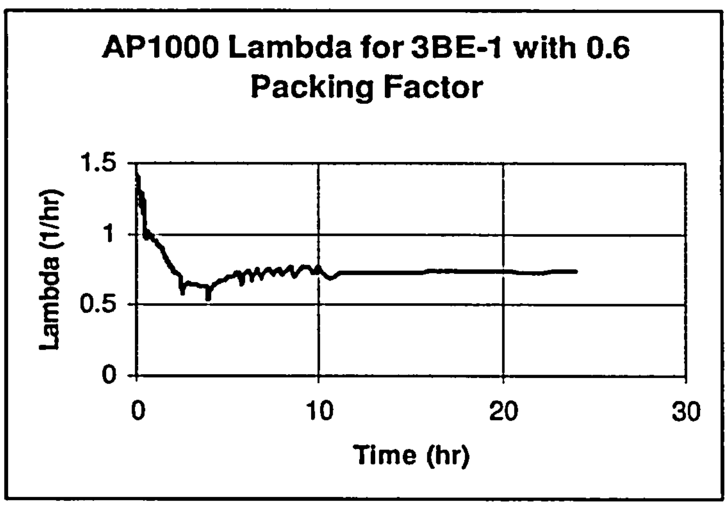
For the steam-only tests, it was determined that the volume above the operating deck was typically well mixed with somewhat higher temperature above the steam release point along the centerline of the vessel, and lower temperatures along the walls. Flow patterns were observed to be upflow along the centerline and downflow along the walls. The volumes below the operating deck were stagnant, air-rich, and generally much colder than the volume above the deck. Gas velocities were found to be related to the velocity and orientation of the steam jet, but were generally found to be on the order of ~1 m/s.

For the non-condensable gas injection tests, helium was used to simulate hydrogen that would be evolved during a severe accident. For these tests, helium was injected along with the steam jet. Gas samples were taken at various points in the containment. For all these tests, the gases were found to be well-mixed above the operating deck. Once again, the volume below the operating deck was air-rich and colder. Very little helium was observed in this volume.

It is expected that any aerosol particles suspended in the containment atmosphere would follow the flow patterns observed in these tests, at similar velocities. Since the bulk flow velocity is much higher than the thermophoretic velocity which is on the order of fractions of cm/sec, the containment will be well-mixed with regard to the distribution of aerosol particles. The results of these tests are applicable to the AP1000 since the scaled parameters and test conditions cover the ranges expected for the AP1000 containment under accident conditions. These tests form the basis of the WGOTHIC and MAAP4 code validation for AP1000 containment analysis.

Revised STARNAUA Analysis Results

The overall aerosol removal coefficients~~rate~~ (lambda) calculated by STARNAUA are~~is~~ shown in the figure below. These~~is~~ removal coefficients are~~rate is~~ used in the LOCA dose analysies to determine allowable atmospheric dispersion factors (χ/Q) for offsite and control room doses to remain below the acceptance criteria.



Revised Dose Analyses

The radiological consequences of the LOCA have been recalculated taking into account the revised aerosol removal coefficients. In order to continue to obtain doses that are within the dose acceptance limits, the atmospheric dispersion factors have been redefined for the Site

Boundary, the Low Population Zone outer boundary, and the Control Room. These revised atmospheric dispersion factors are provided below:

Site Boundary	6.5E-4 sec/m^3	(8.0E-4 was the previous value)

Low Population Zone

0 – 8 hours	3.5E-4 sec/m^3	(5.0E-4 was the previous value)
8 – 24 hours	2.5E-4 sec/m^3	(3.0E-4 was the previous value)
24 – 96 hours	1.0E-4 sec/m^3	(1.5E-4 was the previous value)
96 – 720 hours	8.0E-5 sec/m^3	(this value was not changed)

Control Room

Atmospheric Dispersion Factors at HVAC Intake (sec/m^3)				
	Plant Vent or PCS Air Diffuser as Release Point		Ground Level Containment Release Points	
	New Value	Old Value	New Value	Old Value
0 -2 hr	2.45E-3	2.5E-3	2.45E-3	2.5E-3
2-8 hr	1.65E-3	1.7E-3	1.65E-3	1.7E-3
8-24 hr	6.6E-4	1.0E-3	6.6E-4	1.0E-3
24-96 hr	5.0E-4	8.0E-4	5.0E-4	8.0E-4
96-720 hr	4.0E-4	7.0E-4	4.0E-4	8.0E-4

Atmospheric Dispersion Factors at Control Room Door (sec/m^3)				
	Plant Vent or PCS Air Diffuser as Release Point		Ground Level Containment Release Points	
	New Value	Old Value	New Value	Old Value
0 -2 hr	1.0E-3	1.0E-3	1.0E-3	1.5E-3
2-8 hr	7.0E-4	8.0E-4	7.0E-4	8.0E-4
8-24 hr	3.5E-4	4.0E-4	3.5E-4	4.0E-4
24-96 hr	3.0E-4	3.0E-4	3.0E-4	4.0E-4
96-720 hr	2.5E-4	2.5E-4	3.0E-4	4.0E-4

The LOCA doses resulting from leakage of activity from the containment are recalculated to be:

Site Boundary	24.4 rem TEDE
Low Population Zone	23.5 rem TEDE
Control Room	4.5 rem TEDE (with Emergency Habitability System in service)
Control Room	4.8 rem TED (with HVAC in service)

~~Insert dose results~~

b) Westinghouse believes that the use of a particle density fraction of 0.8 is consistent with empirical data. While the 0.8 particle density fraction together with the assumption that the aerosol void are water-filled are believed to be appropriate for the AP1000 post-LOCA containment environment, this analysis of aerosol removal has been performed using a reduced particle density fraction of 0.6 combined with the assumption that the voids are air-filled.

c) Core nuclide inventories vary with both power and burnup. For nuclides with relatively short half-lives (e.g., I-131 and I-133), the inventory in the core is dependent primarily on power level with core burnup having little impact. However, for nuclides that have long half-lives (e.g., I-129, Cs-134, and Cs-137) or are stable (e.g., I-127), both the core power level and core burnup will strongly affect the nuclide inventory in the core. The AP1000 power level is comparable to currently operating Westinghouse four-loop plants and is designed to operate with an 18-month fuel cycle. If this is compared with a Westinghouse three-loop plant operating with an annual fuel cycle, the short-lived nuclides will be found to be roughly proportional to power level but the long-lived and stable nuclides will be significantly greater for the AP1000 because of the longer operating time over which these nuclides are created.

