AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 2.3.4-1 Revision 2

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

Summary of Issue:

The hypothetical reference control room χ/Q values calculated by the applicant are listed in Table 15.3-9a of this report. A site selected for an AP1000 facility should have control room χ/Q values equal to or less than the hypothetical Reference χ/Q values shown Table 15.3-9a. In the event a site selected for the AP1000 design exceeds the hypothetical reference χ/Q values, the COL applicant should demonstrate that the radiological consequences associated with the design-basis accidents, using its site-specific χ/Q values, continues to meet the dose reference values given in GDC 19 of 10 CFR Part 50. The staff initially asked the applicant if the methodology and all inputs and assumptions would be evaluated as part of the COL review. The applicant provided a detailed response stating that the methodology, inputs and assumptions would be provided by the COL applicant and also provided additional information about the analysis. The staff issued a second RAI to inquire if the applicant was seeking certification of any of the AP1000 design values used as inputs to the control room χ/Q calculations. The applicant subsequently provided certain design-specific information that was used as input to the assessment and for which the applicant was seeking certification. The staff review of this topic is ongoing, and may reveal other concerns with respect to χ/Q. The staff has identified unresolved issues related to adequate justification for assuming a diffuse release, estimation of initial sigma values, other release assumptions, building cross-sectional areas, and distances between release/receptor pairs. This is Open Item 2.3.4-1. This is also COL Action Item 2.3.4-1 since the resultant χ/Q values are also a function of the site-specific meteorology which cannot be reviewed until site selection.

Follow-on Question:

Westinghouse submitted revisions to address this question in DCD Revision 7. The staff had further questions in response to those revisions, particularly relating to the modeling used in the demonstration case control room χ/Q values.

Westinghouse Response:

The AP1000 control room χ/Q values used in the AP1000 dose analyses were based on the calculation performed for the AP600 Design Certification. This calculation examined a wide range of site meteorological data and plant orientations to develop a conservative set of χ/Q values for use in the AP600 dose analyses. However, following the issue of Regulatory Guide 1.194 (which provides specific NRC staff guidance on the use of the ARCON96 code for calculating control room χ/Q) it was determined that the the modeling assumptions used for AP600 did not fully comply with that Regulatory Guide.

An additional issue is whether the control room χ/Q values are being approved for AP1000 as part of Design Certification. Unlike the offsite χ/Q values that were identified as site interface parameters that the COL applicant would later verify for their site, the control room χ/Q values were not identified as a site interface for either the AP600 or AP1000. Westinghouse agrees with the NRC that control room χ/Q values should be identified as a site interface parameter. Thus a COL applicant would verify as part of the COL process that the calculated control room χ/Q values for their site are bounded by those assumed in the DCD dose analysis. However, unlike the site boundary χ/Q values that are based solely on the site meteorological data, the control room χ/Q values are determined based both on site meteorological data and assumptions related to plant design features and layout. Westinghouse believes that the assumptions related to the plant design features are important in ultimately determining acceptable control room doses for design basis accidents, and therefore should be approved as part of Design Certification.

Therefore the following approach is being taken to resolve these issues:

1. Bounding control room χ/Q values will be established for the AP1000. These values will be determined for the various source – receptor locations that are applicable for the various design basis accidents as appropriate. χ/Q values that will still yield doses within the dose acceptance limits will be calculated consistent with the dose analysis methodology and assumptions described in the DCD Chapters 6 and 15. Consistent with the approach of treating the control room χ/Q values as interface parameters, Westinghouse will revise some assumptions described in the current DCD dose analysis to remove excess conservatism to provide the COL applicant greater flexibility in demonstrating acceptability. These changes were incorporated in DCD Revision 7; minor additional changes are provided in the attached DCD markup and will be included in DCD Revision 8.
2. The key control room χ/Q modeling assumptions related to the plant design will be added to DCD Appendix 15A. This was also incorporated in DCD Revision 7, but has been revised in view of revised modeling techniques now applied; changes are shown in the attached pages and are to be incorporated into DCD Revision 8.
3. A set of χ/Q values for typical site meteorology and plant orientation have been calculated in accordance with the guidance set forth in Regulatory Guide 1.194. The purpose of the calculation is to define the modeling assumptions for calculating the control room χ/Q values for AP1000, and will serve as an example of an approved method for the COL applicant to follow to determine the acceptability of their site to meet the control room χ/Q values. Relevant portions of this calculation are now included in DCD Appendix 15A.

