ATTACHMENT 1

Diablo Canyon Power Plant
Technical Assessment
Prepared by WECTEC Global Project Services Inc. (WECTEC)

Implementation of Alternative Source Terms
Summary of Dose Analyses and Results
Revision 4

WECTEC

TECHNICAL REPORT
WECTEC Global Project Services, Inc.

IMPLMENTATION OF ALTERNATIVE SOURCE TERMS
Summary of Dose Analyses and Results

Revision 4

DIABLO CANYON POWER PLANT

Prepared for:
Pacific Gas & Electric Company

October 5, 2016
QA CATEGORY I
Safety Related

RECORD OF REVISIONS

Rev. No.	Description of Changes	Pages Revised	Pages Added	Pages Replaced
0	Original Issue	N/A	N/A	N/A
1	• Includes post-LOCA dose consequences at the TSC • Includes post-LOCA dose to the CR operator during routine ingress/egress to the CR for the 30-day duration of the accident • Removes text on assessment of impact of CS operation in the recirculation mode on other aspects of plant design. This information is deemed unnecessary since DCPP is currently licensed to operate with CS in the recirculation mode. • Updates Section 2.1 (Proposed Changes to Current Licensing Basis), to identify the Condition II and III accidents in UFSAR Section 15.5 that are not being reanalyzed but are being demonstrated as being compliant with 10CFR50.67 using comparison to the DBAs analyzed in this report. • Included AST vs CLB parameter Tables; i.e., Appendix B • For completeness, included Table 4.3-1 and updated Table 7.3-2 to provides "isotopic gap inventories" • Deleted Section 2.4 (Compliance with RG 1.183), and added a discussion on LOOP assumptions in Section 7.0 • Updated text / dose estimates to address the effect of increased exhaust flow from the EDRT / AB pipe tunnel areas, on the post-LOCA dose contribution due to MEDT leakage • Incorporated credit for the AB filter when estimating the dose consequences at the TSC & CR for the RHR pump seal failure release pathway. This was necessary to accommodate the dose increase due to the increased exhaust flow from the EDRT / AB pipe tunnel areas • Updated Dose summary Table to report doses only up to 2 significant figures. • Incorporates several editorial changes and text enhancements to provide additional clarification.	Full Revision	Full Revision	Full Revision
2	a. Updates the fuel gap activity fractions from the highest fractions per isotope / isotope class provided in NUREG/CR 5009 / Safety Guide 25 / RG 1.183 R0 to that provided in Table 3 of Draft Guide (DG) 1199 for the LRA and the FHA. <u>Background</u>: Subsequent to receipt of NRC SRXB RAI-2 against DCPP AST Licensing Application Request (LAR) 15-03, and during the related PG&E / NRC conference call dated 10/13/2015, it was agreed that in lieu of responding to the NRC request for "fission gas release calculations using NRC approved methods and bounding power history" to demonstrate that use of the highest fuel gap fractions per isotope / isotope class provided in NUREG/CR 5009 / Safety Guide 25 / RG 1.183 R0 was conservative, PG&E could instead update the Fuel Handling accident (FHA) and Locked Rotor Accident (LRA) dose consequence analyses using the gap	As noted by Revision bars	N/A	N/A