Design Control Document (DCD) Revision:

Revisions will be made to Tier 1 Table 5.0-1, Tier 2 Section 2.3.4, Table 2-1, Section 15.6.5.3.5, Section 15.6.5.3.8, Table 15.6.5-3, Table 15A-5, Table 15A-6, and Appendix 15B, as shown on the following pages.

Table 5.0-1 (cont.)
Site Parameters

Soil	
Average Allowable Static Soil Bearing Capacity	Greater than or equal to 8,600 lb/ft^2 over the footprint of the nuclear island at its excavation depth
Maximum Allowable Dynamic Bearing Capacity for Normal Plus Safe Shutdown Earthquake (SSE)	Greater than or equal to 120,000 lb/ft^2 at the edge of the nuclear island at its excavation depth
Shear Wave Velocity	Greater than or equal to 8000 ft/sec based on low-strain, best-estimate soil properties over the footprint of the nuclear island at its excavation depth
Liquefaction Potential	None
Seismic	
SSE	SSE free field peak ground acceleration of 0.30 g at foundation level of nuclear island with modified Regulatory Guide 1.60 response spectra (See Figures 5.0-1 and 5.0-2.)
Fault Displacement Potential	None
Atmospheric Dispersion Factors (X/Q)	
Site Boundary (0-2 hr)	≤ 6.5~~8.0~~ x 10^{-4} sec/m^3
Site Boundary (annual average)	≤ 2.0 x 10^{-5} sec/m^3
Low Population Zone Boundary	
0 - 8 hr	≤ 3.5~~5.0~~ x 10^{-4} sec/m^3
8 - 24 hr	≤ 2.5~~3.0~~ x 10^{-4} sec/m^3
24 - 96 hr	≤ 1.0~~1.5~~ x 10^{-4} sec/m^3
96 - 720 hr	≤ 8.0 x 10^{-5} sec/m^3

Table 5.0-1 (cont.)
Site Parameters

Control Room Atmospheric Dispersion Factors (χ /Q) for Accident Dose Analysis

	Plant Vent or PCS Air Diffuser(3)	Ground Level Containment Release Points(4)	PORV and Safety Valve Releases(5)	Steam Line Break Releases	Fuel Handling Area(6)
χ/Q (s/m^3) at HVAC Intake for the Identified Release Points(1)					
0 - 2 hours	2.45E-3	2.45E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	1.65~~7~~E-3	1.65~~7~~E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	6.6~~1.0~~E-4~~3~~	6.6~~1.0~~E-3	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	5~~8~~.0E-4	5~~8~~.0E-4	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	4~~7~~.0E-4	4~~8~~.0E-4	4.5E-3	5.0E-3	1.0E-3
χ/Q (s/m^3) at Control Room Door for the Identified Release Points(2)					
0 - 2 hours	1.0E-3	1.0~~5~~E-3	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	7~~8~~.0E-4	7~~8~~.0E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	3.5~~4.0~~E-4	3.5~~4.0~~E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	3.0E-4	3~~4~~.0E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	2.5E-4	3~~4~~.0E-4	8.0E-4	8.0E-4	1.0E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the nonsafety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.
4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.
5. The listed values bound the dispersion factors for releases from the steam line safety and power-operated relief valves, and the condenser air removal stack. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and the secondary side release from a rod ejection accident. Additionally, these dispersion coefficients are conservative for the small line break outside containment.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases

associated with spent fuel pool boiling.

2.3.4 Short-Term Diffusion Estimates

In the absence of a specific site for use in determining values for short-term diffusion, a study was performed to determine the atmospheric dispersion factors (χ/Q values) that would envelope most current plant sites and that could be used to calculate the radiological consequences of design basis accidents. The χ/Q values thus derived for offsite are provided in Table 2-1.

This set of offsite χ/Q values is representative of potential sites for construction of the AP1000. The values are appropriate for analyses to determine the radiological consequences of accidents. These values were selected to bound ~~80 to 90 percent of~~most U.S. sites.

The χ/Q values for the control room air intake or the door leading to the control room are dependent not only on the site meteorology but also on the plant design and layout. These χ/Q values are addressed in Appendix 15A. Separate sets of χ/Q values are identified for each combination of activity release location and receptor location.

Table 2-1 (Sheet 2 of 3)

SITE PARAMETERS

Plant Grade Elevation	Less than plant elevation 100′ except for portion at a higher elevation adjacent to the annex building
Precipitation	
Rain	19.4 in./hr (6.3 in./5 min)
Snow/Ice	75 pounds per square foot on ground with exposure factor of 1.0 and importance factors of 1.2 (safety) and 1.0 (non-safety)
Atmospheric Dispersion Values - $\chi/Q^{(e)}$	
Site boundary (0-2 hr)	≤ 6.5~~8.0~~ x 10^{-4} sec/m^3
Site boundary (annual average)	≤ 2.0 x 10^{-5} sec/m^3
Low population zone boundary	
0 - 8 hr	≤ 3.5~~5.0~~ x 10^{-4} sec/m^3
8 - 24 hr	≤ 2.5~~3.0~~ x 10^{-4} sec/m^3
24 - 96 hr	≤ 1.0~~5~~ x 10^{-4} sec/m^3
96 - 720 hr	≤ 8.0 x 10^{-5} sec/m^3
Population Distribution	
Exclusion area (site)	0.5 mi

Notes:

(a) Maximum and minimum safety values are based on historical data and exclude peaks of less than 2 hours duration.

