ATTACHMENT 8

Diablo Canyon Power Plant
Updated Final Safety Analysis Report Markup
(For Information Only)

releases of radioactive materials to the atmosphere and (2) coping with radiological emergencies.

2.3.1.4 Safety Guide 23, February 1972 – Onsite Meteorological Programs

An onsite meteorological monitoring program that is capable of providing meteorological data needed to estimate potential radiation doses to the public as a result of routine or accidental release of radioactive material to the atmosphere and to asses other environmental effects is provided.

2.3.1.5 Regulatory Guide 1.97, Revision 3 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Control room display instrumentation for use in determining the magnitude of the release of radioactive materials and in continuously assessing such releases during and following an accident is provided.

2.3.1.6 Regulatory Guide 1.111, March 1976 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

Annual average relative concentration values are used during the postulated accident to estimate the long-term atmospheric transport and dispersion of gaseous effluents in routine releases.

2.3.1.7 Regulatory Guide 1.111, Revision 1, July 1977 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

In accordance with the requirement of Regulatory Guide 1.145, Revision 1 annual average relative concentration values are developed for each sector, at the outer low population zone (LPZ) boundary distance for that sector, using the method described in Regulatory Position C.1.c of Regulatory Guide 1.111, Revision 1. This information is used as input to develop the design basis radiological analysis χ/Q values at the LPZ using Regulatory Guide 1.145, Revision 1 methodology.

2.3.1.8 Regulatory Guide 1.145, Revision 1, February 1983 – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

The method outlined in Regulatory Guide 1.145, Revision 1, (with the exception of methodology associated with elevated or stack releases, i.e., Regulatory Positions C.1.3.2, C.2.1.2 and C.2.2.2), is used for calculating short-term atmospheric dispersion factors for off-site locations such as the exclusion area boundary or the low population zone for design basis radiological analysis dispersion factors.

2.3.1.9 Regulatory Guide 1.194, June 2003 – Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

The method outlined in Regulatory Positions C.1 through C.3, and the adjustment factor for vertically orientated energetic releases from steam relief valves and atmospheric dump valves allowed by Regulatory Position C.6 of Regulatory Guide 1.194, June 2003 is used to determine short-term on-site atmospheric dispersion factors in support of design basis radiological habitability assessments.

2.3.1.~~7~~10 NUREG-0737 (Item III.A.2), November 1980 – Clarification of TMI Action Plan Requirements

Item III.A.2 - Improving Licensee Emergency Preparedness–Long-Term:

Reasonable assurance is provided that adequate protective measures can and will be taken in the event of a radiological emergency. The requirements of NUREG-0654, Revision 1, November 1980, which provides meteorological criteria to ensure that the methods, systems and equipment for monitoring and assessing the consequences of radiological emergencies are in use, is implemented.

Item III.A.2.2 - Meteorological Data: NUREG-0737, Supplement 1, January 1983 provides the requirements for III.A.2.2 as follows:

Reliable indication of the meteorological variables specified in Regulatory Guide 1.97, Revision 3, for site meteorology is provided.

2.3.1.~~8~~11 IE Information Notice 84-91, December 1984 – Quality Control Problems of Meteorological Measurements Programs

Meteorological data that are climatically representative, of high quality, and reliable in providing credible dose calculations and recommendations for protective actions in an emergency situation, and for doses calculated to assess the impact of routine releases of radioactive material to the atmosphere are available.

2.3.2 REGIONAL CLIMATOLOGY

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

2.3.2.1 Data Sources

The information used in determining the regional meteorological characteristics of Diablo Canyon Power Plant (DCPP) site consists of climatological summaries, technical studies, and reports by Dye (Reference 2), Edinger (Reference 3), Elford (Reference 4),

description of computer program EN-113) using a continuous temporally representative 5-year period of hourly meteorological data from the onsite meteorological tower (i.e., January 1, 2007 through December 31, 2011). EN-113 calculates χ/Q values for the various averaging periods using hourly meteorological data related to wind speed, wind direction, and stability class.

Equations used to determine the χ/Q's are as follows:

$$\chi/Q_1 = \{(u)[(\pi)(\sigma_y)(\sigma_z) + (A/2)]\}^{-1} \quad (2.3\text{-}7)$$

$$\chi/Q_2 = [(u)(3\pi)(\sigma_y)(\sigma_z)]^{-1} \quad (2.3\text{-}8)$$

$$\chi/Q_3 = [(u)(\pi)(\Sigma_y)(\sigma_z)]^{-1} \quad (2.3\text{-}9)$$

where:

χ/Q = relative concentration (sec/m^3);
σ_y, σ_z = horizontal and vertical dispersion coefficients, respectively, based on stability class and horizontal downwind distance (m);
u = wind speed at the 10-meter elevation (m/sec);
A = cross-sectional building area (m^2);
Σ_y = $(M)(\sigma_y)$ for distances of 800 meters or less; and
Σ_y = $[(M-1)(\sigma_{y800m}) + \sigma_y]$ for distances greater than 800 meters with M representing the meander factor in Reference 22, Figure 3.

Per Regulatory Guide 1.145, Revision 1, χ/Q_1 and χ/Q_2 values are calculated by EN-113 and the higher value selected. This value is then compared to the χ/Q_3 value calculated by EN-113, and the smaller value is then selected as the appropriate value.

The EAB distances for the sixteen 22.5°-azimuth downwind sectors are derived from Figure 2.3-2, taking into consideration a 45-degree azimuth sector centered on each 22.5°-azimuth sector as described in Regulatory Guide 1.145, Revision 1, Regulatory Position C.1.2. The EAB χ/Q values for the radiological releases from each unit are conservatively based on the EAB distances from the outer edge of each containment building.

An LPZ distance of 6 miles (9,654 meters) is used in the analysis. The use of one LPZ distance in all downwind directions from the center of the site for all release points is reasonable given the magnitude of this distance relative to the separation of the release point locations from one another (refer to Figure 2.3-5).

The containment building cross-sectional area along with the containment building height is used for the annual average χ/Q calculations (used as input to develop the accident χ/Q values at the LPZ using Regulatory Guide 1.145 methodology). The applicable methodology for the annual average χ/Q calculations is identified in

Regulatory Guide 1.111, Revision 1, Regulatory Position C.1.c (Reference 28). These annual average χ/Q values are used to calculate the intermediate averaging time χ/Q values for the periods of 2-8 hours, 8-24 hours, 1-4 days, and 4-30 days by logarithmic interpolation.

The following conservative assumptions are made for these calculations:

- Releases are treated as point sources;
- Releases are treated as ground-level as there are no release conditions that are sufficiently high to escape the aerodynamic effects of the plant buildings;
- The distances from the Unit 1 and Unit 2 releases are determined from the closest edge of the containment buildings to the EAB;
- The plume centerline from each release is transported directly over the receptor; and
- A terrain recirculation factor of 4 is used in the calculation of the annual average χ/Q values
- ~~and no~~ Rradioactive decay or plume depletion due to deposition is not considered.

The highest EAB and LPZ χ/Q values from among all 22.5°-downwind sectors for each release/receptor combination and accident period are summarized in Table 2.3-145. EAB χ/Q values are presented for releases from Unit 1 and Unit 2, while the LPZ χ/Q values are applicable to both units. The 0.5% sector dependent χ/Q values are presented with the worst case downwind sector indicated in parentheses.

2.3.5.2.2 On-Site Atmospheric Dispersion Factors

The control room and technical support center χ/Q values for radiological releases from Unit 1 and Unit 2 are calculated using the NRC "**A**tmospheric **R**elative **CON**centrations in Building Wakes" (ARCON96) methodology as documented in NUREG/CR-6331, Revision 1 (Reference 29). Input data consist of: hourly on-site meteorological data; release characteristics (e.g., release height, building area affecting the release); and various receptor parameters (e.g., distance and direction from release to control room air intake and intake height). Refer to Section 15.5.8.11 for a description of computer program ARCON96).

A continuous temporally representative 5-year period of hourly on-site meteorological data from the DCPP onsite meteorological tower (i.e., January 1, 2007 through December 31, 2011) is used for the ARCON96 analysis. Each hour of data, at a minimum, has a validated wind speed and direction at the 10-meter level and a temperature difference between the 76- and 10-meter levels. This period of data is temporally representative and meets the requirements of Safety Guide 23, February 1972 (Reference 21).

The ARCON96 modeling follows the ground level release requirements of Regulatory Position C.3 of Regulatory Guide 1.194, June 2003 (Reference 30) relative to

determination of: (1) release height (i.e., ground-level vs. elevated); (2) release type (i.e., diffuse vs. point); and (3) configuration of release points and receptors (i.e., building cross-sectional area, release heights, line-of-sight distance between release and receptor locations, initial diffusion coefficients etc.).

Releases are assumed to be ground-level as none of the release points meet the definition of an elevated release as required by Regulatory Position C.3.2.2 of Regulatory Guide 1.194, June 2003 (i.e., do not meet the requirement to be at a minimum 2.5 times the height of plant buildings).

Only the containment building edge releases are treated as diffuse sources as the releases occur from the entire surface of the building. In these cases, initial values of the diffusion coefficients (sigma y, sigma z) are determined in accordance with the requirements in Regulatory Guide 1.194, June 2003 Regulatory Position C.3.2.4. Release and receptor locations are applied in accordance with Regulatory Guide 1.194, June 2003 Regulatory Position C.3.4 requirements for building geometry and line-of-site distances (refer to Figure 2.3-5).

The following recommended default values from Regulatory Guide 1.194, June 2003, Table A-2, are judged to be applicable to DCPP:

Wind direction range = 90 degrees azimuth;

Wind speed assigned to calm = 0.5 m/sec;

Surface roughness length = 0.20 m; and

Sector averaging constant = 4.3 (dimensionless)

The following assumptions are made for χ/Q calculations:

- The plume centerline from each release is transported directly over the control room or technical support center air intake/receptor (conservative);
- The distances from the Unit 1 and Unit 2 containment building surfaces to the receptors are determined from the closest edge of the containment buildings and the release/receptor elevation differences are set to zero (conservative);
- The applicable structure relative to quantifying building wake effects on the dispersion of the releases is based on release/receptor orientation relative to the plant structures;
- The releases from the Unit 1 and Unit 2 containment building surfaces are treated as diffuse sources;

- All releases are treated as ground level as there are no release conditions that merit categorization as an elevated release (i.e., 2.5 times containment building height) at this site (conservative); and

- The χ/Q value from the accident release point to the center of the control room boundary at roof level is utilized for control room in-leakage since the above χ/Q can be considered an average value for in-leakage locations around the control room envelope. The χ/Q from the accident release point to the center of the control room boundary at roof level is also utilized for control room ingress/egress. The outer doors to the control room are located at approximately the middle of a) the east side (i.e., auxiliary building side) wall of the control room and b) the west side (i.e., turbine building side) wall of the control room. Similarly, the χ/Q from the accident release point to the center of the TSC at its roof level is utilized for TSC in-leakage since the above χ/Q can be considered an average value for in-leakage locations around the TSC building envelope.

Summarized below are some of the other salient aspects of the control room and technical support center χ/Q analyses, as applicable.

<u>Control Room Receptors within 10-meters of Release</u>

Regulatory Guide 1.194, June 2003, Regulatory Position C.3.4 recommends that ARCON96 methodology not used for analysis at distances less than about 10 meters. However, as an exception to Regulatory Guide 1.194, June 2003, Regulatory Position C.3.4 the ARCON96 methodology has been applied for two cases when the distance from the release to the receptor is less than 10 meters. The distances in question (i.e., 9.4 meters for Unit 1 containment building to Unit 1 control room normal intake and 7.8 meters for Unit 2 containment building to Unit 2 control room normal intake) is considered acceptable since the dominating factors in the calculation are building cross-sectional area and plume meander, not the normal atmospheric dispersion coefficients.

<u>Control Room Receptors at 1.5-meters from Release</u>

Since the Unit 1 and Unit 2 MSSVs, 10% ADVs, and MSLB release points are located within 1.5 meters line-of-sight distance from the affected unit's control room normal intake, this near-field distance is considered outside of the ARCON96 application domain. Although ARCON96 is capable of estimating near-field dispersion, the 1.5-meter line-of-sight distance from the releases to the receptors is much less than the 10-meter distance recommended as the minimum applicable distance in Regulatory Position C.3.4 of Regulatory Guide 1.194, June 2003. Thus no χ/Qs are developed for the above release point / receptor combinations.

<u>Energetic Releases</u>

The vertical velocity of the MSSV and 10% ADVs releases is at least 95 times larger than the 95th percentile wind speed of 1 m/sec and approximately 5 times larger than the highest observed 10-meter wind speed (i.e., 18.9 m/sec) within the 5-year meteorological data base. The large vertical velocities of the MSSV and 10% ADVs releases, ranging from 94.9 to 98.9 m/sec, preclude any down-washing of the releases by the aerodynamic effects of the containment buildings such that the control room normal intake of the same unit as the release (e.g., Unit 1 MSSV/10% ADVs releases to Unit 1 CR normal intake) is not contaminated given that the horizontal distance is only 1.5 meters. Moreover, this short distance precludes the releases from reaching the control room normal intakes of the same unit given the height of the MSSV and 10% ADVs releases (i.e., 27.1 and 26.5 meters, respectively) relative to the height of the normal intakes (i.e., 22 meters). Plume rise calculations indicate that the MSSV and ADV release heights will be enhanced by 11 meters at the 95th percentile wind speed of 1 m/sec due to the large vertical velocities of the releases. Thus, for purposes of estimating dose consequences, it is appropriate to use the χ/Q associated with the normal control room intake of the opposite unit for releases from the MSSVs / 10% ADVs as the worst case control room intake location.

Vertically-Oriented Energetic Releases

Regulatory Position C.6 of Regulatory Guide 1.194, June 2003 establishes the use of a deterministic reduction factor of 5 applied to ARCON96 χ/Q values for energetic releases from steam relief valves or atmospheric dump valves. These valves must be uncapped and vertically-oriented and the time-dependent vertical velocity must exceed the 95th-percentile wind speed at the release point height by at least a factor of 5. Since the DCPP MSSVs and 10% ADVs are vertically oriented / uncapped and will have a vertical velocity of at least 94.9 m/sec for the first 10.73 hours of the accident, the reduction factor of 5 is clearly applicable to the DCPP MSSV and 10% ADVs releases. Note that since χ/Q values are averaged over the identified period (i.e., 0-2 hours, 2-8 hours, 8-24 hours, etc.), and the vertical velocity has been estimated to occur for 10.73 hours, application of the factor of 5 reduction is not appropriate for χ/Q values applicable to averaging periods beyond the 2-8 hours averaging period. For assessment of an environmental release between 8 to 10.73 hours, continued use of the 2-8 hour χ/Q, with the factor of 5 reduction, is acceptable and conservative.

Dual Intakes

The Unit 1 and Unit 2 control room pressurization air intakes which also serve the technical support center, may be considered dual intakes for the purpose of providing a low contamination intake regardless of wind direction for any of the release points since the two control room pressurization air intakes are never within the same wind direction window; defined as a wedge centered on the line of sight between the release and the receptor with the vertex located at the release point. The size of the wedge for each release-receptor combination is 90

degrees azimuth with the use of ARCON96, as described in Regulatory Position C.3.3.2 of Regulatory Guide 1.194, June 2003.

Redundant Radiation Monitors

Per Regulatory Guide 1.194, June 2003, Regulatory Position C.3.3.2.3, based on the dual intake design of the control room pressurization intakes, and the availability of redundant PG&E Design Class I radiation monitors at each pressurization intake (which provide the capability of initial selection of the cleaner intake and support the expectation that the operator will manually make the proper intake selection throughout the event), allows the χ/Q values applicable to the more favorable control room pressurization intake can be reduced by a factor of 4 and utilized to estimate the dose consequences.