Rev. No.	Description of Changes	Pages Revised	Pages Added	Pages Replaced
	fractions from Table 3 of DG-1199. This latter option was acceptable to NRC if PG&E could demonstrate that DCPP falls within, and intends to operate within, the maximum allowable power operating envelop for PWRs shown in Figure 1 of DG-1199. Upon receipt of confirmation from DCPP Reactor Engineering that DCPP operation does fall within Figure 1 of DG-1199, PG&E elected to update the referenced analyses using gap fractions from DG-1199, and re-submit the dose consequences associated with the FHA and LRA in support of AST Implementation b. The conservative response time assumed for the CR intake monitors to detect the cloud of activity resulting from the FHA is increased from 10 to 20 seconds. c. Table 5.2-4 - For consistency in description throughout all atmospheric dispersion factor tables, the following changes were made in the "Source and Receptor" Column i. "140' leakage" was replaced by "Containment Penetration (GE Area)" ii. "Penetration Leakage" was replaced by "Containment Penetration (GW/FW Area)" d. A few minor word or phrase changes made to improve clarity or support consistency in text.			
3	a) Updated Section 2.2 to remove the commitment to re-classify as PG&E Design Class I the portion of the 2-inch gas decay tank vent line that connects to the Plant Vent. b) Updated Section 5.2 to include a discussion of the cumulative dose impact in the control room as a result of a worst case break in the PG&E Design Class II a) 2-inch gas decay tank vent line and b) 16-inch GSC/SJAE exhaust header (including connected PG&E Design Class II lines.)	As noted by Revision bars	N/A	N/A
4	a. Updates the atmospheric dispersion factor (χ/Q) values presented in the following Tables (as well as the meteorological data files names in the Tables in Appendix A) Tables in Chapter 5 Table 5.1-1 – EAB/LPZ χ/Q values Table 5.2-2 - Unit 1 CR χ/Q values Table 5.2-3 - Unit 2 CR χ/Q values Table 5.2-4- TSC χ/Q values Tables in Chapter 7 Table 7.2-5 – CR χ/Q values (LOCA) Table 7.2-6 - TSC χ/Q values (LOCA) Table7.3-3 - CR χ/Q values (FHA) Table 7.4-2 - CR χ/Q values (LRA) Table 7.5-2 - CR χ/Q values (CREA)	As noted by Revision bars	N/A	N/A

Rev. No.	Description of Changes	Pages Revised	Pages Added	Pages Replaced
	Table 7.6-2 - CR χ/Q values (MSLB) Table 7.7-3 - CR χ/Q values (SGTR) Table 7.8-2 - CR χ/Q values (LOL) <u>Tables in Appendix B</u> Table B.1-2 – AST vs CLB_EAB/LPZ χ/Q values Table B.2-2A - Duplicate of Table 7.2-5 (LOCA) Table B.3-2A – Duplicate of Table 7.3-3 (FHA) Table B.6-2A – Duplicate of Table 7.6-2 (MSLB) Table B.7-2A – Duplicate of Table 7.7-3 (SGTR) <u>Background</u>: On 2/17/2016, the NRC issued ARCB RAIs with respect to the meteorological data used in support of DCPP AST Licensing Application Request (LAR) 15-03. Specifically, the NRC identified an inconsistency with RG 1.23 methodology in determination of *stability classes* in the DCPP meteorological data provided by PG&E in support of the AST application. Upon investigation PG&E confirmed that the *hourly stability classes* in the met data used to develop the AST dose consequence analyses had been developed by site meteorologists using methodology *different* from that recommended by RG 1.23. It is PG&E's intent to be consistent with RG 1.23 methodology in the development of hourly stability classes for purposes of developing post-accident off-site and on-site consequences. Therefore, the χ/Q values supporting this application have been revised accordingly and the updated values presented in the tables listed above. PG&E has also concluded that the use of the updated χ/Qs <u>does not adversely impact the dose consequences</u> reported in LAR 15-03 for the EAB / LPZ / CR or TSC Based on the above • The <u>updated χ/Qs</u> reported herein will become the χ/Qs of record upon AST implementation and will be the <u>DCPP licensing basis</u>. • The <u>dose consequences reported in LAR 15-03</u> are considered conservative and bounding and will <u>remain unchanged.</u> b. In response to a verbal request from NRC, PG&E has also added notes in the Chapter 7 χ/Q tables listed above to further explain the basis of selection of the accident specific χ/Q values used to establish the dose consequences at the CR and TSC (LOCA only) to ensure bounding dose estimates for an accident at either unit.			
5	a. Section 2.1 – Remove TS 5.5.9 from list of Impacted Tech Specs (Per Supplement to LAR 15-03, 9/15/2016)	As noted by	N/A	N/A