(b) Maximum and minimum normal values are the 1 percent exceedance magnitudes.

(c) With ground response spectra (at foundation level of nuclear island) as given in Figures 3.7.1-1 and 3.7.1-2.

(d) The noncoincident wet bulb temperature is applicable to the cooling tower only.

(e) For AP1000, the terms "site boundary" and "exclusion area boundary" are used interchangeably. Thus, the χ/Q specified for the site boundary applies whenever a discussion refers to the exclusion area boundary.

Table 2-1 (Sheet 3 of 3)

SITE PARAMETERS

Control Room Atmospheric Dispersion Factors (χ /Q) for Accident Dose Analysis					
χ/Q (s/m³) at HVAC Intake for the Identified Release Points(1)					
	Plant Vent or PCS Air Diffuser(3)	**Ground Level Containment Release Points(4)**	**PORV and Safety Valve Releases(5)**	**Steam Line Break Releases**	**Fuel Handling Area(6)**
0 - 2 hours	2.45E-3	2.45E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	1.65~~7~~E-3	1.65~~7~~E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	6.6~~1.0~~E-4~~3~~	6.6~~1.0~~E-4~~3~~	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	5~~8~~.0E-4	5~~8~~.0E-4	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	4~~7~~.0E-4	4~~8~~.0E-4	4.5E-3	5.0E-3	1.0E-3
χ/Q (s/m³) at Control Room Door for the Identified Release Points(2)					
	Plant Vent or PCS Air Diffuser(3)	**Ground Level Containment Release Points(4)**	**PORV and Safety Valve Releases(5)**	**Steam Line Break Releases**	**Fuel Handling Area(6)**
0 - 2 hours	1.0E-3	1.05E-3	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	7~~8~~.0E-4	7~~8~~.0E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	3.5~~4.0~~E-4	3.5~~4.0~~E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	3.0E-4	3~~4~~.0E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	2.5E-4	3~~4~~.0E-4	8.0E-4	8.0E-4	1.0E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.

4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.

5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves and the condenser air removal stack. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident. Additionally, these dispersion coefficients are conservative for the small line break outside containment.

6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches that may be used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when high iodine activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply or to bring the normal control room HVAC into operation with the supplemental filtration train.

~~The second approach, with the emergency habitability system in use, results in the more conservative determination of doses for this event.~~

The main control room is accessed by a vestibule entrance which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The equivalent inflow of unfiltered air due to expected ingress/egress has been determined to be 5.0 cfm.

Activity entering the main control room is assumed to be uniformly dispersed. No credit is taken for the removal of airborne activity in the main control room although elemental iodine and particulates would be removed by deposition and sedimentation.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.
The reported exclusion area boundary doses are for the time period of 1.0~~0.8~~ to 3.0~~2.8~~ hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 8 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of the three dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

Table 15.6.5-3

RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT WITH CORE MELT

	TEDE Dose (rem)
Exclusion zone boundary dose (1.0~~0.8~~ – 3.0~~2.8~~ hr)[1]	24.4~~2~~
Low population zone boundary dose (0 - 30 days)	23.5~~2~~
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	4.5~~6~~ rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	4.6~~7~~6 rem
Main control room dose (normal HVAC operating in the supplemental filtration mode)	
– Airborne activity entering the main control room	4.8 rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	4.96 rem

Note:

1. This is the 2-hour period having the highest dose.

Table 15A-5 OFFSITE ATMOSPHERIC DISPERSION FACTORS (X/Q) FOR ACCIDENT DOSE ANALYSIS	
Site boundary χ/Q (s/m^3) 0 – 2 hours$^{(1)}$	6.5~~8.0~~x10^{-4}
Low population zone χ/Q (s/m^3) 0 – 8 hours 8 – 24 hours 24 – 96 hours 96 – 720 hours	 3.5~~5.0~~x10^{-4} 2.5~~3.0~~x10^{-4} 1.0~~5~~x10^{-4} 8.0x10^{-5}

Note:

1. Nominally defined as the 0- to 2-hour interval but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR Part 50.34 requirements.

Table 15A-6

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

χ/Q (s/m³) at HVAC Intake for the Identified Release Points[1]					
	Plant Vent or PCS Air Diffuser[3]	**Ground Level Containment Release Points[4]**	**PORV and Safety Valve Releases[5]**	**Steam Line Break Releases**	**Fuel Handling Area[6]**
0 - 2 hours	2.45E-3	2.45E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	1.65~~7~~E-3	1.65~~7~~E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	6.6~~1.0~~E-4~~3~~	6.6~~1.0~~E-4~~3~~	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	5~~8~~.0E-4	5~~8~~.0E-4	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	4~~7~~.0E-4	4~~8~~.0E-4	4.5E-3	5.0E-3	1.0E-3

χ/Q (s/m³) at Control Room Door for the Identified Release Points[2]					
	Plant Vent or PCS Air Diffuser[3]	**Ground Level Containment Release Points[4]**	**PORV and Safety Valve Releases[5]**	**Steam Line Break Releases**	**Fuel Handling Area[6]**
0 - 2 hours	1.0E-3	1.0~~5~~E-3	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	7~~8~~.0E-4	7~~8~~.0E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	3.5~~4.0~~E-4	3.5~~4.0~~E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	3.0E-4	3~~4~~.0E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	2.5E-4	3~~4~~.0E-4	8.0E-4	8.0E-4	1.0E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.
4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.
5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves and the condenser air removal stack. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident. Additionally, these dispersion coefficients are conservative for the small line break outside containment.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.