PG&E Design Class II Lines Connecting to PG&E Design Class 1 Plant Vent

The 16 inch PG&E Design Class II gland seal steam exhauster line connects to the PG&E Design Class I plant vent. In addition, the plant vent expansion joint may experience a tear during a seismic event, however the plant vent will remain intact and functional.

a) The gland seal steam 16 inch exhauster line connects to the plant vent at El 144'-6" (Centerline) on the North-East side / South–East side of the Unit 1 and Unit 2 containments, respectively. It has been determined that should a failure occur due to a seismic event, it would occur at the interface of this line and the plant vent.
b) The plant vent expansion joint is located at El 155.83' North-East side / South–East side of the Unit 1 and Unit 2 containments, respectively. As discussed earlier, the plant vent expansion joint may experience a tear during a seismic event.

An assessment of the potential release locations identified above indicates that the χ/Q values developed for the plant vent are either conservative or representative of these potential release points.

Release points and receptor locations are provided in Figure 2.3-5, while Table 2.3-146 provides the release point / receptor combinations that were evaluated. Tables 2.3-147 and 2.3-148 provide the control room χ/Q values for the individual release point-receptor combinations for Unit 1 and Unit 2, respectively.

The χ/Q values selected for use in the dose consequence analyses are intended to support bounding analyses for an accident that occurs at either unit. They take into consideration the various release points-receptors applicable to each accident in order to identify the bounding χ/Q values and reflect the allowable adjustments and reductions in the values as discussed earlier and further summarized in the notes of Tables 2.3-147 and 2.3-148.

Table 2.3-149 presents the χ/Q values for the individual post-LOCA release point TSC receptor combinations for Unit 1 and Unit 2 applicable to the TSC normal intake and the center of the TSC boundary at roof level (considered an average value for potential TSC unfiltered in-leakage locations around the envelope). The Unit 1 and Unit 2 control room pressurization air intakes also serve the TSC during the emergency mode. Thus, the χ/Qs presented in Tables 2.3-147 and 2.3-148 for the control room pressurization intakes inclusive of the credit for dual intake design and ability to select the more favorable intake are also applicable to the TSC.

2.3.6 LONG-TERM (ROUTINE) DIFFUSION ESTIMATES

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

2.3.6.1 Objective

Annual relative concentrations (χ/Q) were estimated for distances out to 80 kilometers from onsite meteorological data for the period May 1973 through April 1975. These relative concentrations are presented in Table 2.3-2; they were estimated using the models described in Reference 18. The same program also produces cumulative frequency distributions for selected averaging periods using overlapping means having hourly updates. For critical offsite locations, measured lateral standard deviations of wind direction, σ_A, and bulk Richardson number, R_i, were used as the stability parameters in the computations. The meteorological input data were measured at the 10 meter level of the meteorological tower at DCPP site. Annual averaged relative concentrations calculated by the above methods are presented in Table 2.3-4.

2.3.6.2 Calculations

The meteorological instrumentation that was used to obtain the input data for the previously discussed relative concentration calculations at DCPP site is described in Section 2.3.4. Procedures for obtaining annual averaged relative concentrations are described in detail in Reference 15.

2.3.6.3 Meteorological Parameters

The following assumptions were used in developing the meteorological input parameters required in the dispersion model:

(1) There is no wind direction change with height

(2) Wind speed changes with height can be estimated by a power law function where the exponent, P, varies with stability class and is assigned the following values:

Pasquill Stability Class *Exponent (P)*

2.3.8.4 Safety Guide 23, February 1972 – Onsite Meteorological Programs

As discussed in Section 2.3.4, the ~~preoperational~~ meteorological data collection program was designed and has been updated continually to meet the requirements of Safety Guide 23, February 1972.

2.3.8.5 Regulatory Guide 1.97, Revision 3 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Wind speed, wind direction, and estimation of atmospheric stability indication in the control room provide information for use in determining the magnitude of the release of radioactive materials and in continuously assessing such releases during and following an accident (refer to Table 7.5-6 for a summary of compliance to Regulatory Guide 1.97, Revision 3).

2.3.8.6 Regulatory Guide 1.111, March 1976 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

The pre-operational values of dilution factor and deposition factor used in the calculation of annual average offsite radiation dose are discussed in Section 11.3.7. The values of deposition rate were derived from Figure 7 of Regulatory Guide 1.111, March 1976, for a ground-level release.

2.3.8.7 Regulatory Guide 1.111, Revision 1, July 1977 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

The annual average relative concentration values are developed for each sector, at the outer LPZ boundary distance for that sector, using the method described in Regulatory Position C.1.c of Regulatory Guide 1.111, Revision 1. These values are used to calculate the intermediate averaging time χ/Q values at the LPZ for the periods of 2-8 hours, 8-24 hours, 1-4 days, and 4-30 days following the postulated accident. This information is used as input to develop the accident χ/Q values at the LPZ using Regulatory Guide 1.145, Revision 1 methodology. Refer to Section 2.3.5.2.

2.3.8.8 Regulatory Guide 1.145, Revision 1, February 1983 – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

The short-term atmospheric dispersion factors applicable to the exclusion area boundary and the low population zone for post-accident accident releases from Unit 1 and Unit 2 are calculated using methodology applicable to "ground level" releases provided in Regulatory Guide 1.145, Revision 1. Refer to Section 2.3.5.2.

2.3.8.9 Regulatory Guide 1.194, June 2003 – Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

The control room and technical support center atmospheric dispersion factors for radiological releases from Unit 1 and Unit 2 are calculated using methodology outlined in Regulatory Positions C.1 through C.3, and the adjustment factor for vertically orientated energetic releases from steam relief valves and atmospheric dump valves allowed by Regulatory Position C.6, and NRC ARCON96 methodology as documented in NUREG/CR-6331, Revision 1. Refer to Section 2.3.5.2.

2.3.8.~~7~~10 NUREG-0737 (Item III.A.2), November 1980 – Clarification of TMI Action Plan Requirements

Item III.A.2 - Improving Licensee Emergency Preparedness–Long-Term:

As discussed in Section 2.3.4, the primary and backup meteorological data are available in the control room and emergency response facilities via the TRS servers and EARS, in accordance with NUREG-0654, Revision 1, November 1980.

As discussed in Section 2.3.4, the measurement subsystems consist of a primary meteorological tower and a backup meteorological tower. The primary meteorological computer and the backup meteorological computer communicate with each other, the EARS and also with the TRS server. Primary and backup meteorological data are available on the PPCs via the TRS servers and thus in the control room and emergency response facilities.

Item III.A.2.2 - Meteorological Data: NUREG-0737, Supplement 1, January 1983:

Table 7.5-6 and Section 2.3.8.5 summarize DCPP conformance with Regulatory Guide 1.97, Revision 3. Wind direction, wind speed, and estimation of atmospheric stability are categorized as Type E variables, based on Regulatory Guide 1.97, Revision 3. The PPC is used as the indicating device to display meteorological instrument signals. In addition, Type E, Category 3, recorders are located in the meteorological towers.

2.3.8.~~8~~11 IE Information Notice 84-91, December 1984 – Quality Control Problems of Meteorological Measurements Programs

In addition to the primary meteorological towers, a supplemental meteorological measurement system is provided in the vicinity of the plant site in order to meet IE Information Notice 84-91. As discussed in Section 2.3.4.5, this supplemental measurement system consists of three Doppler SODAR and seven tower sites located as indicated in Figure 2.3-4. The primary and secondary meteorological towers in conjunction with the supplemental system adequately predict the meteorological conditions at the site boundary (800 meters) and beyond.

24. ANSI/ANS 2.5, American National Standard for Determining Meteorological Information at Nuclear Power Sites, American Nuclear Society, 1984.

25. National Oceanic and Atmospheric Administration, An Evaluation of Wind Measurements by Four Doppler SODARS, NOAA Wave Propagation Laboratory, 1984.

26. Deleted in Revision 20.

27. PG&E reports previously submitted as Appendices 2.3A-K, 2.4A-C, and 2.5A-F of the FSAR Update, Revision 0 through Revision 10 (Currently maintained at PG&E Nuclear Power Generation Licensing office files).

28. Regulatory Guide 1.111, Revision 1, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors, USNRC.

29. Ramsdell, J. V. Jr. and C. A. Simonen, Atmospheric Relative Concentrations in Building Wakes. Prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Revision 1, May 1997.

30. Regulatory Guide 1.194, June 2003, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, USNRC.

TABLE 2.3-145
EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE ATMOSPHERIC DISPERSION FACTORS

Receptor	χ/Q (sec/m^3) 0 - 2 hours	2 - 8 hours	8 - 24 hours	1 - 4 days	4 - 30 days
Unit 1 EAB (NW)	2.50E-04	-	-	-	-
Unit 2 EAB (SSE)	2.30E-04	-	-	-	-
Unit 1/2 LPZ NW)	2.12E-05	9.26E-06	6.26E-06	2.67E-06	7.86E-07

Notes:

1: An EAB χ/Q value of 2.5E-04 sec/m^3 is used for radiological dose calculations from all release points.

2. The 0.5% sector dependent χ/Q values are presented with the worst case downwind sector indicated in parentheses.

TABLE 2.3-146
ON-SITE ATMOSPHERIC DISPERSION FACTOR EVALUATION
POST-ACCIDENT RELEASE POINT / RECEPTOR COMBINATIONS

Release Points

1. Unit 1 Containment Building Edge
2. Unit 2 Containment Building Edge
3. Unit 1 Plant Vent
4. Unit 2 Plant Vent
5. Unit 1 Refueling Water Storage Tank (RWST) Vent[1]
6. Unit 2 RWST Vent[1]
7. Unit 1 Containment Penetration (GE Area)
8. Unit 2 Containment Penetration (GE Area)
9. Unit 1 Containment Penetration (GW/FW Area)
10. Unit 2 Containment Penetration (GW/FW Area)
11. Unit 1 Fuel Handling Building
12. Unit 2 Fuel Handling Building
13. Unit 1 Equipment Hatch
14. Unit 2 Equipment Hatch
15. Unit 1 Main Steam Safety Valves (MSSVs)
16. Unit 2 MSSVs
17. Unit 1 10% Atmospheric Dump Valves
18. Unit 1 10% Atmospheric Dump Valves
19. Unit 1 Main Steam Line Break Location
20. Unit 2 Main Steam Line Break Location

On-Site Receptors

1. Unit 1 Control Room Normal Intake
2. Unit 2 Control Room Normal Intake
3. Unit 1 Control Room Emergency Intake
4. Unit 2 Control Room Emergency Intake
5. Control Room Center (i.e., In-leakage)
6. TSC Normal Intake[2]
7. TSC Center[2](i.e., In-leakage)

Notes:

1. χ/Q values for RWST releases to the control room normal intakes are not needed for the dose calculations since the normal intakes are isolated prior to releases occurring from the RWST vent.
2. χ/Q values developed only for the LOCA (i.e., release points 1 through 10).

TABLE 2.3-146A
ON-SITE ATMOSPHERIC DISPERSION FACTOR EVALUATION
POST-ACCIDENT RELEASE POINT & RECEPTOR LOCATION

ID[1] (See Figure 2.3-5)	Release/Receptor	Description
Note 2	Release Point	Unit 1 Containment Building (CB) edge
Note 2	Release Point	Unit 2 Containment Building (CB) edge
1	Release Point	U1 Plant Vent
2	Release Point	U2 Plant Vent
3	Receptor	U1 Control Room Normal Intake
4	Receptor	U2 Control Room Normal Intake
5	Receptor	U1 Control Room Emergency Intake
6	Receptor	U2 Control Room Emergency Intake
7	Release Point	U1 RWST Vent
8	Release Point	U2 RWST Vent
9	Receptor	Control Room Center (location assigned for unfiltered inleakage)
10	Release Point	Unit 1 Containment Penetration Area, GE
11	Release Point	Unit 2 Containment Penetration Area, GE
12	Release Point	Unit 1 Containment Penetration Area, FW/GW
13	Release Point	Unit 2 Containment Penetration Area, FW/GW
14	Release Point	U1 Fuel Handling Building
15	Release Point	U2 Fuel Handling Building
16	Release Point	U1 Equipment Hatch
17	Release Point	U2 Equipment Hatch
18	Release Point	U1 MSSV
19	Release Point	U2 MSSV
20	Release Point	U1 10% ADVs
21	Release Point	U2 10% ADVs
22	Release Point	U1 MSL Break location
23	Release Point	U2 MSL Break location
24	Receptor	TSC Normal Intake
25	Receptor	TSC Center (location assigned for unfiltered inleakage)

Note 1: Refer to Figure 2.3-5 for location of the above release points / receptors on the site layout and arrangement drawing

Note 2: Though not depicted in Figure 2.3-5, atmospheric dispersion factors were also calculated from the closest edge of the containment building to the various receptors; this release point was treated as a diffuse source.

TABLE 2.3-147
UNIT 1 CONTROL ROOM INTAKE AND CENTER ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release Point and Receptor	0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
Unit 1 Containment Building Edge to Unit 1 Control Room (CR) Normal Intake	1.28E-03	7.12E-04	2.87E-04	2.90E-04	2.84E-04
Unit 1 Containment Building Edge to Unit 2 CR Normal Intake	6.52E-04	3.51E-04	1.51E-04	1.49E-04	1.37E-04
Unit 1 Containment Building Edge to Unit 1 CR Emergency Intake[4]	4.11E-04	2.30E-04	9.62E-05	8.69E-05	7.03E-05
Unit 1 Containment Building Edge to Unit 2 CR Emergency Intake[4]	1.67E-04	7.95E-05	2.63E-05	2.81E-05	2.34E-05
Unit 1 Containment Building Edge to CR Center	8.85E-04	4.43E-04	1.75E-04	1.77E-04	1.65E-04
Unit 1 Plant Vent to Unit 1 CR Normal Intake	1.67E-03	1.22E-03	4.90E-04	4.90E-04	4.44E-04
Unit 1 Plant Vent to Unit 2 CR Normal Intake	9.10E-04	6.57E-04	2.68E-04	2.62E-04	2.45E-04
Unit 1 Plant Vent to Unit 1 CR Emergency Intake[4]	5.59E-04	3.38E-04	1.32E-04	1.12E-04	8.38E-05
Unit 1 Plant Vent to Unit 2 CR Emergency Intake[4]	2.26E-04	1.48E-04	5.40E-05	5.47E-05	4.45E-05
Unit 1 Plant Vent to CR Center	1.26E-03	8.96E-04	3.44E-04	3.44E-04	2.99E-04
Unit 1 Containment Penetration (GE Area) to Unit 1 CR Normal Intake	6.84E-03	3.08E-03	1.21E-03	1.12E-03	8.75E-04
Unit 1 Containment Penetration (GE Area) to Unit 2 CR Normal Intake	2.24E-03	1.15E-03	3.98E-04	3.89E-04	3.20E-04
Unit 1 Containment Penetration (GE Area) to Unit 1 CR Emergency Intake[4]	3.75E-04	2.33E-04	9.12E-05	8.45E-05	6.62E-05
Unit 1 Containment Penetration (GE Area) to Unit 2 CR Emergency Intake[4]	2.55E-04	1.25E-04	4.42E-05	4.38E-05	3.55E-05
Unit 1 Containment Penetration (GE Area) to CR Center	3.22E-03	1.42E-03	5.54E-04	5.20E-04	4.21E-04
Unit 1 Containment Penetration (GW/FW Area) to Unit 1 CR Normal Intake	4.90E-03	3.45E-03	1.37E-03	1.37E-03	1.28E-03
Unit 1 Containment Penetration (GW/FW Area) to Unit 2 CR Normal Intake	1.38E-03	9.83E-04	3.92E-04	3.88E-04	3.65E-04
Unit 1 Containment Penetration (GW/FW Area) to Unit 1 CR Emergency Intake[4]	8.20E-04	5.40E-04	2.15E-04	1.87E-04	1.43E-04
Unit 1 Containment Penetration (GW/FW Area) to Unit 2 CR Emergency Intake[4]	2.58E-04	1.54E-04	4.95E-05	5.26E-05	4.48E-05
Unit 1 Containment Penetration (GW/FW Area) to CR Center	2.59E-03	1.81E-03	7.29E-04	7.15E-04	6.64E-04
Unit 1 RWST Vent to Unit 1 CR Emergency Intake[4,5]	3.27E-04	1.90E-04	7.13E-05	6.99E-05	5.76E-05
Unit 1 RWST Vent to Unit 2 CR Emergency Intake[4,5]	2.10E-04	9.83E-05	3.73E-05	3.53E-05	2.86E-05
Unit 1 RWST Vent to CR Center[5]	1.07E-03	4.86E-04	1.99E-04	1.75E-04	1.43E-04