Rev. No.	Description of Changes	Pages Revised	Pages Added	Pages Replaced
	b. Updated Chapter 5 as noted below i. Added description of the EAB assumed in dose analyses for the area over the ocean ii. Included the methodology used to calculate the building wake cross-sectional area for both on-site and site boundary calculations iii. Clarified the terrain adjustment factors used for annual average calculations iv. Included Tables 5.1-2 (0.5% sector dependent χ/Q values and distances for the EAB and LPZ) and 5.1-3 (5% overall χ/Qs for EAB and LPZ) v. Updated the exit velocity to wind speed ratio calculations in Section 5.2 under section titled "Energetic Releases" vi. Updated the following χ/Q values to correct for minor errors in determining the azimuth direction a) In Table 5.2-2: U1 Plant Vent to CR Center and b) In Table 5.2-3: U2 Plant Vent to CR Center and U2 RWST to U1 CR Emergency Intake. vii. Added TSC χ/Qs for Non-LOCA release points in Table 5.2-4 c. Updated the affected χ/Q values in Table 7.2-5 to reflect the minor error in determining the azimuth direction discussed above. d. Added Section 7.9 to address the dose consequences in the TSC due to Non-LOCA events e. Added Table 7.9-1 to summarize the χ/Q values used in estimating the dose consequences in the TSC due to Non-LOCA events f. In Appendix A - Added Unit 1 and Unit 2 release point and receptor configuration information for the χ/Q values developed for the TSC due to Non-LOCA releases g. A few minor word or phrase changes made to improve clarity or support consistency in text. Background: On 9/7/2106, the NRC issued additional RAIs against DCPP AST LAR 15-03 which primarily focused on request for clarification on a) some of the inputs used in the χ/Q calculations, b) the exact location of the site boundary and c) confirmation that the Loss-of-Coolant Accident (LOCA) is the bounding accident with respect to dose consequences in the TSC. In addition, • per Supplement to LAR 15-03, 9/15/2016, TS 5.5.9 was removed from the list of impacted Technical Specifications. • affected χ/Q values were updated to reflect a minor error in determining the azimuth direction (4 cases). Note: In accordance with the updated analysis of record, the reported post-LOCA dose consequences in the control room is not affected.	Revision bars		

TABLE OF CONTENTS

1.0 INTRODUCTION

As holder of operating licenses issued prior to January 10, 1997, and in accordance with 10CFR50.67 (Reference 1) and Standard Review Plan 15.0.1 (Reference 2), Pacific Gas and Electric (PG&E) proposes to revise the accident source term used in the Diablo Canyon Power Plant (DCPP) Units 1 and 2 design basis site boundary and control room dose analyses, with the full implementation of Alternative Source Terms (AST) as defined in Regulatory Guide (RG) 1.183, Section 1.2.1 (Reference 3).

The first use of the AST for DCPP was a selective application per RG 1.183, Section 1.2.2, to revise the Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB) in order to implement changes to the PG&E Design Class I ventilation filter testing program for the Control Room Ventilation System (CRVS), Auxiliary Building Ventilation System (ABVS) and the Fuel Handling Building Ventilation System (FHBVS). The FHA analysis demonstrated that acceptable doses would occur at both offsite locations and in the control room without taking credit for filtration by ventilation systems or control room isolation. The application was reviewed and approved by the Nuclear Regulatory Commission (NRC) in its Safety Evaluation Report (SER) for License Amendment Nos. 163 and 165 to DCPP Facility Operating License Nos. DPR-80 and DPR-82, respectively. (Reference 4)

With this application, and in the interest of evaluating DCPP design against a more realistic accident sequence, as well as in gaining dose analysis margin, the methodology / scenarios used in the following design basis accident (DBA) analyses discussed in the DCPP UFSAR (some of which utilize pre-NUREG-0800 assumptions), are being updated to reflect the AST guidance provided in RG 1.183.

1. Loss of Coolant Accident (LOCA)
2. FHA in the Containment
3. Locked Rotor Accident (LRA)
4. Control Rod Ejection Accident (CREA)
5. Main Steam Line Break (MSLB)
6. Steam Generator Tube Rupture (SGTR)
7. Loss-of Load (LOL) Event

The Loss of Load Event is not addressed in Regulatory Guide 1.183 but has been selected by PG&E for re-evaluation to provide a bounding analysis for Condition II events that have no fuel damage but may have radioactivity in the environmental releases that occur as a result of reactor trip. The control room ventilation system is expected to remain in normal operation mode for the duration of these events.

In addition, the dose consequence analysis associated with the FHA in the FHB has been revised to address updated design input as discussed below.

The dose consequence analyses addressed in this application resolve the findings of a Licensing Basis Verification Project (LBVP) which was initiated by the licensee and has resulted

in a total upgrade of the listed radiological post-accident dose consequence analyses. The effect of the findings on current plant design, have been addressed in Prompt Operability Assessments which resulted in the implementation of several temporary compensatory measures.