APPENDIX 15B

REMOVAL OF AIRBORNE ACTIVITY FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA

The AP1000 design does not depend on active systems to remove airborne particulates or elemental iodine from the containment atmosphere following a postulated loss-of-coolant accident (LOCA) with core melt. Naturally occurring passive removal processes provide significant removal capability such that airborne elemental iodine is reduced to very low levels within a few hours and the airborne particulates are reduced to extremely low levels within 12 hours.

15B.1 Elemental Iodine Removal

Elemental iodine is removed by deposition onto the structural surfaces inside the containment. The removal of elemental iodine is modeled using the equation from the Standard Review Plan (Reference 1):

$$\lambda_d = \frac{K_w A}{V}$$

where:

λ_d = first order removal coefficient by surface deposition

K_w = mass transfer coefficient (specified in Reference 1 as 4.9 m/hr)

A = surface area available for deposition

V = containment building volume

The available deposition surface is 219,000 ft^2, and the containment building net free volume is 2.06×10^6 ft^3. From these inputs, the elemental iodine removal coefficient is 1.7 hr^{-1}.

Consistent with the guidance of Reference 1, credit for elemental iodine removal is assumed to continue until a decontamination factor (DF) of 200 is reached in the containment atmosphere. Because the source term for the LOCA (defined in subsection 15.6.5.3) is modeled as a gradual release of activity into the containment, the determination of the time at which the DF of 200 is reached needs to be based on the amount of elemental iodine that enters the containment atmosphere over the duration of core activity release.

15B.2 Aerosol Removal

The deposition removal of aerosols from the containment atmosphere is accomplished by a number of processes including sedimentation, diffusiophoresis, and thermophoresis. All three of the deposition processes are significant contributors to the overall removal process in the AP1000. The large contributions from diffusiophoresis and thermophoresis to the total removal are a direct consequence of the high heat transfer rates from the containment atmosphere to the containment wall that characterize the passive containment cooling system.

Because of the AP1000 passive containment cooling system design, there are high sensible heat transfer rates (resulting in higher thermophoretic removal of aerosols) when condensational heat transfer is low (and the aerosol removal by diffusiophoresis is also low). The reverse is also true. Thus, there is an appreciable deposition removal throughout the accident from either diffusiophoresis or thermophoresis, in addition to the removal by sedimentation.

15B.2.1 Mathematical Models

The models used for the three aerosol removal processes are discussed as follows.

15B.2.1.1 Sedimentation

Gravitational sedimentation is a major mechanism of aerosol removal in a containment. A standard model (Stokes equation with the Cunningham slip correction factor) for this process is used. The Stokes equation (Reference 2) is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu}$$

where:

υ_s = settling velocity of an aerosol particle

ρ_p = material density of the particle

g = gravitational acceleration

r = particle radius

μ = gas viscosity

Cn = Cunningham slip correction factor, a function of the Knudsen number (Kn) which is the gas molecular mean free path divided by the particle radius

However, the Stokes equation makes the simplifying assumption that the particles are spherical. The particles are expected to be nonspherical, and it is conventional to address this by introducing a "dynamic shape factor" (Reference 2) in the denominator of the Stokes equation, such that the settling velocity for the nonspherical particle is the same as for a spherical particle of equal volume. The value of the dynamic shape factor (ϕ) thus depends on the shape of the particle and, in general, must be experimentally determined.

The concept of dynamic shape factor can also be applied to a spherical particle consisting of two components, one of which has the density of the particle material, while the other component has a different density (Reference 9). In this manner, the impact of the void fraction in the particle can be modeled. Thus, the revised Stokes equation is:

$$\upsilon_s = \frac{2\rho_p g r^2 Cn}{9\mu\phi}$$

The derivation of ϕ follows.

The two-component particle is considered to have a density ρ_{av} and an effective radius of r_e. Assuming that the second component of the particle is the void volume and letting the void fraction be ε, then the average density of the particle is:

ρ_{av} = the average density of the particle = $\rho_p (1-\varepsilon) + \rho_v \varepsilon$

where:

ρ_v	=	density of the void material (0.0 for gas filled, 1.0 for water filled)
ε	=	void fraction
ρ_p	=	material density (solid particle with no voids)

The definition of ϕ is obtained from the Stokes equation and the equation for mass of a sphere:

$$\frac{2\rho_p g r^2 Cn}{9\mu\phi} = \frac{2\rho_{av} g r_e^2 Cn}{9\mu}$$

which reduces to:

$$\rho_p r^2 = \phi \rho_{av} r_e^2$$

and

$$\frac{4\rho_p \pi r_0^3}{3} = \frac{4\rho_{av} \pi r_e^3}{3}$$

which reduces to:

$$\rho_p r_0^3 = \rho_{av} r_e^3$$

Then:

$$\phi = \frac{\rho_p r^2}{\rho_{av} r_e^2}$$

and

$$r_e = r\left(\frac{\rho_{av}}{\rho_p}\right)^{-1/3}$$

From these two relationships, the dynamic shape factor is given by:

$$\phi = \left(\frac{\rho_{av}}{\rho_p}\right)^{-1/3}$$

15B.2.1.2 Diffusiophoresis

Diffusiophoresis is the process whereby particles are swept to a surface (for example, containment wall) by the flow set up by a condensing vapor (Stefan flow). The deposition rate is independent of the particle size and is proportional to the steam condensation rate on the surface. The standard equation for this phenomenon is due to Waldmann and Schmitt (Reference 3):

$$\upsilon_d = \frac{\sqrt{M_v}}{\sqrt{M_v} + \chi_{a/v}\sqrt{M_a}} \frac{W}{\rho_v}$$

where:

υ_d = diffusiophoretic deposition velocity

$\chi_{a/v}$ = ratio of mole fraction of air to mole fraction of steam in the containment atmosphere

M_v = molecular weight of steam

M_a = molecular weight of air

W = steam condensation rate on the wall

ρ_v = mass density of steam in the containment atmosphere

Because of the design of the passive containment cooling system, steam condensation rates are high at certain times in the design basis LOCA; thus at these times, diffusiophoretic deposition rates are significant.