TABLE 2.3-147 (Continued)
UNIT 1 CONTROL ROOM INTAKE AND CENTER ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release Point and Receptor	0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
Unit 1 MSSVs to Unit 1 CR Normal Intake [1, 2]	N/A	N/A	N/A	N/A	N/A
Unit 1 MSSVs to Unit 2 CR Normal Intake [3]	4.29E-03	2.76E-03	1.04E-03	1.06E-03	9.46E-04
Unit 1 MSSVs to Unit 1 CR Emergency Intake [3, 4]	4.66E-04	2.92E-04	1.16E-04	1.04E-04	8.08E-05
Unit 1 MSSVs to Unit 2 CR Emergency Intake[3, 4]	3.14E-04	1.53E-04	5.12E-05	5.29E-05	4.38E-05
Unit 1 MSSVs to CR Center[3]	1.39E-02	7.40E-03	2.38E-03	2.56E-03	2.15E-03
Unit 1 10% ADVs to Unit 1 CR Normal Intake[1, 2]	N/A	N/A	N/A	N/A	N/A
Unit 1 10% ADVs to Unit 2 CR Normal Intake[3]	4.30E-03	2.79E-03	1.05E-03	1.06E-03	9.49E-04
Unit 1 10% ADVs to Unit 1 CR Emergency Intake[3, 4]	4.66E-04	2.92E-04	1.16E-04	1.04E-04	8.07E-05
Unit 1 10% ADVs to Unit 2 CR Emergency Intake[3, 4]	3.13E-04	1.54E-04	5.13E-05	5.30E-05	4.39E-05
Unit 1 10% ADVs to CR Center[3]	1.39E-02	7.45E-03	2.39E-03	2.59E-03	2.15E-03
Unit 1 MSL Break Location to Unit 1 CR Normal Intake[1]	N/A	N/A	N/A	N/A	N/A
Unit 1 MSL Break Location to Unit 2 CR Normal Intake	4.23E-03	2.90E-03	1.13E-03	1.11E-03	1.02E-03
Unit 1 MSL Break Location to Unit 1 CR Emergency Intake[4]	4.35E-04	2.94E-04	1.15E-04	1.01E-04	7.76E-05
Unit 1 MSL Break Location to Unit 2 CR Emergency Intake[4]	3.06E-04	1.54E-04	5.19E-05	5.32E-05	4.38E-05
Unit 1 MSL Break Location to CR Center	1.24E-02	7.10E-03	2.24E-03	2.43E-03	2.07E-03
Unit 1 FHB to Unit 1 CR Normal Intake	6.98E-03	-	-	-	-
Unit 1 FHB to Unit 2 CR Normal Intake	2.93E-03	-	-	-	-
Unit 1 FHB to Unit 1 CR Emergency Intake[4]	3.31E-04	-	-	-	-
Unit 1 FHB to Unit 2 CR Emergency Intake[4]	2.56E-04	-	-	-	-
Unit 1 FHB to CR Center	3.78E-03	-	-	-	-
Unit 1 Equipment Hatch to Unit 1 CR Normal Intake	2.61E-02	-	-	-	-
Unit 1 Equipment Hatch to Unit 2 CR Normal Intake	2.88E-03	-	-	-	-
Unit 1 Equipment Hatch to Unit 1 CR Emergency Intake[4]	4.36E-04	-	-	-	-
Unit 1 Equipment Hatch to Unit 2 CR Emergency Intake[4]	2.64E-04	-	-	-	-
Unit 1 Equipment Hatch to CR Center	5.51E-03	-	-	-	-

Notes (Refer to Section 2.3.5.2 for additional detail):

1. ARCON96 based χ/Q s are not applicable for these cases given that the horizontal distance from the source to the receptor is 1.5 meters (which is much less than the 10 meters required by ARCON96 methodology).

2. Due to the proximity of the release from the MSSVs/10% (ADVs, to the normal operation control room intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the control room normal intake of the affected unit.

3. For releases from the MSSVs and 10% ADVs (they are uncapped / vertically oriented, have a high vertical velocity discharge for the first 10.73 hours of the accident), a χ/Q reduction factor of 5 is applicable to the values listed above. Since χ/Q values are averaged over the identified period (i.e., 0-2 hours, 2-8 hours, 8-24 hours, etc.), and the vertical velocity has been estimated only up to 10.73 hours, application of the factor of 5 reduction is not appropriate for χ/Q values applicable to averaging periods beyond the 2-8 hours averaging period. For assessment of an environmental release between T= 8 to 10.73 hours, continued use of the 2-8 hour χ/Q with the factor of 5 reduction is acceptable and slightly conservative.

4. The more favorable χ/Q value presented above for the control room pressurization Intakes is further reduced by a factor of 4 to address the "dual intake" credit and the capability of initial selection of the cleaner intake and expectation that the operator will manually make the proper intake selection throughout the event.

5. χ/Q values for RWST releases to the control room normal intakes are not needed for the dose calculations since the normal intakes are isolated prior to releases occurring from the RWST vent.

TABLE 2.3-148
UNIT 2 CONTROL ROOM INTAKE AND CENTER ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release Point and Receptor	0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
Unit 2 Containment Edge to Unit 2 CR Normal Intake	1.96E-03	9.42E-04	4.48E-04	3.98E-04	3.18E-04
Unit 2 Containment Edge to Unit 1 CR Normal Intake	6.93E-04	3.84E-04	1.67E-04	1.42E-04	1.08E-04
Unit 2 Containment Edge to Unit 1 CR Emergency Intake[4]	1.70E-04	1.06E-04	4.23E-05	3.81E-05	2.95E-05
Unit 2 Containment Edge to Unit 2 CR Emergency Intake[4]	3.85E-04	1.47E-04	5.94E-05	5.84E-05	4.84E-05
Unit 2 Containment Edge to CR Center	1.08E-03	5.46E-04	2.47E-04	2.12E-04	1.68E-04
Unit 2 Plant Vent to Unit 1 CR Normal Intake	1.51E-03	9.41E-04	3.86E-04	3.23E-04	2.23E-04
Unit 2 Plant Vent to Unit 1 CR Normal Intake	7.88E-04	4.86E-04	2.01E-04	1.69E-04	1.17E-04
Unit 2 Plant Vent to Unit 1 CR Emergency Intake[4]	2.03E-04	1.29E-04	5.13E-05	4.32E-05	3.19E-05
Unit 2 Plant Vent to Unit 2 CR Emergency Intake[4]	5.71E-04	2.96E-04	1.20E-04	1.04E-04	8.19E-05
Unit 2 Plant Vent to CR Center	1.13E-03	7.08E-04	2.85E-04	2.39E-04	1.70E-04
Unit 2 Containment Penetration (GE Area) to Unit 2 CR Normal Intake	6.71E-03	3.12E-03	1.21E-03	1.22E-03	1.02E-03
Unit 2 Containment Penetration (GE Area) to Unit 1 CR Normal Intake	2.14E-03	1.39E-03	5.72E-04	4.83E-04	3.62E-04
Unit 2 Containment Penetration (GE Area) to Unit 1 CR Emergency Intake[4]	2.28E-04	1.60E-04	6.25E-05	5.52E-05	4.21E-05
Unit 2 Containment Penetration (GE Area) to Unit 2 CR Emergency Intake[4]	3.97E-04	1.76E-04	6.93E-05	6.44E-05	5.27E-05
Unit 2 Containment Penetration (GE Area) to CR Center	3.16E-03	1.85E-03	7.17E-04	6.84E-04	5.43E-04
Unit 2 Containment Penetration (GW/FW Area) to Unit 2 CR Normal Intake	3.55E-03	1.19E-03	4.82E-04	4.56E-04	3.03E-04
Unit 2 Containment Penetration (GW/FW Area) to Unit 1 CR Normal Intake	1.22E-03	6.26E-04	2.53E-04	2.12E-04	1.41E-04
Unit 2 Containment Penetration (GW/FW Area)to Unit CR Emergency Intake[4]	2.28E-04	1.62E-04	6.58E-05	5.43E-05	3.99E-05
Unit 2 Containment Penetration (GW/FW Area)to Unit 2 CR Emergency Intake[4]	8.64E-04	4.23E-04	1.50E-04	1.48E-04	1.20E-04
Unit 2 Containment Penetration (GW/FW Area)to CR Center	2.21E-03	1.17E-03	4.70E-04	3.90E-04	2.61E-04
Unit 2 RWST Vent to Unit 1 CR Emergency Intake[4,5]	1.91E-04	1.21E-04	4.58E-05	4.39E-05	3.53E-05
Unit 2 RWST Vent to Unit 2 CR Emergency Intake[4,5]	3.29E-04	1.61E-04	6.10E-05	5.53E-05	4.45E-05
Unit 2 RWST Vent to CR Center[5]	1.07E-03	5.80E-04	2.18E-04	2.19E-04	1.79E-04

TABLE 2.3-148 (Continued)
UNIT 2 CONTROL ROOM INTAKE AND CENTER ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release Point and Receptor	0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
Unit 2 MSSVs to Unit 1 CR Normal Intake [3]	3.87E-03	2.42E-03	9.89E-04	8.17E-04	6.09E-04
Unit 2 MSSVs to Unit 2 CR Normal Intake[1,2]	N/A	N/A	N/A	N/A	N/A
Unit 2 MSSVs to Unit 1 CR Emergency Intake[3,4]	2.89E-04	1.91E-04	7.45E-05	6.62E-05	5.08E-05
Unit 2 MSSVs to Unit 2 CR Emergency Intake[3,4]	4.90E-04	2.29E-04	8.24E-05	8.07E-05	6.49E-05
Unit 2 MSSVs to CR Center[3]	1.22E-02	8.10E-03	3.27E-03	2.76E-03	2.08E-03
Unit 2 10% ADVs to Unit 1 CR Normal Intake[3]	3.88E-03	2.43E-03	9.94E-04	8.19E-04	6.10E-04
Unit 2 10% ADVs to Unit 2 CR Normal Intake[1,2]	N/A	N/A	N/A	N/A	N/A
Unit 2 10% ADVs to Unit 1 CR Emergency Intake[3,4]	2.87E-04	1.92E-04	7.48E-05	6.61E-05	5.07E-05
Unit 2 10% ADVs to Unit 2 CR Emergency Intake[3,4]	4.90E-04	2.29E-04	8.24E-05	8.08E-05	6.48E-05
Unit 2 10% ADVs to CR Center[3]	1.22E-02	8.16E-03	3.28E-03	2.78E-03	2.09E-03
Unit 2 MSL Break Location to Unit 1 CR Normal Intake	3.81E-03	2.40E-03	1.01E-03	8.09E-04	5.88E-04
Unit 2 MSL Break Location to Unit 2 CR Normal Intake[1]	N/A	N/A	N/A	N/A	N/A
Unit 2 MSL Break Location to Unit 1 CR Emergency Intake[4]	2.75E-04	1.91E-04	7.45E-05	6.53E-05	4.86E-05
Unit 2 MSL Break Location to Unit 2 CR Emergency Intake[4]	4.76E-04	2.24E-04	8.14E-05	7.94E-05	6.40E-05
Unit 2 MSL Break Location to CR Center	1.09E-02	7.35E-03	3.01E-03	2.48E-03	1.86E-03
Unit 2 FHB to Unit 1 CR Normal Intake	2.72E-03	-	-	-	-
Unit 2 FHB to Unit 2 CR Normal Intake	6.98E-03	-	-	-	-
Unit 2 FHB to Unit 1 CR Emergency Intake[4]	2.49E-04	-	-	-	-
Unit 2 FHB to Unit 2 CR Emergency Intake[4]	3.50E-04	-	-	-	-
Unit 2 FHB to CR Center	3.71E-03	-	-	-	-
Unit 2 Equipment Hatch to Unit 1 CR Normal Intake	2.49E-03	-	-	-	-
Unit 2 Equipment Hatch to Unit 2 CR Normal Intake	2.51E-02	-	-	-	-
Unit 2 Equipment Hatch to Unit 1 CR Emergency Intake[4]	2.49E-04	-	-	-	-
Unit 2 Equipment Hatch to Unit 2 CR Emergency Intake[4]	4.68E-04	-	-	-	-
Unit 2 Equipment Hatch to CR Center	5.19E-03	-	-	-	-

Notes (Refer to Section 2.3.5.2 for additional detail):

1 ARCON96 based χ/Qs are not applicable for these cases given that the horizontal distance from the source to the receptor is 1.5 meters (which is much less than the 10 meters required by ARCON96 methodology).

2 Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation control room intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the control room normal intake of the affected unit.

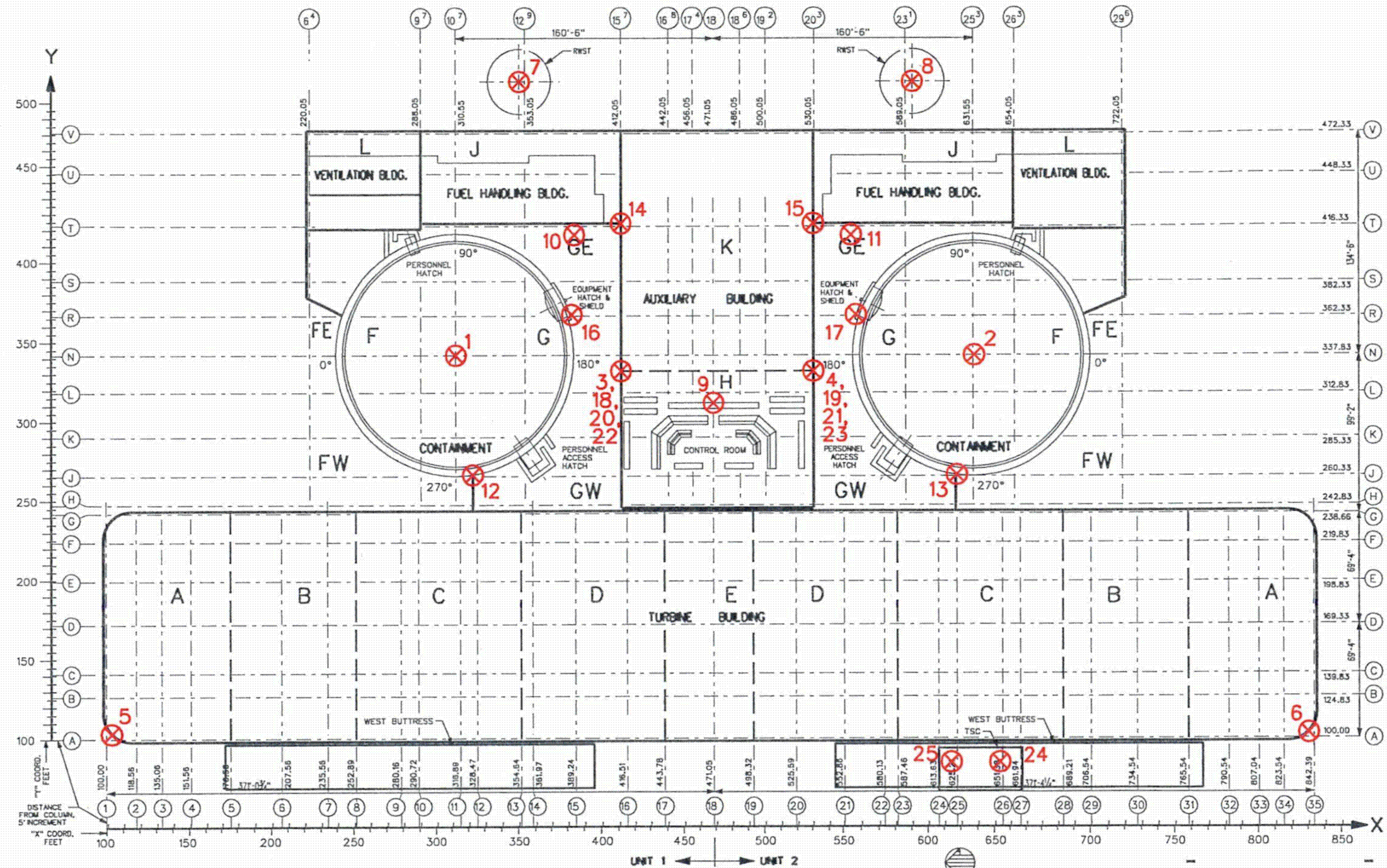
3 For releases from the MSSVs and 10% ADVs (they are uncapped / vertically oriented, have a high vertical velocity discharge for the first 10.73 hours of the accident), a χ/Q reduction factor of 5 is applicable to the values listed above. Since χ/Q values are averaged over the identified period (i.e., 0-2 hours, 2-8 hours, 8-24 hours, etc.), and the vertical velocity has been estimated only up to 10.73 hours, application of the factor of 5 reduction is not appropriate for χ/Q values applicable to averaging periods beyond the 2-8 hours averaging period. For assessment of an environmental release between T= 8 to 10.73 hours, continued use of the 2-8 hour χ/Q with the factor of 5 reduction is acceptable and slightly conservative.