   The ARCON96 modeling approach used (as illustrated in attached DCD mark-up, Table 15.A-7 and Figure 15A-1) is fully in compliance with Regulatory Guide 1.194.

Design Control Document (DCD) Revision:
See attached draft markup of the DCD.

PRA Revision:
None

2. Site Characteristics — AP1000 Design Control Document

Table 2-1 (Sheet 3 of 3)

SITE PARAMETERS

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

χ/Q (s/m^3) 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	2.5E-3	2.5E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	1.7E-3	1.7E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	1.0E-3	1.0E-3	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	8.0E-4	8.0E-4	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	7.0E-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]

	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	1.5E-3	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	8.0E-4	8.0E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	4.0E-4	4.0E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	3.0E-4	4.0E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	2.5E-4	4.0E-4	8.0E-4	8.0E-4	1.0E-3

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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

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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.

The emergency air storage tanks are sized to provide the required air flow to the main control room pressure boundary for 72 hours. After 72 hours, the main control room is cooled by drawing in outside air and circulating it through the room, as discussed in subsection 6.4.2.2.

The temperature and humidity in the main control room pressure boundary following a loss of the nuclear island nonradioactive ventilation system remain within limits for reliable human performance (References 2 and 3) over a 72-hour period. The initial values of temperature/relative humidity in the MCR are 75°F/60 percent. At 3 hours, when the non-1E battery heat loads are exhausted, the conditions are 87.2°F/41 percent. At 24 hours, when the 24 hour battery heat loads are terminated, the conditions are 84.4°F/45 percent. At 72 hours, the conditions are 85.8°F/ 39 percent.

Sufficient thermal mass is provided in the walls and ceiling of the main control room to absorb the heat generated by the equipment, lights, and occupants. The temperature in the instrumentation and control rooms and dc equipment rooms following a loss of the nuclear island nonradioactive ventilation system remains below acceptable limits as discussed in subsection 6.4.4. As in the main control room, sufficient thermal mass is provided surrounding these rooms to absorb the heat generated by the equipment. After 72 hours, the instrumentation and control rooms will be cooled by drawing in outside air and circulating it through the room, as discussed in subsection 6.4.2.2.

In the event of a loss of ac power, the nuclear island nonradioactive ventilation system isolation valves automatically close and the main control room emergency habitability system isolation valves automatically open. These actions protect the main control room occupants from a potential radiation release. In instances in which there is no radiological source term present, the compressed air storage tanks are refilled via a connection to the breathable quality air compressor in the compressed and instrument air system (CAS). The compressed air storage tanks can also be refilled from portable supplies by an installed connection in the CAS.

6.4.4 System Safety Evaluation

Doses to main control room personnel were calculated for both the situation in which the emergency habitability system (VES) is relied upon to limit the amount of activity the personnel are exposed to and the situation in which the nuclear island nonradioactive ventilation system (VBS) is available to pressurize the main control room with filtered air and provide recirculation cleanup. Doses were calculated for the following accidents:

	VES Operating	VBS Operating
Large Break LOCA	4.8 rem TEDE	~~2.8~~ rem TEDE
Fuel Handling Accident	4.5 rem TEDE	~~2.1~~ rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	4.8 rem TEDE	~~2.1~~ rem TEDE
(Accident-initiated iodine spike)	2.1 rem TEDE	~~1.8~~ rem TEDE
Steam Line Break		
(Pre-existing iodine spike)	3.4 rem TEDE	~~2.1~~ rem TEDE
(Accident-initiated iodine spike)	3.7 rem TEDE	~~1.0~~ rem TEDE
Rod Ejection Accident	2.1 rem TEDE	~~1.2~~ rem TEDE

Comment: "VBS Operating" values have not yet been recalculated and may change, but will be bounded by "VES Operating" case.