15B.2.1.3 Thermophoresis

Thermophoresis is the process whereby particles drift toward a surface (for example, the containment wall) under the influence of a temperature gradient in the containment atmosphere at the surface. The effect arises because the gas molecules on the hot side of the particles undergo more collisions with the particle than do those on the cold side. Therefore, there is a net momentum transfer to the particle in the hot-to-cold direction. There are several models in the literature for this effect; the one used is the Brock equation in a form due to Talbot et al. (Reference 4). As indicated below, this model is in agreement with experimental data. The thermophoretic deposition rate is somewhat dependent on particle size and is proportional to the temperature gradient at the wall, or equivalently, the sensible heat transfer rate to the wall. The Talbot equation is:

$$\upsilon_{th} = \frac{2\, C_s\, Cn\, (\mu_g/\rho_g)[\alpha + C_T Kn]dT}{[1+2(\alpha + C_T Kn)][1+3C_M Kn]}\left(\frac{1}{T}\right)\frac{dT}{dy}$$

where:

υ_{th} = thermophoretic deposition velocity

α = k_g/k_p which is the ratio of the thermal conductivities of the gas (evaluated at the gas temperature at each time step) and the aerosol particle (k_p is set equal to the thermal conductivity of water – the results are not sensitive to k_p or α.)

Kn = Knudsen number (equal to the gas molecular mean free path divided by the particle radius)

Cn = Cunningham slip correction factor, a function of the Knudsen number

μ_g = gas viscosity

ρ_g = gas density

C_S = slip accommodation coefficient (Reference 4 gives the best value as 1.17.)

C_T = thermal accommodation coefficient (Reference 4 gives the best value as 2.18.)

C_M = momentum accommodation coefficient (Reference 4 gives the best value as 1.14.)

The temperature gradient at the wall, dT/dy, can be evaluated as

$$\frac{dT}{dy} = \frac{\phi_s}{k_g}$$

where ϕ_s is the sensible heat flux to the wall, and k_g is the thermal conductivity of the gas. ~~It is obtained as the difference between the total and condensation heat fluxes.~~ The sensible heat flux is determined using standard heat transfer models for convective and radiative heat transfer for the calculated containment thermal conditions.

15B.2.2 Other Removal Mechanisms

In addition to the above mechanisms, there are others that were not considered, including turbulent diffusion and turbulent agglomeration. The neglect of these mechanisms adds further conservatism to the calculation.

15B.2.3 Validation of Removal Mechanisms

The aerosol processes are well established and have been confirmed in many separate effects experiments, which are discussed in standard references (References 2 through 4). The Stokes formula for sedimentation velocity has been well confirmed for particles whose diameters are less than about 50 μm. In the present calculations, these make up basically all of the aerosol.

There are some separate effects validations of the diffusiophoretic effect, but the best confirmation comes from integral experiments such as the LACE tests (Reference 5). Calculations of these and other integral tests accurately predict the integrated mass of plated aerosol material only if diffusiophoresis is taken into account. If it is neglected, the predicted plated mass is about two orders of magnitude too small, compared to the observed plated mass.

The Talbot equation for the thermophoretic effect has been experimentally confirmed to within about 20 to 50 percent over a wide range of particle sizes (Reference 4). The temperature gradient at the wall, which drives this phenomenon, can be approximated by the temperature difference between the bulk gas and the wall divided by an appropriate length scale obtained from heat transfer correlations. Alternatively, because sensible heat transfer rates to the wall are available, it is easier and more accurate to use these rates directly to infer the temperature gradient.

15B.2.4 Parameters and Assumptions for Calculating Aerosol Removal Coefficients

The parameters and assumptions were selected to conservatively model the environment that would be expected to exist as a result of a LOCA with concurrent core melt.

15B.2.4.1 Containment Geometry

The containment is assumed to be a cylinder with a volume of 55,481 m^3 (1.959×10^6 ft^3). This volume includes those portions of the containment volume that would be participating in the aerosol transport and mixing; this excludes dead-ended volumes and flooded compartments. The horizontal surface area available for aerosol deposition by sedimentation is 2900 m^2 (31,200 ft^2). This includes projecting areas such as decks in addition to the floor area and excludes areas in dead-ended volumes and areas that would be flooded post-LOCA. The surface area for Brownian diffusive plateout of aerosols is 8008 m^2 (86,166 ft^2).

15B.2.4.2 Source Size Distribution

The aerosol source size distribution is assumed to be lognormal, with a geometric mean radius of 0.22 μm and a geometric standard deviation equal to 1.81. These values are derived from an evaluation of a large number of aerosol distributions measured in a variety of degraded-fuel tests and experiments. The sensitivity of aerosol removal coefficient calculations to these values is small.

15B.2.4.3 Aerosol Void Fraction

Review of scanning electron microscope photographs of deposited aerosol particles from actual core melt and fission product vaporization and aerosolization experiments (the Argonne STEP-4 test and the INEL Power Burst Facility SFD 1-4 test) indicates that the deposited particles are relatively dense, supporting a void fraction of 0.2.

The above-mentioned test results indicate that a void fraction of 0.2 is appropriate for modeling the aerosols resulting from a core melt. As part of the sensitivity study that was performed for the AP600 project, a case was run with a void fraction of 0.9. That analysis showed that the high void fraction resulted in an integrated release of aerosols over a 24-hour period that was less than 14 percent greater than that calculated when using the void fraction of 0.2. Thus, it is clear that the removal of aerosols from the containment atmosphere is not highly sensitive to the value selected for the void fraction. This is largely due to the fact that, while the selected value for void fraction has a significant impact on the calculated sedimentation removal, the impact on thermophoresis and diffusiophoresis removal is slight or none. The impact for AP1000 of using the higher value for void fraction would be less than was determined for the AP600 since sedimentation removal comprises a smaller fraction of the total removal calculated for the AP1000.