4. The more favorable χ/Q value presented above for the control room pressurization Intakes is further reduced by a factor of 4 to address the "dual intake" credit and the capability of initial selection of the cleaner intake and expectation that the operator will manually make the proper intake selection throughout the event.

5. χ/Q values for RWST releases to the control room normal intakes are not needed for the dose calculations since the normal intakes are isolated prior to releases occurring from the RWST vent.

TABLE 2.3-149
UNITS 1 AND 2 TECHNICAL SUPPORT CENTER INTAKE AND CENTER ATMOSPHERIC DISPERSION FACTORS (SEC/M^3)

Release Point and Receptor	0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
UNIT 1					
Unit 1 Containment Building Edge to TSC Normal Intake	2.57E-04	1.18E-04	4.27E-05	4.24E-05	3.50E-05
Unit 1 Containment Building Edge to TSC Center	2.90E-04	1.33E-04	4.98E-05	4.83E-05	4.02E-05
Unit 1 Plant Vent to TSC Normal Intake	3.12E-04	1.77E-04	6.91E-05	6.29E-05	5.21E-05
Unit 1 Plant Vent to TSC Center	3.54E-04	1.95E-04	7.71E-05	6.70E-05	5.67E-05
Unit 1 RWST Vent to TSC Normal Intake	2.72E-04	1.27E-04	4.80E-05	4.49E-05	3.71E-05
Unit 1 RWST Vent to TSC Center	2.94E-04	1.38E-04	5.40E-05	4.89E-05	3.97E-05
Unit 1 Containment Penetration (GE Area) to TSC Normal Intake	3.64E-04	1.74E-04	6.55E-05	6.14E-05	5.00E-05
Unit 1 Containment Penetration (GE Area) to TSC Center	4.27E-04	1.91E-04	7.45E-05	6.84E-05	5.62E-05
Unit 1 Containment Penetration (GW/FW Area) to TSC Normal Intake	4.80E-04	2.51E-04	8.31E-05	8.64E-05	6.95E-05
Unit 1 Containment Penetration (GW/FW Area) to TSC Center	5.98E-04	3.03E-04	1.04E-04	1.03E-04	8.46E-05
UNIT 2					
Unit 2 Containment Building Edge to TSC Normal Intake	5.48E-04	2.00E-04	8.52E-05	8.37E-05	6.84E-05
Unit 2 Containment Building Edge to TSC Center	5.57E-04	2.01E-04	8.81E-05	8.89E-05	6.92E-05
Unit 2 Plant Vent to TSC Normal Intake	5.52E-04	2.35E-04	1.06E-04	8.71E-05	6.95E-05
Unit 2 Plant Vent to TSC Center	5.43E-04	2.16E-04	9.97E-05	8.11E-05	6.58E-05
Unit 2 RWST Vent to TSC Normal Intake	3.63E-04	1.68E-04	6.47E-05	6.04E-05	4.91E-05
Unit 2 RWST Vent to TSC Center	3.72E-04	1.68E-04	6.64E-05	6.17E-05	5.10E-05
Unit 2 Containment Penetration (GE Area) to TSC Normal Intake	5.47E-04	2.41E-04	9.36E-05	8.83E-05	7.02E-05
Unit 2 Containment Penetration (GE Area) to TSC Center	5.72E-04	2.43E-04	9.75E-05	9.12E-05	7.52E-05
Unit 2 Containment Penetration (GW/FW Area) to TSC Normal Intake	1.80E-03	7.72E-04	3.07E-04	2.87E-04	2.33E-04
Unit 2 Containment Penetration (GW/FW Area) to TSC Center	1.83E-03	7.49E-04	3.16E-04	2.92E-04	2.41E-04



COORDINATE-COLUMN LINE CROSS REFERENCE

3.1.4.1 Criterion 11, 1967 - Control Room (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Discussion

The plant is provided with a centralized control room common to both units that contains the controls and instrumentation necessary for operation of both units under normal and accident conditions.

The control room is continuously occupied by the operating personnel under all operating and accident conditions. Sufficient shielding, distance, and containment are provided to ensure that the control room personnel are not subject to radiation exposures in excess of 10 CFR 20 limits. Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem ~~whole body~~total effective dose equivalent (TEDE)~~, or its equivalent to any part of the body,~~ for the duration of the accident to meet the requirements of GDC 19, ~~1971~~1999 (refer to Section 3.1.4.1.1). Control room shielding is described in Section 12.1, and postaccident control room exposures are described in Section 15.5.

The control room ventilation system is described in Section 9.4.1. It consists of a dual system providing a large percentage of recirculated air. In the event of fire in the control room, provisions are made for 100 percent outside air makeup operation. In the event of airborne toxic gas outside the control room, provisions are made for operation with 100 percent recirculated air. In the event of airborne radioactivity outside the control room, provisions are made to isolate and pressurize the control room.

The risk of fire is minimized by the use of noncombustible and fire retardant materials in the construction of the control room and its furnishings. Fire fighting equipment is located in the control room, and the use and storage of combustible supplies are minimized.

Provisions are made to enable plant operators to readily shut down and maintain the plant at safe shutdown (Mode 3) by means of controls located outside the control room.

3.1.4.1.1 Criterion 19, ~~1971~~ 1999 – Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection

shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Holders of operating licenses using an alternative source term under 10 CFR 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.

~~A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.~~

Discussion

Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem ~~whole body, or its equivalent to any part of the body,~~TEDE for the duration of the accident to meet the requirements of GDC 19, ~~1971~~1999. ~~Refer to Section 6.4.~~The DCPP design includes use of alternative source terms in accordance with 10 CFR 50.67 and Regulatory Guide 1.183, July 2000. Refer to Section 6.4 and 15.5.

3.1.8.22 Criterion 58, 1967 - Inspection of Containment Pressure-Reducing Systems (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Discussion

The containment heat removal systems (CHRS) is comprised of the containment fan cooler system (CFCS) and the containment spray system (CSS). Where practicable, all active components and passive components of the ~~containment cooling system~~CHRS are inspected periodically to demonstrate system readiness. The pressure-containing systems are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves. During operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks. The containment fan coolers are normally in use, which provides an additional check on the readiness of the system. Five fan coolers are provided. Each is sized for one-quarter the capacity needed to maintain the containment temperature below 120°F during normal plant operation. Following a LOCA, two of the five fan coolers provide sufficient capacity to maintain containment pressure below design value when used in conjunction with one containment

spray pump during the injection phase. ~~During the recirculation phase, containment spray operation is not required.~~

Additional details are found in Section 6.2.

Note that there is no direct correlation between the design class of an SSC and the seismic design requirements. Refer to Section 3.2.2.4 for the discussion of seismic qualification requirements.

3.2.2.1.1 PG&E Design Class I

PG&E Design Class I is applicable to SSCs that are important to safety, including SSCs required to assure the following:

(a) The integrity of the reactor coolant pressure boundary.

(b) The capability to shut down the reactor and maintain it in a safe shutdown condition.

(c) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100 or 10 CFR 50.67.

(d) All plant features designated as PG&E Design Class I are designed to remain functional when subjected to the additional forces associated with the design basis earthquakes that they are required to withstand: the Design earthquake (DE), the Double Design earthquake (DDE) and/or the Hosgri earthquake (HE). Refer to Section 3.2.2.4 for additional information regarding the earthquake requirements for the design of SSCs.

The following SSCs, including their foundations and supports, are classified as PG&E Design Class I:

(1) The reactor coolant pressure boundary

(2) The reactor core and reactor vessel internals

(3) Systems (Refer to Notes (i) and (iii) below) or portions of systems that are required for emergency core cooling, post-accident containment heat removal, or post-accident containment atmosphere cleanup (Refer to Note (iv) below)

(4) Systems or portions of systems that are required for reactor shutdown and residual heat removal

(5) Those portions of the main steam, feedwater, and steam generator blowdown systems extending from and including the secondary side of the steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including

PG&E Design Class II or III may have some seismically qualified components (as defined in the Q-list) and have no PG&E Quality/Code Class designation. However, some PG&E Design Class II components have been seismically designed; e.g., items in the Seismically Induced Systems Interaction Program, specific components required for post-HE shutdown, (e.g. CCW header C components), and items that were designed for the DE (for specific SSC classifications refer to the Q-list). For this reason, there is not a direct correlation between PG&E Design Class and the level of seismic qualification (except that PG&E Design Class I SSCs have seismically qualified components). In addition, the classification of seismically qualified components does not indicate which of the three design basis earthquakes the SSC has been qualified for, nor whether that qualification is for passive or active function. The design basis function of the equipment determines the type of seismic qualification required. These classifications and their relationships are summarized in Tables 3.2-1 and 3.2-2.

The definitions of the three DCPP Earthquakes are included below:

Design Earthquake

The design earthquake (0.2g) is defined as the maximum size earthquake that can be expected to occur at DCPP during the life of the reactor. The design earthquake is the equivalent of the operating basis earthquake, as described in 10 CFR 100, Appendix A.

Double Design Earthquake

The double design earthquake (0.4g) is defined as the hypothetical earthquake that would produce accelerations twice those of the design earthquake. The double design earthquake is the equivalent of the safe shutdown earthquake, as described in 10 CFR 100, Appendix A.

Hosgri Earthquake

The Hosgri earthquake (0.75g) is defined as the predicted ground motion at DCPP due to a Richter magnitude 7.5 earthquake on the offshore Hosgri fault. The Hosgri earthquake does not correspond to an operating basis earthquake or safe shutdown earthquake.

3.2.2.5 PG&E Instrument System Classifications

DCPP Instrumentation is classified as Instrument Class IA, IB, IC, ID, and II depending upon the function performed. Instrument Class IA, IB, IC and ID devices have PG&E Design Class I functions or other special requirements. Instrument Class II devices have PG&E Design Class II functions. Instrument classes are defined as follows:

(1) *Instrument Class IA* - Class IA instruments and controls are those that initiate and maintain safe shutdown of the reactor, mitigate the consequences of an accident, or prevent exceeding 10 CFR Part 100 or

10 CFR 50.67 off-site dose limits. Class IA instruments and controls enable the PG&E Design Class I systems to automatically accomplish their appropriate safety functions, or they enable operating personnel to manually accomplish appropriate safety actions when a monitored condition reaches a preset level.

(2) *Instrument Class IB* - Class IB instruments and controls are those that are required for post-accident monitoring (PAM) of Category 1 and 2 variables in accordance with Regulatory Guide 1.97, Revision 3 (refer to Table 7.5-6). Regulatory Guide 1.97, Revision 3, Category 3 instruments required for PAM are Instrument Class II. PAM instrumentation enables operating personnel to:

(a) Determine when a condition monitored by PG&E Design Class I instruments and controls reaches a level requiring manual action

(b) Assess the accomplishment of plant safety functions

(c) Assess fission product barrier integrity

(d) Provide information to indicate the operability of individual systems important to safety

(e) Assess the magnitude of radioactive materials releases from the plant

PAM instruments are further divided into five variable types (A through E) as detailed in Regulatory Guide 1.97, Revision 3. These are further detailed in Section 7.5.

(3) *Instrument Class IC* - Instrument Class IC instruments and controls have the passive function of maintaining the pressure boundary integrity (PBI) of PG&E Design Class I piping systems. Passive valve operators that are within PG&E Design Class I piping systems are Instrument Class IC and are seismically analyzed to assure structural and pressure boundary integrity of the valve operator assembly. This classification also includes instruments installed in PG&E Design Class I HVAC ducting that are required to maintain pressure boundary integrity. In addition, this classification is used for instruments that are part of seismically qualified, PG&E Design Class II systems and denotes that the instruments are required to maintain their pressure boundary integrity.

(4) *Instrument Class ID* - Instrument Class ID instruments and controls are components that have certain PG&E Design Class I attributes, but do not require conformance with all Instrument Class IA, IB or IC requirements. The Instrument Class ID designation signifies that only certain design

Environmental Qualification Report (Revision 1) (Reference 14) to the NRC. In addition to the item-by-item NUREG-0588 comparison, this report consolidated and summarized extensive information concerning the development and implementation of the EQ program for DCPP. NRC review and acceptance of the DCPP EQ program was documented in Appendix B of SSER No. 15, dated September 1981 (Reference 15).

In 1983, the NRC issued 10 CFR 50.49 (Reference 16) to codify requirements for the environmental qualification of electric equipment important to safety. This regulation now forms the basis of the EQ program at DCPP. The regulation applies to the following three categories of equipment:

- Safety-related electric equipment ("Class 1E"): This equipment is that relied upon to remain functional during and following design basis events to ensure (a) the integrity of the reactor coolant pressure boundary, (b) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (c) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR 50.67 ~~100~~ guidelines. It is noted that DCPP has incorporated a full implementation of Alternative Source Terms (AST) as defined in Regulatory Guide 1.183, July 2000, Section 1.2.1. However, the post-LOCA integrated doses utilized for radiological environmental qualification of PG&E Design Class I equipment continue to be based on TID-14844 assumptions. This approach is acceptable based on Section 1.3.5 of Regulatory Guide 1.183 and NUREG-0933, Section 3.0, Item 187 which concluded that there is no clear basis for a requirement to modify the design basis for EQ to adopt AST since there would be no discernible risk reduction associated with adopting AST for EQ.

- Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of the safety functions specified above by the safety-related equipment.

- Certain postaccident monitoring equipment as described in RG 1.97, Revision 3 (Reference 17).

The regulation explicitly excludes equipment located in a mild environment. A mild environment is defined as an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

Also significant in 10 CFR 50.49 are the provisions in its paragraphs (k) and (l):

> "(k) Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section [i.e., in accordance with 10 CFR 50.49] if the Commission has previously required qualification of the equipment in

Steam generator tube inspection is performed in accordance with the Technical Specifications (Reference 6) and the DCPP surveillance test procedure. Eddy current non-destructive testing is used to perform tube inspections. The steam generator tube surveillance program ensures that the structural and leakage integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on NEI 97-06 (Reference 5). Inservice inspection of SG tubing is essential in order to maintain surveillance of the conditions of the tubes in the event there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of SG tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Tube degradation will be detected during scheduled inservice SG tube examinations. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20 percent of the original tube wall thickness. Plugging is required for all tubes with imperfections exceeding the plugging limit defined in the Technical Specifications. Degradation may be left in service if qualified non-destructive examination sizing techniques verify that the imperfection is less than the plugging limit (reference PG&E response to NRC Generic Letter 97-05).

5.5.2.5.2 Primary-to-Secondary Leakage

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the SG tubes. The extent of cracking during plant operation is limited by the limitation on SG tube leakage between the Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 150 gallons per day per SG). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. DCPP has demonstrated that primary-to-secondary leakage of 150 gallons per day per SG can readily be detected during power operation. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

The dose consequence analyses in Section 15.5.18.1 provides the radiological assessment for address accident-induced leakage up to 0.75 gpm (total for all 4 SGs)10.5 gpm at standard temperature and pressureroom temperature conditions in any one SG following an SLB.