	VES Operating	VBS Operating
Locked Rotor Accident		
(Accident without feedwater available)	0.9 rem TEDE	~~0.9~~ rem TEDE
(Accident with feedwater available)	0.7 rem TEDE	~~1.6~~ rem TEDE
Small Line Break Outside Containment	1.2 rem TEDE	~~0.7~~ rem TEDE

Comment: "VBS Operating" values have not yet been recalculated and may change, but will be bounded by "VES Operating" case.

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For all events the dose are within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are delineated in the LOCA dose analysis discussion in subsection 15.6.5.3.

No radioactive materials are stored or transported near the main control room pressure boundary.

As discussed and evaluated in subsection 9.5.1, the use of noncombustible construction and heat and flame resistant materials throughout the plant reduces the likelihood of fire and consequential impact on the main control room atmosphere. Operation of the nuclear island nonradioactive ventilation system in the event of a fire is discussed in subsection 9.4.1.

The exhaust stacks of the onsite standby power diesel generators are located in excess of 150 feet away from the fresh air intakes of the main control room. The onsite standby power system fuel oil storage tanks are located in excess of 300 feet from the main control room fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the main control room.

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

A supply of protective clothing, respirators, and self-contained breathing apparatus adequate for 11 persons is stored within the main control room pressure boundary.

The main control room emergency habitability system components discussed in subsection 6.4.2.3 are arranged as shown in Figure 6.4-2. The location of components and piping within the main control room pressure boundary provides the required supply of compressed air to the main control room pressure boundary, as shown in Figure 6.4-1.

During emergency operation, the main control room emergency habitability system passive heat sinks are designed to limit the temperature inside the main control room to remain within limits for reliable human performance (References 2 and 3) over 72 hours. The passive heat sinks limit the air temperature inside the instrumentation and control rooms to 120°F and dc equipment rooms to 120°F. The walls and ceilings that act as the passive heat sinks contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours.

The main control room emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks. Sixty scfm of ventilation flow is sufficient to pressurize the control room to at least positive 1/8-inch water gauge differential

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	2.5E-3	2.5E-3	2.0E-2	2.4E-2	6.0E-3
2 - 8 hours	1.7E-3	1.7E-3	1.8E-2	2.0E-2	4.0E-3
8 - 24 hours	1.0E-3	1.0E-3	7.0E-3	7.5E-3	2.0E-3
1 - 4 days	8.0E-4	8.0E-4	5.0E-3	5.5E-3	1.5E-3
4 - 30 days	7.0E-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	1.5E-3	4.0E-3	4.0E-3	6.0E-3
2 - 8 hours	8.0E-4	8.0E-4	3.2E-3	3.2E-3	4.0E-3
8 - 24 hours	4.0E-4	4.0E-4	1.2E-3	1.2E-3	2.0E-3
1 - 4 days	3.0E-4	4.0E-4	1.0E-3	1.0E-3	1.5E-3
4 - 30 days	2.5E-4	4.0E-4	8.0E-4	8.0E-4	1.0E-3

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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.

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15. Accident Analyses — AP1000 Design Control Document

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.