For additional conservatism the AP1000 aerosol removal analysis uses a void fraction of 0.4 and assumes the voids are filled with air.

15B.2.4.4 Fission Product Release Fractions

Core inventories of fission products are from ORIGEN calculations for the AP1000 at end of the fuel cycle. Fractional releases to the containment of the fission products are those specified in subsection 15.6.5.3.

15B.2.4.5 Inert Aerosol Species

The inert species include SnO_2, UO_2, Cd, Ag, and Zr. These act as surrogates for all inert materials forming aerosols. The ratio of the total mass of inert species to fission product species was assumed to be 1.5:1. This value and the partitioning of the total inert mass among its constituents are consistent with results from degraded fuel experiments (Reference 6).

15B.2.4.6 Aerosol Release Timing and Rates

Aerosol release timing is in accordance with the source term defined in subsection 15.6.5.3. Aerosol release takes place in two main phases: a gap release lasting for 0.5 hour, followed by an early in-vessel release of 1.3 hours duration. During each phase, the aerosols are assumed to be released at a constant rate. These rates were obtained for each species by combining its core inventory, release fraction, and times of release. Only cesium and iodine are released during the gap release phase. During the in-vessel release phase, the other fission product and inert species are released as well.

15B.2.4.7 Containment Thermal-hydraulic Data

The thermal-hydraulic parameters used in the aerosol removal calculation are the containment gas temperature, the containment pressure, the steam condensation rate on the wall, the steam mole fraction, and the ~~total~~ convective and radiative heat transfer rates, all as functions of time. The AP1000-specific parameters were obtained using MAAP4 (Reference 7) for the 3BE-1 severe accident sequence (medium LOCA with failure to inject water from the refueling water storage tank into the reactor vessel). The convective and radiative heat transfer rates are separately calculated using standard heat transfer models for free convection and radiative heat transfer using the gas and wall temperature conditions from the MAAP4 analysis. The thermal-hydraulic data are thus consistent with a core melt sequence.

15B.2.5 Aerosol Removal Coefficients

The aerosol removal coefficients are provided in Table 15B-1 starting at the onset of core damage through 24 hours. The removal coefficients for times beyond 24 hours are not of concern because there would be so little aerosol remaining airborne at that time. The values range between ~~0.6~~0.54 hr^{-1} and 1.3~~5~~ hr^{-1} during the time between the onset of core damage (0.167 hour) and 24 hours.

These removal coefficients conservatively neglect steam condensation on the airborne particles, turbulent diffusion, and turbulent agglomeration. Additionally, the assumed source aerosol size is conservatively small being at the low end of the mass mean aerosol size range of 1.5 to 5.5 μm used in NUREG/CR-5966 (Reference 8). Selection of smaller aerosol size would underestimate sedimentation.

Unlike the case for the elemental iodine removal, there is no limit assumed on the removal of aerosols from the containment atmosphere.

15B.3 References

1. NUREG-0800, Section 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System."

2. Fuchs, N. A., The Mechanics of Aerosols, Pergamon Press, Oxford, 1964.

3. Waldmann, L., and Schmitt, K. H., "Thermophoresis and Diffusiophoresis of Aerosols," Aerosol Science, C. N. Davies, ed., Academic Press, 1966.

4. Talbot, L., Chang, R. K., Schefer, R. W., and Willis, D. R., "Thermophoresis of Particles in a Heated Boundary Layer," J. Fluid Mech. 101, 737-758 (1980).

5. Rahn, F. J., "The LWR Aerosol Containment Experiments (LACE) Project," Summary Report, EPRI-NP-6094D, Electric Power Research Institute, Palo Alto, Nov. 1988.

6. Petti, D. A., Hobbins, R. R., and Hagrman, D. L., "The Composition of Aerosols Generated during a Severe Reactor Accident: Experimental Results from the Power Burst Facility Severe Fuel Damage Test 1-4," Nucl. Tech. 105, p.334 (1994).

7. MAAP4 - Modular Accident Analysis Program for LWR Power Plants, Computer Code Manual, May 1994.

8. Powers D. A., and Burson, S. B., "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, June 1993.

9. Powers, D. A., "Monte Carlo Uncertainty Analysis of Aerosol Behavior in the AP600 Reactor Containment under Conditions of a Specific Design-Basis Accident, Part 1," Technical Evaluation Report, Sandia National Laboratories, June 1995.

Table 15B-1

AEROSOL REMOVAL COEFFICIENTS IN THE AP1000 CONTAINMENT FOLLOWING A DESIGN BASIS LOCA WITH CORE MELT

Time Interval (hours)	Removal Coefficient (hr^{-1})
0.167 - 0.~~3~~2	1.3~~25~~
0.~~3~~2 - 0.29~~47~~	1.2~~5~~
0.294 - 0.346~~1~~	1.14
0.34~~6~~ - 0.~~8~~4	1.2~~5~~
0.~~8~~4 - 0.84~~1.0~~	1.0~~4~~
0.84~~1~~ - 1.08~~2~~4	0.95~~1.3~~
1.08~~2~~ - 1.16~~29~~	0.94~~1.0~~
1.16~~2~~ - 1.234~~9~~	0.93~~1.2~~
1.234 - 1.37~~55~~	0.91~~6~~
1.37~~5~~ - 1.41~~5~~8	0.9~~1.0~~
1.41~~5~~ - 1.46	0.88~~1.1~~
1.46 - 1.56~~8~~	0.86~~1.2~~
1.56~~8~~ - 1.62~~.2~~	0.84~~1.15~~
1.62~~.2~~ - 1.69~~2.6~~	0.82~~1.1~~
1.69~~2~~ - 1.784~~.2~~	0.8~~1.05~~
1.784 - 1.86~~5.0~~	0.78~~1.1~~
1.86~~5~~ - 2.06~~8.2~~	0.75~~1.15~~
2.06~~8~~ - 2.38~~2~~4	0.72~~1.1~~
2.38 - 2.5	0.7
2.5 - 2.7	0.6
2.7 - 3.28	0.65
3.28 - 3.89	0.63
3.89 - 4.0	0.54
4.0 - 4.2	0.6
4.2 - 4.37	0.64
4.37 - 5.02	0.66
5.02 - 7.2	0.7
7.2 - 24	0.73

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 15.3.6-1 Revision 1

Original RAI Number(s): 470.009, 470.011

Summary of Issue:

The staff has not completed its evaluation of the applicant's assumptions on aerosol removal in containment, as discussed in RAIs 470.009 and 470.011. To verify the applicant's assessment, the staff will perform independent radiological consequence calculations for a postulated design-basis LOCA coincident with the loss of spent fuel pool cooling capability once these issues are resolved. This is Open Item 15.3.6-1.