5.5.3 REACTOR COOLANT PIPING

Reactor coolant piping provides a flowpath connecting the major components of each RCS loop.

5.5.3.1 Design Bases

from the containment atmosphere during the injection and recirculation phases. The CSS function of mixing the containment atmosphere for hydrogen control is discussed below.

(4) *The Containment Fan Cooler System (CFCS)*

The CFCS functions in conjunction with the CSS to limit the temperature and pressure in the containment structure in the event of a LOCA or MSLB (refer to Section 6.2.2). The CFCS also provides mixing of the sprayed and unsprayed regions of the containment atmosphere to improve airborne fission product removal (refer to Section 6.2.3). The CFCS function of mixing the containment atmosphere for hydrogen control is discussed below.

(5) *The Spray Additive System*

The SAS functions by adding sodium hydroxide, an effective iodine scrubbing solution, to the CSS water to reduce the content of iodine and other fission products in the containment atmosphere and prevent the re-evolution of the iodine in the recirculated containment spray and core cooling solution following a LOCA (refer to Section 6.2.3).

(6) *Containment Combustible Gas Control*

The long-term buildup of gaseous hydrogen in the containment following a LOCA is primarily controlled by ensuring a mixed containment atmosphere and providing equipment for monitoring hydrogen concentrations. The CFCS and the CSS are ~~is~~ the primary means credited for containment atmosphere mixing (refer to Section 6.2.5).

(7) *The Fuel Handling Building Ventilation System (FHBVS)*

The FHBVS provides the capability for ~~a~~ significant reduction in the amounts of volatile radioactive materials that could be released to the atmosphere in the event of a major fuel handling accident (refer to Section 9.4.4).

(8) *The Auxiliary Building Ventilation System(ABVS)*

The ABVS provides the capability for significant reductions in the amounts of volatile radioactive materials that could be released to the atmosphere in the event of leakage from the residual heat removal (RHR) system recirculation loop following a LOCA (refer to Section 9.4.2).

6.2.1.6 Materials

Containment structural heat sink materials used for containment integrity analyses following a LOCA or main steam line break are listed in Table 6.2D-19; corresponding material properties are listed in Table 6.2D-20 (refer to Section 6.2D.3.2.4). A current record of paint used on containment heat sink structures and equipment is maintained in engineering files.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

The containment heat removal systems (CHRS) are the containment fan cooler system (CFCS) and the containment spray system (CSS). The functional performance objectives of the CHRS are:

(1) The CFCS limits the containment ambient temperature during normal plant operating conditions (refer to Section 9.4.5);

(2) The CFCS and CSS reduce the containment ambient temperature and pressure following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) inside containment. While performing this cooling function, the CHRS also helps limit offsite radiation levels by reducing the pressure differential between containment and outside atmosphere, thus reducing the driving force for leakage of fission products from the containment atmosphere;

(3) The CFCS provides mixing of the sprayed and unsprayed regions of the containment to improve airborne fission product removal (refer to Section 6.2.3);

(4) The CSS removes airborne fission products from the containment atmosphere following a LOCA (refer to Section 6.2.3);

(5) The CSS, in conjunction with the spray additive system (SAS), prevents the re-evolution of the iodine in the recirculated containment spray and core cooling solution (i.e. sump water) following a LOCA (refer to Section 6.2.3);

(6) The CFCS and CSS provide~~s~~ a mixed atmosphere for hydrogen control (refer to Section 6.2.5).

Used in conjunction with one another during the injection phase, one containment spray pump and two containment fan cooler units (CFCUs) will provide the heat removal capability to maintain the post-accident containment atmospheric pressure and temperature below the design values of 47 psig and 271°F, respectively. The CFCS is credited for long-term containment pressure and temperature control throughout the injection and recirculation phases following a LOCA or MSLB. The operation of CSS in

the injection phase is credited ~~only for operation during the spray injection phase~~for heat removal following a LOCA or MSLB. To address the delayed radioactivity release associated with the core damage sequence of Alternate Source Terms (AST), credit is taken for CSS operation in both the injection and ~~following a LOCA or MSLB; it is not required for operation during the~~ recirculation phases to support fission product removal following a LOCA (refer to Section 15.5).~~ for mitigating the effects of a LOCA.~~ The physical SSC design bases, testing, and inspection requirements of the CFCS and CSS are discussed in this section.

6.2.2.1 Design Bases

6.2.2.1.1 General Design Criterion 2, 1967 - Performance Standards

The CFCS and CSS are designed to withstand the effects of, or are protected against, natural phenomena, such as earthquakes, flooding, tornadoes, winds, and other local site effects.

6.2.2.1.2 General Design Criterion 10, 1967 - Containment

The CFCS and CSS are designed to aid other ESFs in retaining the functional capability of the containment to protect the public in the event of gross equipment failures, such as a large coolant boundary break.

6.2.2.1.3 General Design Criterion 11, 1967 - Control Room

The CFCS and CSS are designed to support actions to maintain and control the safe operational status of the plant from the control room.

6.2.2.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

The CFCS and CSS are provided with instrumentation and controls as required to monitor and maintain the CHRS variables within prescribed operating ranges.

6.2.2.1.5 General Design Criterion 15, 1967 - Engineered Safety Features Protection Systems

The CFCS and CSS are provided with instrumentation for sensing accident conditions.

6.2.2.1.6 General Design Criterion 19, ~~1971~~ 1999 - Control Room

The CFCS and CSS, in conjunction with the SAS, are designed to limit radiation exposure to personnel to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, ~~or its equivalent to any part of the body,~~for the duration of the accident.

6.2.2.2 System Description

The CHRS is designed to provide sufficient heat removal capability to maintain the post-accident containment atmospheric pressure and temperature below the design values of 47 psig and 271°F, respectively (refer to Section 3.8.1.1). The containment atmospheric temperature after a MSLB does briefly increase greater than the containment design temperature, but, as discussed in Section 6.2D.4.2.4, there is no explicit design temperature limit for a MSLB. Heat energy sources considered are described in Section 6.2D.2.1.12.

The CFCS also functions during normal operating conditions to limit the containment ambient temperature to 120°F and is described in Section 9.4.5.

The CFCS, shown schematically in Figure 9.4-4, consists of five identical fan coolers, each including cooling coils, fan and drive motor, locked-open air flow dampers and pressure relief dampers, duct distribution system, instrumentation, and control. During operation of the units, air is drawn into the cooling coils, cooled, and discharged back through the ductwork to the containment atmosphere.

The design parameters for the CHRS components and materials are listed in Table 6.2-26. Codes and standards used as a basis for the design of the components are given in Table 6.2-25.

Ductwork distributes the cooled air to the various containment compartments and areas. During normal and post-accident operations, the flow sequence through each air fan cooler is as follows: locked-open normal and accident air flow dampers, cooling coils, fan, and distribution ductwork.

Airflow through the exhaust ducting, towards the fan, can occur when the fan is idle. Incorporated into the fan/motor coupling is an anti-reverse rotation device (AARD) that precludes the fan motor from rotating backwards. This device replaces backdraft dampers previously installed in the fan discharge duct.

The CSS, shown schematically in Figure 3.2-12, consists of two pumps, spray ring headers and nozzles, valves, and connecting piping. Following a LOCA, water from the refueling water storage tank (RWST) is initially used for containment spray. Later, water recirculated from the containment sump ~~can be~~is supplied by the residual heat removal (RHR) pumps for recirculation spray. ~~If no component failures affect the RHR train capability, the emergency procedures direct the initiation of recirculation sprays. However, s~~Single failures that result in the loss of one RHR train ~~cause the decision of how to~~are taken into consideration in system alignment to ensure appropriate division of ~~divide~~ the recirculation flow between containment spray and core injection to ensure fission product removal from the containment atmosphere and core cooling. ~~to be made by the technical support center in charge of accident mitigation.~~

(2) The CCW exit temperatures are indicated locally outside the containment building.

(3) Bearing temperatures are indicated and alarmed on the plant process computer.

6.2.2.3.5 General Design Criterion 15, 1967 - Engineered Safety Features Protection Systems

The CHRS is designed with instrumentation that measures containment pressure. The CSS will be actuated through ESFAS by a "P" signal (high-high containment pressure actuation setpoint) on coincidence of two-out-of-four high-high containment pressure signals. Coincidence of "S" (safety injection) and "P" signals starts the containment spray pumps and opens the discharge valves to the spray headers. Refer to Section 7.3 for a description of the ESFAS.

6.2.2.3.6 General Design Criterion 19, ~~1971~~1999- Control Room

The CSS is designed to deliver borated water from the RWST to the containment atmosphere during the injection phase following a LOCA. The CSS, in conjunction with the CFCS, reduces the containment ambient temperature and pressure following a LOCA or MSLB, thus reducing the driving force for leakage of fission products from the containment atmosphere. The CFCS provides mixing of the sprayed and unsprayed regions of the containment to improve airborne fission product removal. The SAS, as discussed in Section 6.2.3, injects sodium hydroxide to the suction of the CSS for delivery to the containment atmosphere to prevent the re-evolution of the iodine in the recirculated core cooling solution. The CSS removes radioactive iodines and particulates from the containment atmosphere during the injection and recirculation phases following a LOCA to ensure that radiological exposures for control room personnel are within 5 rem TEDE, for the duration of the accident~~whole body~~.

Refer to Section 15.5 for radiological consequences of plant accidents. Therefore, the CSS, in conjunction with the CFCS and SAS, limits the control room doses to 5 rem TEDE~~whole body~~.

6.2.2.3.7 General Design Criterion 21, 1967 - Single Failure Definition

The CFCS and CSS are designed such that no single failure in either train will prevent the CHRS from performing its design function. The CSS is comprised of two completely independent trains of containment spray. Each CSS train of pumps and valves are powered by separate Class 1E 4.16-kV and 480-V power supplies, respectively. The CFCS is comprised of five independent fan coolers; two are powered from Class 1E 480-V Bus F, two are powered from Class 1E 480-V Bus G, and one is powered from Class 1E 480-V Bus H.

Any single failure will still leave sufficient CSS and CFCS capability to together mitigate DBAs. Used in conjunction with one another during the injection phase, one containment spray pump and two containment fan cooler units will provide the heat removal capability to maintain the post-accident containment pressure below the design value of 47 psig. CHRS design parameters are listed in Table 6.2-26.

A single failure analysis on all active components of the CHRS was performed to show that the failure of any single component will not prevent performance of the design function. This analysis is summarized in Table 6.2-27.

6.2.2.3.8 General Design Criterion 37, 1967 - Engineered Safety Features Basis for Design

Operation of t~~T~~he CSS ~~is credited only for operation~~ during the injection phase following a LOCA and MSLB is credited for heat removal. As stated in Section 6.2.2, credit is taken for CSS operation during the injection and recirculation phases to support fission product removal following a LOCA. The CFCS is credited for containment pressure and temperature control and mixing of the sprayed and unsprayed regions of the containment to improve airborne fission product removal throughout the injection and recirculation phases following a LOCA or MSLB.

The CSS and CFCS limit the effects of post blowdown energy additions to the containment during the injection phase following a LOCA. For a detailed description of the analytical methods and models used to assess the performance capability of the CHRS, refer to the containment integrity analysis presented in Appendix 6.2D.

The CHRS provides a backup to the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As discussed in this section, the CHRS is designed to withstand any size reactor or secondary coolant pressure boundary break, including a LOCA or MSLB.

6.2.2.3.8.1 Containment Spray System

The CSS, in conjunction with the CFCS, reduces the containment ambient temperature and pressure following a LOCA or MSLB, thus reducing the driving force for leakage of fission products from the containment atmosphere. The CSS removes radioactive - airborne iodine and particulates from the containment atmosphere following a LOCA (refer to Section 6.2.3). The CSS, in conjunction with the spray additive system (SAS), prevents the re-evolution of the iodine in the recirculated spray and core cooling solution (also known as sump water) following a LOCA (refer to Section 6.2.3).

6.2.2.3.8.1.1 Containment Spray Pumps

The containment spray pumps are designed to perform at rated capacity against a total head composed of containment design pressure, nozzle elevation head, and the line and nozzle pressure losses. Adequate NPSH is available for operation of the

6.2.2.3.8.1.3.2 Recirculation Phase

~~If the~~ CSS ~~is used~~ operation in the spray recirculation ~~phase~~mode is manually initiated within 12 minutes of termination of injection spray. ~~, r~~Recirculation spray is provided by the RHR pumps, which draw suction from the containment sump. During the spray recirculation phase, a single RHR pump delivers a minimum flow rate of 1211 gpm to a single spray header. The minimum ECCS core flow available when containment spray is operating in the recirculation mode exceeds the required core flow acceptance criteria taking into consideration the worst case single active failure to minimize the number of pumps available for core cooling. Spray recirculation phase operation of the RHR pumps is discussed in Section 6.3.2.4.3.1 and Section 5.5.6.

6.2.2.3.8.2 Containment Fan Cooler System

6.2.2.3.8.2.1 Containment Fan Coolers

In addition to limiting maximum containment pressure and temperature, the containment fan coolers provide mixing of the containment atmosphere for radioactive iodine and particulate removal, and control of hydrogen buildup as discussed in Sections 6.2.3 and 6.2.5, respectively.

The heat removal capability of the containment fan cooler cooling coils is 81 x 10^6 Btu/hr per fan cooler unit at saturation conditions (271°F, 47 psig), with 2000 gpm cooling water supply at 125°F (refer to Table 6.2-26).

The design internal pressure of each coil is 200 psig and the coils can withstand an external pressure of 47 psig at a temperature of 271°F without damage.

Each cooling coil assembly has a top and bottom horizontal coil casing made of galvanized steel. The safety function of the cooling coil casings is to fill the air gap between each stacked cooling coil assembly and direct airflow through the cooling fins to ensure that adequate heat transfer occurs within the containment fan cooler units.

Each fan can provide a minimum flowrate of 47,000 cfm (in low speed) when operating against the system resistance of approximately 3-3/4 inches of water existing during the accident condition. Minimum flowrate assumptions used to establish containment mixing for fission product removal is discussed in Section 6.2.3 and 15.5.

As discussed in Section 6.2.2.2.1.2.3, each CFCS unit is comprised of four, pre-fabricated modular units. As a result of the modifications made to the fan cooler enclosure assemblies, only Module 3 serves to direct the airflow into the coils for normal and post-LOCA operations. Modules 1 and 2 are no longer required for service.

Module 1 contains the locked-open accident flow inlet dampers. Locking the accident inlet damper open prevents module overpressurization during a LOCA. Module 2 contains the locked-closed accident flow outlet dampers. Modules 1 and 2 are open to

6.2.3 CONTAINMENT AIR PURIFICATION AND CLEANUP SYSTEMS

The containment air purification and cleanup systems are made up of the spray additive system (SAS), the containment spray system (CSS), and the containment fan cooler system (CFCS). The functional performance objectives of the containment air purification and cleanup systems are:

(1) The SAS provides a chemical additive to the CSS to prevent the re-evolution of the iodine in the recirculated spray and core cooling solution (also known as sump water) following a loss-of-coolant-accident (LOCA);

(2) The CSS removes airborne radioactive iodine and particulates from the containment atmosphere following a LOCA;

(3) The CFCS provides mixing of the sprayed and unsprayed regions of the containment to improve airborne iodine removal. Mixing the containment atmosphere maximizes the gas volume treated by the containment spray;

(4) The CFCS and CSS reduce the containment ambient temperature and pressure following a LOCA or a main steam line break (MSLB). While performing this cooling function, the containment heat removal system (CHRS) also helps limit offsite radiation levels by reducing the pressure differential between containment and outside atmospheres, thus reducing the driving force for leakage of fission products from the containment atmosphere (refer to Section 6.2.2);

(5) The CFCS and CSS provide~~s~~ a mixed containment atmosphere for hydrogen control (refer to Section 6.2.5);

(6) The CFCS limits the containment ambient temperature during normal plant operating conditions (refer to Section 9.4.5).