Table 15A-7

CONTROL ROOM SOURCE/RECEPTOR DATA FOR DETERMINATION OF ATMOSPHERIC DISPERSION FACTORS

Source Description	Release Elevation Note 1 (m)	Straight-Line Distance to Receptor (m)		
		Control Room HVAC Intake (Elevation 19.9 m)	Annex Building Access (Elevation 1.5 m)	Comment
Plant Vent	55.7	53.4	94.0	
PCS Air Diffuser	71.3	60.7	98.1	
Containment Shell (Diffuse Area Source)	Same as receptor elevation (19.9 m or 1.5 m)	11.0	47.2	Note 2
Fuel Building Blowout Panel	17.4	50	89.7	Note 3
Fuel Building Rail Bay Door	1.5	52.4	92.1	Note 3
Steam Vent	17.1	18.3	48.8	
PORV/Safety Valves	19.2	19.8	44.1	
Condenser Air Removal Stack	7.6	63	59.9	Note 3

Notes:
1. All elevations relative to grade at 0.0 m
2. For calculating distance, the source is defined as the point on the containment shell closest to receptor.
3. Vertical distance traveled is conservatively neglected.

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4. . The staging area hatch doors are located such that any release would be into the auxiliary building. The release would then be required to pass through doors into the annex building and travel through the annex building before reaching the ultimate release point. The ultimate release point for the staging area hatch is considered to be the sliding door in the east wall of the annex building between columns 4 and 4.1 at elevation 107'-6" (see Figure 15A-1). Therefore the horizontal distance of 101 ft traveled inside the annex building was included in the source to receptor distance. For conservatism, the vertical distance traveled inside the buildings is not included.¶
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Main Equipment Hatch	1.5	106.2	106.1	Notes 2, 3
Staging Area Hatch	4.6	89.4	101.6	Notes 3, 4

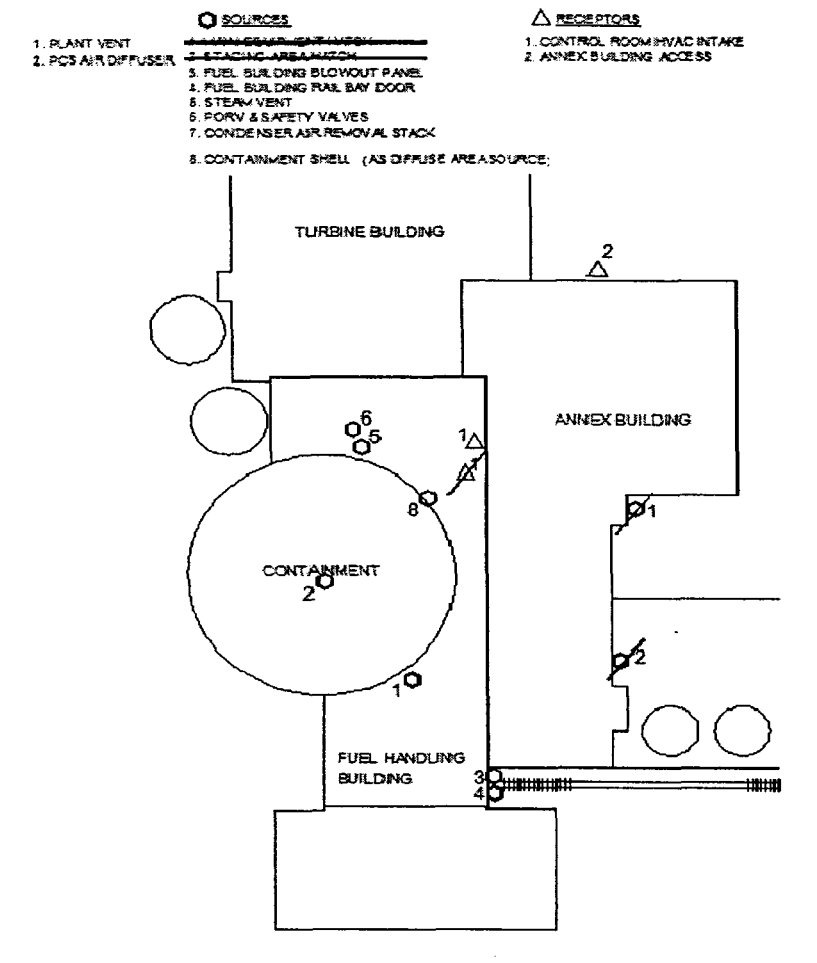

Figure 15A-1

Site Plan with Release and Intake Locations