In addition, the listed dose consequence analyses have been revised to address:

- Updated control room ventilation system parameters resulting from the installation of new back-draft dampers in the control room emergency ventilation filter recirculation lines. These dampers were installed to prevent reverse unfiltered flow into the control room.
- Updated control room unfiltered inleakage (including back-draft damper leakage)

As part of an effort to improve the completeness and quality of design documentation and reflect current NRC guidance, PG&E has also elected to update the offsite atmospheric dispersion factors (χ/Q) using recent meteorological data and RG 1.145, Revision 1 (Reference 5) methodology. The χ/Q values applicable to on-site locations such as the Control Room and Technical Support Center, have been calculated using the "**A**tmospheric **R**elative **CON**centrations in Building Wakes" (ARCON96) methodology (Ramsdell, 1997, Reference 6). The updated off-site and on-site atmospheric factors are utilized in the dose consequence analyses for the accident listed above.

Also, and as part of the effort to clarify and streamline the DCPP licensing basis related to post-accident dose consequences documented in DCPP UFSAR Section 15.5, PG&E is requesting NRC approval of the following proposed changes in licensing basis:

1. Removal of all the "expected" accident dose consequence assessments that were included in DCPP UFSAR Section 15.5.1 as part of the original license application.

   The original DCPP licensing basis included two evaluations, or cases, for each accident. The first case, called the expected case, used values for each factor involved in the accident, which were intended to be estimates of the actual values expected to occur if the accident took place. The resulting doses were close to the doses expected from an accident of this type. The second case, the DBA, used the customary conservative assumptions. The calculated doses for the DBA, while not a realistic estimate of expected doses, provided the basis for determination of the design adequacy of the plant safety systems.

   Current NRC guidance related to expectations for a Safety Analysis Report for a nuclear power plant (e.g., NUREG-0800), does not require the inclusion of dose consequences from "expected" accident scenarios. Since these "expected accident scenario" evaluations are not relevant for determination of the design adequacy of the plant safety systems, PG&E is proposing to remove this information from its licensing basis.

2. Elimination of the dose contribution of a containment purge via the Containment Hydrogen Purge System (CHPS) following a LOCA for purposes of hydrogen control.

The NRC revised 10CFR50.44 (Reference 9) to acknowledge that the amount of combustible gas generated for the design basis LOCA was not a risk significant threat to containment integrity. Thus, with the exception of demonstrating the capability of ensuring a mixed atmosphere within containment, the requirements for hydrogen control pertaining to the design basis LOCAs was eliminated. In the SER for License Amendment Nos. 168 and 169 to DCPP Facility Operating License Nos. DPR-80 and DPR-82, respectively (Reference 10), the NRC confirmed the elimination of hydrogen release concerns associated with a design-basis LOCA, and the associated requirements that necessitated the need for the hydrogen recombiners and backup hydrogen vent and purge systems.

To ensure consistency with the current licensing basis, PG&E is proposing to eliminate the dose contribution due to the containment purge pathway currently included in the LOCA dose consequence analysis in support of hydrogen control.

2.0 REGULATORY APPROACH

In 1962, the U.S. Atomic Energy Commission (AEC) published Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactors" (Reference 7), which specified release fractions of the different categories of fission products from the core to the reactor containment, or "source term", to be considered in the event of a postulated design basis accident (DBA) involving a "substantial meltdown of the core" in a Light Water Reactor. These fractions were based on small-scale experiments performed in the late 1950s in which irradiated UO2 pellets were heated to accident temperatures.

The TID-14844 accident source term formed the basis for NRC Regulatory Guide 1.4 (Reference 8), which was used to determine compliance with the NRC reactor siting criteria, 10CFR100.11 (Reference 11). It was also used to evaluate other plant performance requirements including control room habitability requirements given in 10CFR50 Appendix A, GDC-19 (Reference 12) and expanded in NUREG-0800, Standard Review Plan 6.4 (Reference 13), to 5 rem whole body dose, 30 rem beta dose and 30 rem thyroid dose. The TID-14844 source term was also used for evaluation of equipment qualification (EQ) of class IE electrical components in accordance with 10CFR50.49 (Reference 14), and evaluation of vital area accessibility post-LOCA in accordance with NUREG-0737 (Reference 15).