The SAS boundary consists of all piping and valves between, and including, the isolation valve from the refueling water storage tank (RWST), the spray additive tank and the spray eductors. The safety function of iodine removal occurs during operation of the containment spray system and while core coolant is retained in the containment sump.

The physical SSC design bases, testing and inspection requirements of the CSS and CFCS are discussed in Section 6.2.2, therefore the following section addresses the SAS SSCs only.

The two small charcoal filter units in the containment air purification system are not classified as ESFs and are described in Section 9.4.5. These units are not necessary for cleanup during accident conditions.

6.2.3.1 Design Bases

6.2.3.1.1 General Design Criterion 2, 1967 – Performance Standards

The SAS is designed to withstand the effects of, or is protected against, natural phenomena, such as earthquakes, winds and tornadoes, floods and tsunamis, and other local site effects.

6.2.3.1.2 General Design Criterion 3, 1971 – Fire Protection

The SAS is designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions.

6.2.3.1.3 General Design Criterion 11, 1967 – Control Room

The SAS is designed to support actions to maintain and control the safe operational status of the plant from the control room.

6.2.3.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain SAS variables within prescribed operating ranges.

6.2.3.1.5 General Design Criterion 19, ~~1971~~ 1999 – Control Room

The SAS, in conjunction with the iodine removal function of the CSS and CFCS, is designed to support radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation in excess of 5 rem ~~whole body, or its equivalent to any part of the body~~ TEDE, for the duration of the accident.

6.2.3.1.6 General Design Criterion 21, 1967 – Single Failure Definition

The SAS is designed to tolerate a single failure during the period of recovery following an accident without loss of its protective function, including multiple failures resulting from a single event, which is treated as a single failure.

6.2.3.1.7 General Design Criterion 37, 1967 – Engineered Safety Features Basis for Design

The SAS is designed for in-situ periodic testing and surveillance.

6.2.3.1.15 General Design Criterion 65, 1967 – Testing of Operational Sequence of Air Cleanup Systems

The SAS is designed with the capability to test, under conditions as close to design as practical, the full operational sequence that would bring the SAS into action.

6.2.3.1.16 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The SAS, in conjunction with the iodine removal function of the CSS and CFCS, is designed with provisions for maintaining control over the plant's radioactive gaseous effluents.

6.2.3.1.17 10 CFR 50.55a(f) – Inservice Testing Requirements

American Society of Mechanical Engineers (ASME) code components within the SAS are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

6.2.3.1.18 10 CFR 50.55a(g) – Inservice Inspection Requirements

ASME code components within the SAS are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

6.2.3.1.19 Regulatory Guide 1.183, July 2000, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

The steady state elemental iodine removal coefficients, and the time dependent particulate aerosol removal coefficients, are developed in accordance with Regulatory Guide 1.183, July 2000, Appendix A, Section 3.3.

6.2.3.1.~~19~~ 20 Generic Letter 89-10, June 1989 – Safety-Related Motor-Operated Valve Testing and Surveillance

The SAS PG&E Design Class I motor-operated valves (MOVs) meet the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996.

6.2.3.2 System Description

6.2.3.2.1 General Description

The SAS is designed to add sodium hydroxide to the containment spray water to

The SAS is designed to the fire protection guidelines of Branch Technical Position APCSB 9.5-1 (refer to Appendix 9.5B Table B-1).

6.2.3.3.3 General Design Criterion 11, 1967 – Control Room

Each of the system's MOVs can be operated individually by switches in the control room. Valve position indication on the same control room panel is used to verify valve operability.

The spray additive tank level instruments provide two alarms to announce when the solution in the tank has dropped below a level approaching the Technical Specification minimum requirements.

6.2.3.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Analog and logic channels employed for initiation of SAS operation are discussed in Section 7.3. All alarms will be annunciated in the control room.

The SAS will be actuated by a "P" signal initiated either manually from the control room or on coincidence of two sets of two-out-of-four high-high containment pressure signals. Coincidence of "S" and "P" signals will start the containment spray pumps and open the discharge valves to the spray headers. The "P" signal alone will open the valves associated with the spray additive tank.

A locally mounted indicator on the nitrogen line monitors the spray additive tank pressure while adding nitrogen and during periodic inspections.

A flow element is located in the discharge line from the spray additive tank. Flow indication is provided in the control room.

For the spray additive tank, two separate instruments are provided: one to supply readout in the control room and the other to provide local indication.

6.2.3.3.5 General Design Criterion 19, ~~1971~~1999 – Control Room

The CSS provides iodine removal from the containment atmosphere. The CFCS provides mixing of the containment atmosphere to maximize iodine removal. The SAS increases pH in the containment recirculation sump water. The SAS, CSS and CFCS work in conjunction to ensure that radiological exposures for control room personnel are within 5 rem TEDE ~~whole~~ for the duration of the accident. ~~body,~~ R~~r~~efer to Section 15.5 for radiological consequences of plant accidents.

6.2.3.3.6 General Design Criterion 21, 1967 – Single Failure Definition

A failure analysis was conducted on the active components, MOVs 8994A/B, of the SAS to show that failure of any single active component will not prevent the design function from being fulfilled. This analysis is summarized in Table 6.2-38.

MOV 8992 in the spray additive line, which is required only in the short term, is not included in the single failure analysis of Table 6.2-38 because it is not an active component. It performs no active function, is normally open, receives a "P" signal to ensure positive opening, and is designed to fail as is, i.e., in the open position.

In addition, during power operation, power is removed from the single inline SAS outlet MOV 8992 at the circuit breaker at the motor control center with the valve in the open position to prevent a single active failure.

The single failure analysis for the CSS is given in Section 6.2.2.

6.2.3.3.7 General Design Criterion 37, 1967 – Engineered Safety Features Basis for Design

The design basis of the ~~system~~ SAS is two-fold:

(1) Sufficient sodium hydroxide must be added to the containment spray water to ensure rapid absorption by the spray of elemental iodine present in the containment atmosphere following a LOCA.

(2) During the injection phase operation of the spray pumps, a sufficient amount of sodium hydroxide must be carried to the containment sump water via the containment spray to ensure retention of the iodine in the sump solution.

~~Performance of the CSS as an iodine removal mechanism is conservatively evaluated at the containment design temperature and pressure. Since this peak pressure condition is expected to exist for a few minutes at most, and mass transfer parameters and spray flowrate improve with decreasing pressure, an appreciable margin is added to the evaluation. The design case removal constant for the CSS (λ_s) provided in Table 6.2-36 was calculated by applying the model derived in Section 6.2.3.3.7.3 at this back pressure condition to the sprayed portion of the containment volume.~~
The ~~spray system~~CSS, by virtue of the large contact surface area provided between the droplets and the containment atmosphere, affords an excellent means of absorbing radioactive iodine released as a consequence of a LOCA. Sodium hydroxide is added to the spray fluid to increase the absorption of iodine in the spray to the point where the rate of absorption is largely limited by the transfer rate through the gas film surrounding the drops. Reference 5 describes in detail the analytical and experimental basis for the above containment atmosphere iodine removal mechanism. ~~The approach used is summarized below.~~

However, the SAS does not have a significant effect on aerosol particle removal as this removal process is largely mechanical.

The SAS is dependent upon the CSS for operation. The CSS can function with one spray train operating (abnormal operating mode) or with both spray trains operating (normal operating mode). In addition, the operation of one or both ECCS trains affects the rate of withdrawal of water from the RWST, the duration of the spray injection phase, and thus the amount of sodium hydroxide added to the containment.

The SAS~~Spray iodine removal performance~~ has been evaluated for the design case, a double-ended LOCA, assuming that:

(1) Only one-out-of-two spray pumps operate (one spray train operating)

(2) The ECCS operates at its maximum capacity (two ECCS trains operating)

(3) Borated water is retained in the RWST for the exclusive use of the spray during the first part of the ECCS recirculation phase

The second assumption maximizes ECCS flow from the RWST. Overall, the first two assumptions give the most conservative prediction of sodium hydroxide introduction into the containment (minimum containment recirculation sump pH). The third assumption ensures that sufficient sodium hydroxide solution is added to provide for a sump pH of 8.0 or greater when spray injection terminates. Figure 6.2-15 presents various resulting sump pH versus time curves.

The variation of sump pH with time after the accident is shown for various cases in Figure 6.2-15 (Reference 51). At the time spray injection terminates, the sump water has reached an equilibrium pH of at least 8.0. ~~The sump pH will remain at the same pH after the spray injection phase because no additional water is added to the sump.~~

Any reevolution of dissolved iodine from the sump to the containment atmosphere depends upon the concentration gradient between the liquid and vapor phases. The equilibrium between these iodine concentrations is given by the partition coefficient, H, and is a function of concentration, pH, and temperature. The partition coefficient at pH 8.0 exceeds the value of approximately 4×10^3 required to maintain a decontamination factor of 100 in the containment atmosphere for sump temperature above 120° F, and thus a containment atmosphere decontamination factor of 100 or greater can be expected. Figure 6.2-16 presents equilibrium elemental iodine partition coefficients in the containment at various temperatures for the minimum sump pH case. The equations given by Eggleton (Reference 8) were used to determine the partition coefficients. Although the iodate reaction is expected to contribute significantly to the iodine partition at high sump pH values, it has been neglected in these calculations in the interest of conservatism.

SRP Section 6.1.1, Revision 2, Section II, SRP Acceptance Criteria 2A (Reference 72) and SRP Section 6.5.2, Revision 4, Acceptance Criteria 1G (Reference 61) require that the pH of the sump water be controlled to maintain a minimum value of 7.0 following a LOCA. This is required to prevent re-evolution of the iodine that has been removed from the containment atmosphere by the containment spray and washed into the sump water. A neutral pH also limits material degradation, in particular, stress corrosion cracking of austenitic stainless steel components in the post LOCA environment.

Long-term retention of iodine in the sump fluid is strongly dependent on the pH. In determining whether the plant chemical addition system is adequate for long-term pH control, long-term production of acids (hydrochloric acid (HCl) and nitric acid (HNO_3)) by irradiation was addressed. A conservative analysis was performed to confirm that the sump water pH at thirty (30) days following a LOCA remains greater than 7.5. The analysis assumed the minimum volume / concentration values for NaOH, in combination with the maximum volume / boration values for the water sources contributing to the sump water volume.

SRP Section 6.5.2, Revision 4, Acceptance Criteria II.1.g, states that long term iodine retention may be assumed when the equilibrium sump water pH, after mixing and dilution with the primary coolant, is above 7.0.

NUREG/CR-5732, April 1992 (Reference 73), states that iodine re-evolution is not a factor if the ultimate sump water pH of ≥ 7 is achieved prior to the time when iodine re-evolution could potentially occur. Section 3.1 of NUREG /CR-5732, April 1992 notes that the following phenomena occur during the first time interval between t = 0 to t = 1000 min, a) events "leading" to the formation of I_2 by radiolysis, and b) all HI effects except for those related to pH. Re-evolution (i.e. vapor phase elemental iodine produced by radiolysis and partitioned between the aqueous and gas) can occur in the second time interval which is from t=1000 min to t~2 to 3 weeks. Thus for Light Water Reactors, a pH of 7 must be achieved within 1000 min of the initial post-LOCA release.

At DCPP, RWST drain-down and chemical addition is essentially complete within an hour post-LOCA; thus it is expected that as a result of recirculation, the sump water will be well mixed by t=1000 minutes or ~ 16 hours.

Based on the above assessment it was concluded that the DCPP sump water pH will remain greater than 7.0, and the post-LOCA dose consequence analyses need not consider iodine re-evolution from the sump fluid.

6.2.3.3.7.1 Mixing between Sprayed and Unsprayed Regions of Containment

As stated in Section 6.2.3.1.7, the CFCS is designed to ensure mixing of the sprayed and unsprayed regions of the containment atmosphere, to improve airborne fission product removal following any size reactor coolant pressure boundary break.

The containment sprayed volume for the spray injection mode is calculated based on the assumption that the unoccupied containment volume above the operating floor is 100 percent covered by sprays. It includes the sprayed volume below the grating in the operating floor deck, and the refueling cavity volume. Based on the above, aApproximately 17.5 percent of the containment free air volume is not reached by the spray. The values listed below were used to estimate the total unsprayed volume in the containment.

(1)	Containment radius	70 ft
	Height between operating deck and spring line	91 ft
	Approximate deck area covered only by grating	2,990 ft^2
	Height between elevation 91 feet and deck	49 ft
	Average fall height below deck through grating	12 ft
(2)	These data result in the following volumes:	
	Volume in dome	717,000 ft^3
	Volume in cylinder above deck	1,400,000 ft^3
	Occupied volume above deck	-95,000 ft^3
	Sprayed volume below deck	36,000 ft^3
	Sprayed refueling cavity volume	45,000 ft^3
	Total sprayed volume	2,103,000 ft^3
	Total free volume	2,550,000 ft^3
	Total unsprayed volume	447,000 ft^3
	Percent unsprayed volume	≃17.5%

To justify the assumption that the unoccupied containment volume above the operating floor is 100 percent covered by sprays, the analysis of the containment spray coverage during the spray injection mode with only one CSS train operating estimated the projected unsprayed area percentage of the containment deck area of 42 percent for Unit 1 and 44 percent for Unit 2, using actual spray flow patterns. The spray reduction factor of 0.5 used to address the spray compression effect due to elevated containment pressure was conservatively based on the containment design pressure. It was concluded that the whole volume above the operating floor can be considered well mixed and totally subjected to the sprays given the estimated spray coverage calculated above based on actual spray patterns, and the high levels of turbulence due to spray action, entrainment of the air from the volume outside the spray patterns, and induced upflow of air to balance the downflow experienced in the sprayed volume.

The effect of the reduction in spray flow rate between the spray injection phase (i.e., 2456 gpm) and the spray recirculation phase (i.e., 1211 gpm) on the containment spray coverage was evaluated. The minimum volumetric flow rate of water through the spray nozzles, the associated nozzle water pressure drop, and Figure 1 of NUREG/CR-5966 , June 1993 (Reference 63), were used to establish the effect of the reduction in spray flow rate on the spray pattern. The analysis determined that with the continued use of a

spray reduction factor of 0.5 for spray compression, there was a 3 percent increase in the unsprayed area percentage of the containment deck for both units. It was concluded that the continued use of a spray pattern reduction factor of 0.5 is extremely conservative when applied to the recirculation mode (the containment pressure is substantially lower than the design pressure), and that there is no significant increase in the unsprayed area percentage of the containment deck for both Unit 1 and Unit 2. Thus, a sprayed volume of 82.5 percent of the containment free volume is also applicable for the containment spray recirculation mode.

The CFCUs support post-LOCA mixing of the sprayed and unsprayed volume of the containment at a rate higher than that justified by natural convection. The containment mixing rate between the sprayed and unsprayed regions following a LOCA is determined to be 9.13 turnovers of the unsprayed regions per hour. This mixing rate is based on the operation of two CFCUs, with an assumed total volumetric flow rate of 68,000 cfm between the unsprayed regions and sprayed regions.

Review of the locations of the intake and exhaust registers of the operating fan coolers, the major openings in the containment structure, and various active and passive mixing mechanisms, results in the conclusion that following a LOCA, credit can be taken for a) the entire flowrate provided by each operating CFCU to support mixing between the sprayed and unsprayed regions, and b) homogeneous mixing within the sprayed and unsprayed regions, of the volume of air transferred from one region to the other due to CFCU operation.
~~The calculations of thyroid exposures in Section 15.5.17.2.4 were based on the assumption of uniform mixing in the full free volume of the containment. As a result of the circulation of air from the unsprayed portions of the containment free volume to the sprayed areas by the CFCS, good mixing is provided. As shown in Table 6.2-26, the fan cooler unit capacity is 47,000 cfm. If a simplified two-volume model were used, in combination with assumptions that some iodine was available for leakage from the lower containment section (unsprayed), some reduction in the effective calculated spray removal coefficient would result.~~

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~~*In order to evaluate these and other possible combinations of degraded performance of the spray system, a sensitivity study was performed to determine the effect of a reduced removal coefficient on thyroid exposures. The results of this study are presented in Figures 15.5-6, 15.5-7, and 15.5-8. As shown in these figures, both the short-term and the long-term thyroid exposures are very insensitive to reduced spray removal coefficient down to values as low as 10 hr^{-1}.*~~

6.2.3.3.7.2 Fission Product Removal

Regulatory Guide 1.183, July 2000, Appendix A, Section 3.3 (Reference 62), invokes the guidance of SRP 6.5.2, Revision 4 ~~(Reference 61)~~ and NUREG/ CR-5966, June 1993 (Reference 63) as acceptable models for removal of iodines and particulates.