Westinghouse Response:

The Westinghouse responses to RAI 470.009 transmitted by Westinghouse letter DCP/NRC1535, November 26, 2002 and RAI 470.011 Rev. 1 transmitted by Westinghouse letter DCP/NRC1571, April 11, 2003 address previous NRC comments related to this issue.

Westinghouse Response (Revision 1):

The aerosol removal analysis is revised as discussed in the response to Open Item 15.3-1 Revision 2.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19.1.10.2-1 (Response Revision 2)**

Original RAI Number(s): **720.099**

Summary of Issue: Shutdown Risk Due to Vacuum Refill Operations

The reported CDF from internal events during shutdown operation (1.2E-07/year) covers two plant operational states:

- safe shutdown/cold shutdown with the RCS filled and intact, and
- mid-loop/vessel flange operations with the RCS vented and drained.

Mid-loop/vessel flange operations include. (1) draining to mid-loop, and (2) drained maintenance, and post-refueling maintenance.

Vacuum refill of the RCS from drained conditions (mid-loop) was mentioned in the PRA. However, no risk assessment was performed for this configuration. Vacuum refill of the RCS helps to reduce non-condensible gas pockets in the RCS, eliminating the need for dynamic venting of the RCS and the multiple reactor coolant pump start and stop operations that it requires.

The applicant stated that the shutdown risk due to vacuum refill operations is included in the calculation of shutdown risk during vented drained conditions. The staff is reviewing the applicant's response to RAI 720.99 to determine if the shutdown risk due to vacuum refill operations is included in the calculation of shutdown risk during vented drained conditions. The staff noted during their review of the applicant's response to RAI 720.99 that Investment Protection Short term Availability Controls do not include RNS and its support systems such as Component Cooling Water System, Service Water system, and ac power supplies during vacuum refill operations. Assuming an extended loss of RNS during vacuum refill operations, the staff questions using the RNS suction relief valve to relief RCS pressure should the operators not open the ADS valves. The operators may instead isolate the RNS suction relief valve to isolate RCS leakage. As discussed in Section 19.1.10.2 of this report, this vacuum refill issue is considered to be Open Item 19.1.10.2-1.

Westinghouse Response: Revision 2

The Revision 2 response to this DSER Open Item is based on a teleconference with the NRC staff in a teleconference on 3/25/04. The response has been updated to address the NRC comments.

The Revision 0 response to this DSER Open Item was discussed with the NRC staff on a teleconference. The response has been updated to address the NRC comments.

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

Westinghouse provided its response to RAI 720.099 in order to address this NRC concern about whether vacuum refill operations are bounded by the current AP1000 shutdown PRA. This response (rev. 0) was submitted to the NRC on March 28, 2003 in letter DCP/NRC1558.

The following discussion provides further support of the treatment of vacuum refill operation in the AP1000 PRA. Vacuum refill is performed during Mode 5 conditions with~~, and~~ the RNS ~~is~~ in service to provide core cooling ~~during vacuum refill operations~~. The short-term availability controls provided for AP1000 address the availability of the RNS and its support systems to be available during MODE 5 including vacuum refill operations. The Bases for these short-term availability controls state that "The RCS is also considered open if there is no visible level in the pressurizer". As a result, ~~N~~no special designation of vacuum refill in the short-term availability controls is required.

Vacuum refill operations do not pose additional shutdown risk to the AP1000 because of the following reasons:

1. The decay heat during vacuum refill will be about 50% of that during drained conditions early in a shutdown, which is already considered in the shutdown PRA.

2. Although ADS stage 1/2/3 will be closed, 9 of the 10 ADS paths are required by Technical Specifications to be operable when the RCS has reduced inventory and pressure boundary is closed, as it is during vacuum refill operations. As a result, at least 3 of 4 ADS stage 4 valves will be operable instead of the 2 of 4 during RCS drained / open conditions. Note that with the low decay heat during vacuum refill operations, the ADS Stage 4 valves are sufficient to support core cooling using passive systems without opening the ADS Stage 1, 2, and 3 valves. The ADS Stage 4 valves and IRWST injection would be automatically actuated on a low HL level signal.

3. The time spent in vacuum degassing is small compared to the time spent in drained / open shutdown conditions.

4. During vacuum refill operations, both RNS pumps and support systems are required to be available by the short-term availability controls (MODE 5, reduced inventory conditions). Therefore, there is a high probability that the RNS pumps will continue to operate during vacuum refill. The probability of a loss of RNS or its support systems in MODE 5 is adequately addressed in the shutdown PRA.

The AP1000 Emergency Response Guidelines (ERG – SDG-2, Step 6) provide direction for the operators for a loss of RNS during shutdown conditions, and the ERG response is applicable during vacuum refill operations in MODE 5.

For a loss of RNS cooling during vacuum refill operations, the operators are immediately directed to open the ADS Stage 1, 2, and 3 valves~~, whether the RNS cooling was lost or whether the loss of RNS resulted from a loss of RCS inventory~~. There are two significant reasons why the operators would not be expected to isolate the RNS. First, on a loss of RNS, the primary goal is to restore the RNS operation. The operators would not isolate RNS since the RNS can not be restored to service if the system is isolated from the RCS.