In accordance with the above guidance, the fission product removal coefficients developed for the LOCA are based on the following:

1) Elemental iodine removal coefficients are calculated using guidance provided in SRP 6.5.2 which is invoked by Regulatory Guide 1.183, July 2000, Appendix A, Section 3.3.

2) Time dependent particulate aerosol removal coefficients are estimated using guidance provided in Regulatory Guide 1.183, July 2000, Appendix A, Section 3.3, for alternative source terms, and use of computer program SWNAUA. Refer to Section 15.5.8.12 for a description of computer program SWNAUA.

The total elemental iodine and particulate removal coefficients in the sprayed and unsprayed region of the containment as a function of time are summarized in Table 6.2-32. The methodology utilized to develop these values is summarized below.

6.2.3.3.7.2.1 Elemental Iodine Removal

The methodology presented in Section III, 4.C.i, of SRP 6.5.2, Revision 4 is used to estimate the elemental iodine removal coefficients. The removal of elemental iodine from the containment atmosphere can be attributed due to wall deposition ($\lambda_{E, Wall}$) and due to the action of containment spray ($\lambda_{E, Spray}$).

<u>Elemental Iodine Removal Coefficients Due to Wall Deposition ($\lambda_{E,Wall}$)</u>

The elemental iodine removal coefficients due to wall deposition can be estimated using the equation provided in SRP 6.5.2, Revision 4.

$$\lambda_{E, Wall} = K_w \cdot A / V$$

Where: K_w = mass transfer coefficient (ft/hr)
A = wetted surface area (ft^2)
V = volume of the containment (ft^3)

Note: K_w value of 4.9 m/hr (or 16.08 ft/hr.) conservatively envelopes all available experimental data (SRP 6.5.2, Revision 4 ~~Reference 61~~).

The total containment surface area (435,256 ft^2) is initially available for wall deposition due to condensation on heat sink surfaces prior to spray actuation after accident. Subsequently, due to heat-up, certain portions of the heat sink surfaces become non-condensing and can no longer be considered as "wetted" surfaces. However, after spray actuation, since the heat sink surfaces in the sprayed region are continuously wetted by sprays, elemental iodine removal due to wall deposition in the sprayed region is valid over the entire period of containment spray operation. The wetted surface area within the sprayed volume is conservatively assumed to be limited to the carbon steel lined containment shell surface area (90,560 ft^2) multiplied by the spray coverage

fraction of containment volume. Therefore, wall deposition elemental iodine removal coefficients are calculated during the period (1) prior to containment spray actuation, applicable to the entire containment, and (2) post containment spray actuation, applicable in the sprayed region for the duration of spray operation.

Elemental Iodine Removal Coefficients Due to Sprays in the Sprayed Region ($\lambda_{E,Spray}$)

The elemental iodine removal coefficients due to spray actuation can be estimated using the equation provided in Section III, 4.C.i, of SRP 6.5.2.

$$\lambda_{E,\ Spray} = (6\ K_g \cdot t \cdot F) / (V_s \cdot d)$$

Note: This equation is valid for 10 $hr^{-1} \leq \lambda_{E,\ Spray} \leq 20\ hr^{-1}$ to prevent extrapolation beyond the existing data for boric acid solution with a pH of 5.

Where: K_g = gas phase mass transfer coefficient (ft/hr)
t = time of fall of the spray droplets (= h/U_T) (hr)
F = volumetric spray flow rate (ft^3/hr)
V_s = effectively sprayed containment free volume (ft^3)
d = mass-mean diameter of the spray droplets (ft)
h = mean fall height of the spray droplet (ft)
U_T = terminal velocity of the spray droplet (ft/hr)

The gas phase mass transfer coefficient is determined by using the equation provided in Reference 64.

$$K_g = (D_g/d) \times 2.0 \times (1 + 0.276 \cdot Re^{1/2}\ Sc^{1/3})$$

Where: D_g = diffusivity of iodine in the gas film surrounding the drop (ft^2/hr)
Re = dimensionless Reynolds number
Sc = dimensionless Schmidt number

6.2.3.3.7.2.2 Particulate Removal

There are several aerosol mechanics phenomena that promote the depletion of aerosols from the containment atmosphere. These include the natural phenomena of particle growth due to agglomeration, gravitational settling of particles (also called gravitational sedimentation), ~~diffusional plate-out,~~ diffusiophoresis (particulate removal due to steam condensation), and removal by fluid mechanical interaction with the falling droplets that enter the containment atmosphere through the spray system nozzles (i.e., containment spray).

All of the above phenomena are credited for DCPP. Agglomeration of the aerosol is considered in both sprayed and unsprayed regions. In the sprayed region, the particulate removal calculation takes credit for the removal effectiveness of

diffusiophoresis and sprays. Gravitational settling is considered only in the unsprayed region.

Removal of Particulates by Sprays

The particulate removal rate is calculated using computer code SWNAUA. The model correlations implemented into SWNAUA conservatively underestimates the spray removal coefficient. The spray model incorporated in the SWNAUA code was originally described in Reference 66. When performing DBA calculations to determine particulate removal in the effectively sprayed region of the containment, only the conservatively developed spray removal models and conservative steam condensation rates for the diffusiophoresis calculation are utilized. While agglomeration is considered in the calculation, its impact on the resulting particulate removal rates is negligible. In summary, the aerosol removal rates calculated by SWNAUA are conservative lower bound estimates.

The spray model in SWNAUA evaluates the particulate removal efficiency for each particle size in the aerosol by the following mechanisms: inertial impaction, interception, and Brownian diffusion. The aerosol removal constant due to spray is presented in NUREG-0772, June 1981 (Reference 67) as:

$$\lambda_{spray} = \frac{3 F_m h \varepsilon}{4 R_{sp} \rho_w V} \times \frac{v_{spray} - v_{sed}}{v_{spray}}$$

Where

λ_{spray}	=	Particulate removal constant for spray,
F_m	=	Spray mass flow rate,
h	=	Spray fall height,
ε	=	Collision efficiency,
R_{sp}	=	Spray droplet radius,
ρ_w	=	Density of the spray droplet,
V	=	Effectively sprayed volume of containment,
v_{spray}	=	Velocity of the spray droplets, and
v_{sed}	=	Aerosol sedimentation velocity.

The collision efficiency is divided into three contributing mechanisms as described in BMI-2104 (Reference 68):

$$\varepsilon = \varepsilon_i + \varepsilon_r + \varepsilon_d$$

Where

ε_i	=	Efficiency due to inertial impaction,
ε_r	=	Efficiency due interception and
ε_d	=	Efficiency due to Brownian diffusion.

For viscous flow around the spray droplet, the inertial impaction efficiency is given in Reference 67:

$$\varepsilon_i = \frac{1}{\left[1 + \frac{0.75 \ln (2\,Stk)}{Stk - 1.214}\right]^2}$$

The critical Stokes number, Stk, for viscous flow is 1.214; for Stk below this value, the model assumes the efficiency of inertial impaction is 0.0. The Stk is calculated from Reference 68:

$$Stk = \frac{2 \rho_p r^2 C_c (v_{spray} - v_{sed})}{9 \mu R_{sp}}$$

Where

r	=	Aerosol particle radius,
ρ_p	=	Aerosol density,
C_c	=	Cunningham slip correction factor,
μ	=	Gas viscosity.

For droplet sizes typical of nuclear plant spray systems, the data of Walton and Woolcock (Reference 70) show that collision efficiency will be closer to that predicted for potential flow around the droplet. Calvert (Reference 71) fitted this data to the expression:

$$\varepsilon_i = \left(\frac{Stk}{Stk + 0.7}\right)^2$$

The collision efficiency predicted by this equation is always higher than that predicted by the viscous flow expression given above. The Calvert's fit is employed in this calculation.

As for the remaining constituents of the collision efficiency, the spray model employs an interception efficiency of the form:

$$\varepsilon_r \cong \frac{3}{2}\left(\frac{r}{R_{sp}}\right)^2 x \left(1 - \frac{1}{3}\frac{r}{R_{sp}}\right)$$

which is a conservative approximation of the expression given by Reference 68. The efficiency due to Brownian motion is also taken from this report:

$$\varepsilon_d = 3.5Pe^{-2/3}$$

Where

Pe	=	Peclet number
	=	$2v_{spray}R_{sp}/D_B$,
D_B	=	Aerosol diffusion coefficient
	=	k_{Boltz}TB (Fuchs: Reference 69),
k_{Boltz}	=	Boltzmann constant
	=	1.3804 X 10^{-16} erg/K.
T	=	Temperature, K

Reference 69 gives the aerosol mobility, B:

$$B = \frac{C_c}{6 \pi \mu r}$$

In most cases, the overall collision efficiency is dominated by inertial impaction, but for small aerosols, Brownian diffusion may become dominant. The collision efficiency due to inertial impaction increases as the aerosol size is increased, whereas that due to Brownian diffusion increases as the aerosol size decreases.

Removal of Particulates by Diffusiophoresis

Particulate matter is entrained in the steam as it flows to the condensation surfaces. This phenomenon is called diffusiophoresis. Steam is assumed to condense on spray droplets, on the containment fan cooler units, and on heat sinks.

The key input parameters used for the particulate removal coefficients are presented in Tables 6.2-33, 6.2-34, 6.2-35, and 6.2-37.

~~**6.2.3.3.7.1 Drop Size Distribution**~~

~~The drop size distribution used in the analytical model is based on data obtained from measurements of the actual size distribution from the Spraco 1713A nozzle for the range of pressure drops encountered during operation of the spray system. A complete analysis of the expected drop size distributions, including a statistical analysis is contained in References 5, 6, and 33. The parameters used in applying these distributions to the calculation of the iodine removal coefficient for the DCPP units are given in Tables 6.2-29, 6.2-36, and 6.2-37.~~

~~**6.2.3.3.7.2 Condensation**~~

~~As the spray solution enters the high-temperature containment atmosphere, steam will condense on the spray drops. The amount of condensation is calculated by an enthalpy balance on the drop:~~

$$mh + m_c h_g = m'h_f \quad (6.2\text{-}10)$$

where:

m and m' = mass of the drop before and after condensation
m_c = mass of condensate, lb
h = initial enthalpy of the drop, Btu/lb
h_g and h_f = saturation enthalpy of water vapor and liquid, respectively, Btu/lb

The increase in each drop diameter in the distribution is, therefore, given by:

$$\left(\frac{d'}{d}\right)^3 = \left(\frac{v}{v_f}\right)\left(\frac{h_g - h}{h_{fg}}\right) \quad (6.2\text{-}11)$$

where:

v_f = specific volume of liquid at saturation, ft^3/lb
v = specific volume of the drop before condensation, ft^3/lb
h_{fg} = latent heat of evaporation, Btu/lb
h_g = enthalpy of steam at saturation, Btu/lb
d = drop diameter before condensation, cm
d' = drop diameter after condensation, cm

The increase in drop size due to condensation is expected to be complete in a few feet of fall for the majority of drop sizes in the distribution. More detailed calculations by Parsly show that even for the largest drops in the distribution, thermal equilibrium is reached in less than half the available drop fall height.

6.2.3.3.7.3 Mass Transfer Model

The basic equation for the iodine concentration in the containment atmosphere is derived from a material balance of the elemental iodine in the containment. The iodine removal by the spray system may be expressed by:

$$V_c \frac{dC_g}{dt} = -EF(HC_g - C_{L1}) \quad (6.2\text{-}12)$$

where:

V_c = containment free volume, cc
C_g = iodine concentration in the containment atmosphere, gm/cc
H = iodine partition coefficient, (gm/liter of liquid)/(gm/liter of gas)
F = spray flowrate, cc/sec

The resulting change in the drop size distribution is taken into consideration in the mass transfer calculations described below.

The variable E is the absorption efficiency, which may also be described as the fractional approach to saturation:

$$E = \frac{C_{L2} - C_{L1}}{C_{L^*} - C_{L1}} \qquad (6.2\text{-}13)$$

where:

C_{L1} = iodine concentration in the liquid entering the dispersed phase, gm/cc

C_{L2} = iodine concentration in the liquid leaving the dispersed phase, gm/cc

C_{L^*} = equilibrium iodine concentration in the liquid, gm/cc

This absorption efficiency is calculated from the time-dependent mass transfer model suggested by L. F. Parsly (Reference 7).

The absorption efficiency calculated is a function of drop size, and the removal constant λ_s, in reciprocal hours, for the entire spray is, therefore, obtained by an appropriate summation over all drop size groups:

$$\lambda_s = \sum_{i=1}^{n} \frac{E_i F_i H}{V_c} \qquad (6.2\text{-}14)$$

A further discussion of drop size distribution, drop trajectories, drop coalescence and mass transfer modeling is presented in References 5, 6, and 33.

6.2.3.3.7.4 Experimental Verification of Models

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The ability of the model described to give conservative estimates of actual spray performance was demonstrated in test runs made at Oak Ridge National Laboratory (ORNL) and Battelle Pacific Northwest Laboratory. The results of these tests (Reference 5), shown in Figure 6.2-14 for Run A6, verified that the spray removal model used is conservative in all cases.

6.2.3.3.8 General Design Criterion 38, 1967 – Reliability and Testability of Engineered Safety Features

The SAS testing and inspections are enveloped by the inservice inspection (ISI) and inservice testing (IST) programs. Pressure containing portions of the CSS and SAS are inspected in accordance with ASME BPVC, Section XI, as required by the Technical

6.2.3.3.19 Regulatory Guide 1.183, July 2000, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

The steady state elemental iodine removal coefficients, and the time dependent particulate aerosol removal coefficients, are developed in accordance with Regulatory Guide 1.183, July 2000, Appendix A, Section 3.3, and discussed in detail in Section 6.2.3.3.7.2.

6.2.3.3.~~19~~ 20 Generic Letter 89-10, June 1989 – Safety-Related Motor-Operated Valve Testing and Surveillance

The SAS MOVs are subject to the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996, and meet the requirements of the DCPP MOV Program Plan.

6.2.3.4 Tests and Inspections

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The CSS was tested functionally in accordance with written procedures, as outlined in Chapter 14.

Spray pump delivered flow and head data were recorded to verify that the containment spray pumps meet design criteria.

Spray additive eductor performance data were provided by the manufacturer based on actual tests of a similar eductor. These tests were conducted using a 1.3 specific gravity solution to verify eductor design performance. Additional manufacturer's tests were run using water so that comparative performance data were available for the two different additive solutions at eductor design conditions. Eductor performance was checked subsequent to installation into the system. Spray additive flowrates were measured, with resulting rates in the range 31.5 to 38.5 gpm (35 gpm ± 10 percent) considered acceptable.

Each containment spray header was tested individually by connecting a source of air to the normally capped flange connection on the spray pump discharge header, shutting the manual spray header isolation valve and opening the air test line isolation valve and the motor-operated spray header isolation valve. Individual nozzles were checked for proper performance by streamers, which indicated unobstructed air flow.

The containment sump recirculation mode was tested initially as part of a preoperational flow test under ambient conditions of the safety injection system (SIS). The purpose of the test was to demonstrate the capability of appropriate subsystems to deliver fluid from the containment sump into the reactor coolant system (RCS) in the required time.

Position (5) - The CIS is designed so that the containment setpoint pressure that initiates containment isolation for nonessential penetrations is set to the minimum compatible with normal operating conditions with additional margin to allow for a small pressure transient.

Position (6) - The vacuum/overpressure relief valves~~DCPP purge system valves~~ satisfy Branch Technical Position CSB 6-4, September 1975.

The containment purge system valves are sealed closed in accordance with SRP 6.2.4, Item II.6 and II.14.

Position (7) - The containment vent and purge isolation valves close on a high radiation signal.

6.2.4.1.16 Generic Letter 89-10, June 1989 - Safety-Related Motor-Operated Valve Testing and Surveillance

In the CIS, PG&E Design Class I position-changeable motor-operated valves (MOVs) meet the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996.

6.2.4.1.17 Generic Letter 96-06, September 1996 - Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

The CIS is designed to prevent thermally induced overpressurization of isolated water-filled piping sections in containment during design-basis accidents.

6.2.4.2 System Description

The CIS includes the mechanical and instrumentation fluid penetrations and associated valves and isolation devices. These penetrations are identified in Figure 6.2-19 and Table 6.2-39. The CIS design uses the following premises:

(1) An automatic containment isolation barrier is provided by a closed system, a trip valve, or a check valve.

(2) A closed system meets the following requirements:

a) Inside the containment:

1. No mass transfer with either the RCS or the reactor containment interior

2. Has the same safety classification as ESFs (PG&E Design Class I, Quality/Code Class II)

Position (6) – The ~~DCPP containment purge system valves and~~ vacuum/overpressure relief valves satisfy the operability criteria set forth in BTP CSB 6-4, 1975. The opening of the 12 inch vacuum/overpressure relief valves is restricted to no more than 50 degrees.

The containment purge system valves are sealed closed in accordance with SRP Section 6.2.4, Revision 3, Item II.6 and II.14 during Mode 1 through 4 in accordance with the Technical Specifications, therefore satisfying NUREG-0737, November 1980 Item II.E.4.2, Position (6).

Position (7) - The ~~containment purge and~~ vent isolation valves are closed automatically by any one of the following:

(1) Phase A containment isolation signal

(2) High gaseous or air particulate radioactivity in containment

(3) High radiation at the plant vent

The containment purge valves are isolated during Modes 1 through 4 in accordance with the Technical Specifications, and are closed automatically by High gaseous or air particulate radioactivity in containment. The phase A containment isolation signal is no longer credited for automatic isolation of the containment purge valves.

6.2.4.4.16 Generic Letter 89-10, June 1989 – Safety-Related Motor-Operated Valve Testing and Surveillance

The CIS MOVs are subject to the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996, and meet the requirements of the DCPP MOV program.

6.2.4.4.17 Generic Letter 96-06, September 1996 – Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

Containment isolation valves LWS-FCV-253 and LWS-FCV-500 (Penetrations 49 and 50, respectively) have both been modified through the addition of a pressure relief hole to prevent overpressurization of the associated isolated section of piping during design-basis accidents to ensure containment integrity is maintained. All other piping penetrations that are susceptible to overpressurization either have valves whose design prevents overpressurization (air-operated diaphragm valves, solenoid valves, or air-operated globe valves) or have been drained to prevent overpressurization and thereafter maintained drained as appropriate.

6.2.4.5 Tests and Inspections

collections that may provide a flammability risk and the need to provide equipment for monitoring hydrogen concentrations in the containment atmosphere.

To fulfill the requirements of 10 CFR 50.44, PG&E credits the use of the ~~containment fan cooler system (~~CFCS and CSS~~)~~ as the means of containment atmosphere mixing. Refer to Section 6.2.2 for discussion of ~~other~~ the CFCS and CSS design bases.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

The research and development work discussed in more detail in Section 6.2.5.3 substantially reduced the uncertainties in both the expected rates of hydrogen accumulation and the potential exposures that would result from hydrogen control by venting as follows:

(1) Research on the corrosion of aluminum and associated hydrogen production rates by Westinghouse (References 12 - 15 and 42) has reduced uncertainties on corrosion rates in the expected post-accident environment and allowed a reduction in the expected corrosion rate from the 42 $mg/dm^2/hr$ used in the Unit 2 PSAR to the value shown in Figure 6.2-24.

(2) The amounts of aluminum used in the as-built plant have been minimized through materials design specifications. Zinc is another significant contributor. The uncertainties in the amounts of hydrogen produced from both have been reduced by itemized accounting (refer to Table 6.2-42).

(3) The amounts of hydrogen expected to be produced by the zirconium-water reaction have been reduced by the more stringent limits established on ECCS performance.

(4) Research by the Atomic Energy Commission (AEC) and its contractors (a partial compilation is included in Reference 16) in the context of emergency core cooling system (ECCS) studies has substantially reduced uncertainties in the extent of zirconium-water reactions following a LOCA.

(5) Reevaluation of energy generation rates has allowed reduction of hydrogen generation rate from sump radiolysis.

(6) Research on hydrogen yield in the core and sumps by Westinghouse (References 13-15) has reduced uncertainties in these constants.

(7) Refined analysis of the distribution of fission product decay energy (Reference 17) has resulted in more precise values for the fractions of beta and gamma energies absorbed by water.

CFCS and CSS ~~is~~are designed to ensure containment atmosphere mixing as a result of any size reactor coolant pressure boundary break to prevent the coalescence of local hydrogen concentrations within containment.

6.2.5.1.7 General Design Criterion 41, 1967 – Engineered Safety Features Performance Capability

The CFCS and CSS are~~is~~ designed to provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill containment atmosphere mixing.

6.2.5.1.8 General Design Criterion 42, 1967 – Engineered Safety Features Components Capability

The CFCS and CSS are~~is~~ designed so that the capability of each component and system to perform its required function is not impaired by the effects of a LOCA.

6.2.5.1.9 General Design Criterion 49, 1967 – Containment Design Basis

The containment combustible gas control systems are designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

6.2.5.1.10 General Design Criterion 54, 1971 – Piping Systems Penetrating Containment

The piping that is part of the CHPS and hydrogen monitoring system that penetrate containment is provided with leak detection, isolation, redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating this system. The piping is designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

6.2.5.1.11 General Design Criterion 56, 1971 – Primary Containment Isolation Valves

The CHPS and hydrogen monitoring system contain valves in piping that penetrate containment and connect directly to the containment atmosphere. Remote manual isolation valves are provided outside containment and automatic (check) valves are provided inside containment to ensure containment integrity is maintained.

6.2.5.1.12 10 CFR 50.44 – Combustible Gas Control for Nuclear Power Reactors

The CFCS and CSS ensure~~s~~ a mixed atmosphere is maintained within containment to prevent high localized concentrations of hydrogen gas accumulation.

The hydrogen monitoring system is designed to be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere.

6.2.5.1.13 10 CFR 50.49 – Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

CHPS, EHRS and the hydrogen monitoring system components that require environmental qualification (EQ) are qualified to the requirements of 10 CFR 50.49.

6.2.5.1.14 10 CFR 50.55a(f) – Inservice Testing Requirements

American Society of Mechanical Engineers (ASME) code components of the CHPS and hydrogen monitoring system are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

6.2.5.1.15 10 CFR 50.55a(g) – Inservice Inspection Requirements

ASME code components of the CHPS and hydrogen monitoring system are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

6.2.5.1.16 Regulatory Guide 1.7, Revision 2, November 1978 – Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

The CFCS and CSS are~~is~~ designed to provide a mixed atmosphere in containment and thus control combustible gas concentrations without relying on purging of the containment atmosphere following a LOCA. The CFCS and CSS meet~~s~~ the design, quality assurance, redundancy, energy source, and instrumentation requirements for an engineered safety feature. The hydrogen monitoring system provides a means to measure the hydrogen concentration in the containment.

6.2.5.1.17 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The hydrogen monitoring system is designed to provide continuous indication in the control room of hydrogen concentration in the containment atmosphere following a beyond design basis accident and meets the design provisions of Regulatory Guide 1.97, Revision 3, May 1983 including qualification, redundancy and testability.

6.2.5.1.18 NUREG-0737 (Items II.E.4.1, II.F.1), November 1980 – Clarification of TMI Action Plan Requirements

100 eV or less. Regulatory Guide 1.7, Revision 2, November, 1978 does not, however, allow credit for the reduced hydrogen yields and a yield value of 0.5 molecules per 100 eV is used in the analyses.

All containment volumes are connected by large vent areas to promote good air circulation. Hydrogen will diffuse very rapidly giving an even distribution under the conditions existing in the containment structure. In addition, thermal mixing effects, heating of air above the hot sump water, and possible steam released from the RCS will move the hydrogen-laden air from the points of generation toward the cool external walls. Although hydrogen is lighter than air, it will not concentrate significantly in high areas because of the high diffusion rate, the open design of the containment, and the fan cooler air mixing.

The ability of hydrogen to diffuse rapidly into all volumes is inferred from a CSE experiment (Reference 23). These tests showed very good mixing in the main chamber and a rapid interchange by diffusion and mixing with the atmosphere of other chambers that had limited communication. The diffusivity of hydrogen is approximately 10 times that of iodine, so a more uniform mixture is expected for hydrogen. Also, higher concentration provides greater concentration gradients for better diffusion than indicated by the CSE tests.

Table 6.2-45 summarizes the calculated hydrogen production and accumulation data.

6.2.5.3.6.1.5 Results of the Hydrogen Generation and Accumulation Analyses

The results of the hydrogen generation and accumulation analyses are presented in Figures 6.2-26 through 6.2-29.

6.2.5.3.7 General Design Criterion 41, 1967 – Engineered Safety Features Performance Capability

The CFCS, including required auxiliary systems, is designed to tolerate a single active failure following a LOCA without loss of protective function. Refer to Section 6.2.2.3.2 for further discussion.

6.2.5.3.8 General Design Criterion 42, 1967 – Engineered Safety Features Components Capability

The component design pressure and temperature conditions in Table 6.2-26 are specified as the most severe conditions to which each CFCS and CSS component is exposed during either normal or post-LOCA operation allowing the system to provide a mixed atmosphere in post-LOCA conditions. Refer to Section 6.2.2.3.1 for further discussion.

6.2.5.3.9 General Design Criterion 49, 1967 – Containment Design Basis

6.2.5.3.14 10 CFR 50.55a(f) – Inservice Testing Requirements

Periodic inservice testing (IST) of all containment isolation valves, in the system is performed. The IST requirements are contained within the IST Program Plan and comply with the ASME code for Operation and Maintenance of Nuclear Power Plants. Refer to Section 6.2.2.4 for a discussion of testing of the CFCS.

6.2.5.3.15 10 CFR 50.55a(g) – Inservice Inspection Requirements

The inservice inspection (ISI) requirements for the CHPS and containment hydrogen monitoring penetrations are contained within the ISI Program Plan and comply with the ASME BPVC, Section XI. Refer to Section 6.2.2.4 for a discussion of inspection of the CFCS and CSS.

6.2.5.3.16 Regulatory Guide 1.7, Revision 2, November 1978 – Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

The CFCS and CSS, which serve~~serving~~ as the credited means for containment atmosphere mixing in accordance with 10 CFR 50.44, ~~is~~ are designed and constructed to PG&E Design Class I standards.

The hydrogen monitoring system includes two hydrogen monitors to measure the hydrogen concentration in the containment.

The hydrogen monitoring system was originally designed and constructed as Category 1, as defined by Regulatory Guide 1.97, Revision 2, December 1980 however has been reclassified as Category 3, as defined by Regulatory Guide 1.97, Revision 3, May 1983 as a result of the rulemaking revision to 10 CFR 50.44 (refer to Section 6.2.5.3.17).

The EHRS is available to control containment hydrogen concentration following a LOCA to at or below 4.0 percent by volume without relying on the CHPS. Each of the two redundant recombiners is capable of providing the required removal capacity. The CHPS is also available.

To ensure that the lower flammability limit (4 percent) will not be exceeded, the internal electric hydrogen recombiners will be started at or below 3.5 percent by volume.

The EHRS and CHPS systems meet PG&E Design Class I design and construction standards.

The EHRS provides 100 percent redundancy since each recombiner and its associated power supply and control panel are capable of providing the required hydrogen removal capacity. The second unit, including its associated power supply and control panels, is normally on standby following a postulated LOCA.

Refer to Sections 6.2.5.3.3, 6.2.5.3.16, and 6.2.5.3.17 for additional discussion on the hydrogen monitors.

6.2.5.5.2 Containment Hydrogen Purge System

Instrumentation is provided to monitor the flowrate and the amount of radioactivity released by the purging operation.

The containment radiation monitoring system and the plant vent radiation monitors are used to monitor the radioactivity in containment and the hydrogen purge line. Refer to Sections 9.4.2 and 11.4 for information regarding the plant vent.

A manual sample point is provided on each exhaust line to obtain a grab sample for laboratory analysis.

Flow indicators are provided for each CHPS exhaust line. The indicators are PG&E Design Class I. The range is 500 to 4000 feet per minute (corresponding to flowrates of approximately 45 to 350 cfm).

Containment isolation valves status is shown on the main control board as indicated in Table 6.2-39. Annunciation is provided to alarm on high radioactivity, high flowrate, and fan failure.

Refer to Sections 6.2.5.3.3 and 6.2.5.3.4 for additional discussion on the CHPS instrumentation.

6.2.5.6 Materials

Materials of construction of components are indicated in Section 6.2.5.2.

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6.2.7 REFERENCE DRAWINGS

Figures representing controlled engineering drawings are incorporated by reference and are identified in Table 1.6-1. The contents of the drawings are controlled by DCPP procedures.

CONTAINMENT PIPING PENETRATIONS AND VALVING

Pentr. Nos.	System (Safety Priority) (Note 33)	Appli-cable GDC	GDC Confor-mance	Figure 6.2-19 Sheet No.	Vlv Ltr	Valve ID Number (Note 51)	Valve Type (Note 30)	Operator Type (Note 31)	Cntmt Locat. (Note 32)	PG&E Piping Group	Control Room Indication	Normal Position (Note 35)	Power Fail. Position	Trip On (Note 25)	Used After LOCA (Note 36)	Long Term Post-LOCA Position (Note 37)	Fluid (Note 23)	Temp (Note 24)	Notes
59H	Containment Pressure Transmitters PT-933 & PT-935 (SA)	56	Yes	15 (I)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
					A	Sealed Instrument	Sin	-	I	-	-	-	-	-	-	-	W	Cold	16
					B	Sealed Bellows	Sbl	-	O	-	-	-	-	-	-	-	W	Cold	16
60	Mini-Equipment Hatch (NE)	-	-	-	-	Not a piping penetration	-	-	-	-	-	-	-	-	-	-	-	-	-
61	Containment Purge Supply (NE)	56	Yes	18 (II)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
					D	VAC-FCV-660	But	Air	I	AE	Yes	C	Closed	T	N	C	G	Cold	1856
					E	VAC-FCV-661	But	Air	O	AE	Yes	C	Closed	T	N	C	G	Cold	1856
62	Containment Purge Exhaust (NE)	56	Yes	18 (III)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
					F	VAC-RCV-11	But	Air	I	AE	Yes	C	Closed	T	N	C	G	Cold	1856
					G	VAC-RCV-12	But	Air	O	AE	Yes	C	Closed	T	N	C	G	Cold	1856
63	Containment Pressure and Vacuum Relief (NE)	56	Yes	18 (I)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
					A	VAC-FCV-662	But	Air	I	A	Yes	C	Closed	T	N	C	G	Cold	18
					B	VAC-FCV-663	But	Air	O	A	Yes	C	Closed	T	N	C	G	Cold	18
					C	VAC-FCV-664	But	Air	O	A	Yes	C	Closed	T	N	C	G	Cold	18
					J	Spectacle Flange	Spf	-	O	A	-	N/A	-	-	N/A	N/A	G	Cold	26
64	Fuel Transfer Tube (NE)	-	-	3 (III)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	26
						Not a piping penetration	-	-	-	-	-	-	-	-	-	-	-	-	-
65	Personnel Hatch (NE)	-	-		-	Not a piping penetration	-	-	-	-	-	-	-	-	-	-	-	-	-
66	Emergenc	-	-		-	Not a piping penetration	-	-	-	-	-	-	-	-	-	-	-	-	-