Second, RNS provides the low-temperature overpressure protection for the plant during MODE 5 conditions (including vacuum refill operations), in accordance with Technical Specification 3.4.14. The operators are trained on brittle fracture prevention and the RCS pressure-temperature limits, so they thoroughly understand their priority to maintain the RCS overpressure protection flow path to the RNS during low-pressure, low-temperature shutdown conditions. Therefore, it is extremely unlikely that they would take any intentional or unintentional actions to isolate the RNS following a loss of RNS cooling~~, with one specific exception~~. A PRA insight will be added concerning the importance of the operators not isolating the RNS unless there is a low HL level.

The only time the operators would be expected to isolate the RNS is when there is a significant loss of RCS inventory which causes the HL level to drop to empty. Note that the CVS letdown line is automatically isolated at a relatively high level in the hot legs which would stop the level decrease if this line is the cause of the loss of inventory. In this case, the RNS would not be isolated. If isolation of the CVS line does not stop the loss of inventory, then the inventory loss must be from the RCS or RNS lines. In this case, the operators are directed to identify the source of the leakage. Since there normally would be personnel inside containment during vacuum refill operations, it would be relatively easy to identify the source of the leak.

If the source of the leak is determined to be the RNS, the RNS pumps would be stopped and the lines isolated. In this situation, the ERGs require that the ADS Stage 1, 2, and 3 valves, which maintains LTOPs. If the leak is from the RCS, the operators would stop RNS pumps if the HL level drops to empty in order to prevent damaging the pumps. Again, if the RNS pumps are stopped, the operators are instructed by the ERGs to open the ADS Stage 1, 2, and 3 lines. Note that it is very unlikely that LTOPS would be required in a RCS leak / LOCA condition with a water level in the HL region.

The AP1000 shutdown PRA does not credit the RNS in mitigating shutdown events involving the loss of RCS inventory or failures of the RNS/CCW/SW. The PRA only credits RNS operation following a loss of offsite power during shutdown conditions.

Design Control Document (DCD) Revision:

~~None~~Revise DCD Table 19.59-18 to add insight #82. Note that insights #79 – #81 were added in previous revisions to the corresponding PRA insight table (Table 59-18) but were not added to this DCD table. These insights will be added to the DCD table as shown below.

Table 19.59-18 (Sheet 24 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
76. An alternative gravity injection path is provided through RNS V-023 during cold shutdown and refueling conditions with the RCS open.	Emergency Response Guidelines
The COL applicant is responsible for developing administrative controls to maximize the likelihood that RNS valve V-023 will be able to open if needed during Mode 5 when the RCS is open, and PRHR cannot be used for core cooling.	13.5
77. The IRWST suction isolation valve (V023) and the RCS pressure boundary isolation valves (V001A/B, V002A/B) are environmentally qualified to perform their safety functions.	Tier 1 Information
78. Following an extended loss of RNS during safe/cold shutdown with the RCS intact and PRHR unavailable, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation.	19.59.5
79. Combined License applicants referencing the AP1000 certified design will provide resolution for generic open items and plant-specific action items resulting from NRC review of the I&C platform.	7.1.6
80. The Combined License applicant will provide an analysis that demonstrates that operator actions, which minimize the probability of the potential for spurious ADS actuation as a result of a fire, can be accomplished within 30 minutes following detection of the fire and the procedure for the manual actuation of the valve to allow fire water to reach the automatic fire system in the containment maintenance floor.	9.5.1.8
81. The Combined License applicant will establish procedures to address a fire watch for fire areas breached during maintenance.	9.5.1.8
82. It is important to maintain the low temperature overpressure protection provided by the RNS relief valve to ensure that the reactor vessel pressure and temperature limits are not exceeded during shutdown conditions. Isolation of the RNS and its relief valve are permitted during shutdown conditions in case the hot legs empty due to a loss of RCS inventory; if the RNS is isolated, an alternate vent path would be opened, such as the ADS Stage 1, 2, and 3 valves.	16.1 (LCO Basis 3.4.14)

PRA Revision:

~~None~~Revise PRA Table 59-18 to add insight #82 as shown below:

Table 59-18 (Sheet 24 of 24)

AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
76.	An alternative gravity injection path is provided through RNS V-023 during cold shutdown and refueling conditions with the RCS open.	Emergency Response Guidelines
	The COL applicant is responsible for developing administrative controls to maximize the likelihood that RNS valve V-023 will be able to open if needed during Mode 5 when the RCS is open, and PRHR cannot be used for core cooling.	13.5
77.	The IRWST suction isolation valve (V023) and the RCS pressure boundary isolation valves (V001A/B, V002A/B) are environmentally qualified to perform their safety functions.	Tier 1 Information
78.	Following an extended loss of RNS during safe/cold shutdown with the RCS intact and PRHR unavailable, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation.	19.59.5
79.	Combined License applicants referencing the AP1000 certified design will provide resolution for generic open items and plant-specific action items resulting from NRC review of the I&C platform.	7.1.6
80.	The Combined License applicant will provide an analysis that demonstrates that operator actions, which minimize the probability of the potential for spurious ADS actuation as a result of a fire, can be accomplished within 30 minutes following detection of the fire and the procedure for the manual actuation of the valve to allow fire water to reach the automatic fire system in the containment maintenance floor.	9.5.1.8
81.	The Combined License applicant will establish procedures to address a fire watch for fire areas breached during maintenance.	9.5.1.8
82.	It is important to maintain the low temperature overpressure protection provided by the RNS relief valve to ensure that the reactor vessel pressure and temperature limits are not exceeded during shutdown conditions. Isolation of the RNS and its relief valve are permitted during shutdown conditions in case the hot legs empty due to a loss of RCS inventory; if the RNS is isolated, an alternate vent path would be opened, such as the ADS Stage 1, 2, and 3 valves.	16.1 (LCO Basis 3.4.14)