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 16), which used this research to provide estimates of an accident source term that was more physically based and that could be applied to Light Water Reactors. This effort crystallized into regulatory acceptance for an alternative source term and the issuance of 10CFR50.67 and RG 1.183, which formalized the applicability of AST for operating reactors.

As noted in Section III.2.a, Standard Review Plan 15.0.1, July 2000, (Reference 2), a full implementation of AST addresses a) all the characteristics of AST (i.e., the radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these nuclides), b) replaces the previous accident source term used in all design basis radiological analyses, and c) incorporates the Total Effective Dose Equivalent (TEDE) criteria of 10CFR50.67, and Section II of Standard Review Plan 15.0.1.

With this application, PG&E proposes a "full" implementation of AST as defined in RG 1.183, Section 1.2.1 for DCCP Units 1 and 2. To that end, the dose consequences of the eight accidents discussed in Section 1.0 have been re-analyzed using the guidance provided in RG 1.183. The dose consequences for the other events that have an accident source term, and are a part of the current DCPP licensing basis, are addressed by qualitative comparisons to the accidents that have been analyzed. (Refer to Section 2.1, Item 22).

However, for the reasons outlined below, the post-LOCA integrated doses utilized for radiological environmental qualification of PG&E Design Class I equipment, and the estimated operator mission doses while performing vital functions post-LOCA, will continue to be based on TID-14844 assumptions and will remain unchanged.

- Equipment Qualification (EQ): This above approach is acceptable based on Section 1.3.5 of RG 1.183 which indicates that though EQ analyses impacted by plant modifications should be updated to address the impact of the modification, no plant modification is required to address the impact of the difference in source term characteristics (i.e. AST vs TID-14844) on EQ doses until the generic issue associated with a potential increase in cesium releases is resolved. NUREG-0933, "Resolution of Generic Safety Issues", Section 3.0, Item 187 resolved the generic issue related to the effect of increased cesium releases on EQ doses. The NRC staff 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. (Reference 55)

- Post-LOCA Vital Area Mission Doses: This above approach is acceptable based on the AST benchmarking study reported in SECY-98-154 (Reference 17) which concluded that results of analyses based on TID-14844 would be more limiting earlier on in the event, after which time the AST results would be more limiting. The NRC SER for Fort Calhoun Station's implementation of AST (Reference 18) referenced the SECY-98-154 study as the source for the conclusion that results of analyses based on TID-14844 would be more limiting for periods up to one to four months after which time the AST results would be more limiting. Post-LOCA access to vital areas usually occur within the first one or two weeks when the original TID-14844 source term is more limiting.

2.1 Proposed Changes to Current Licensing Basis

PG&E proposes to revise the DCPP licensing basis to implement the AST described in RG 1.183 through a) reanalysis of the radiological consequences of the UFSAR Chapter 15.5 accidents listed in Chapter 1.0 in accordance with the guidance provided in RG 1.183, and b) implementation of the following changes in plant operations / licensing basis.

1. The TEDE acceptance criterion of 10CFR50.67(b)(2) will replace the previous whole body and thyroid dose acceptance criteria of 10CFR100.11.

2. Use of new offsite atmospheric dispersion factors (χ/Q) based on recent 5-year meteorological data (2007 to 2011) and RG 1.145 methodology.

3. Use of new on-site atmospheric dispersion factors (χ/Q) for locations such as the Control Room and the Technical Support Center based on recent 5-year meteorological data (2007 to 2011) and ARCON96 methodology.

4. Use of inhalation dose conversion factors from Federal Guidance Reports (FGR) No.11 (Reference 19)

5. Proposed changes to the DCPP Plant Technical Specifications / Technical Specifications Bases include

   - TS 1.1 - Dose Equivalent I-131 concentrations will be developed based on the committed thyroid dose equivalent conversion factors provided in EPA FGR No. 11

- TS 3.4.16 – The Noble gas activity shall be limited to ≤ 270 µci/gm DE Xe-133. The current limit of 600 µCi/gm DE Xe-133 corresponds to ~1% fuel defects which is the DCPP design basis value for system and shielding design. The limit is being reduced to control the noble gas activity in the coolant to levels below the design basis values.
- TS 3.6.3 – The 48 inch containment purge valves will meet NUREG 0737, Item II.E.4.2, Position 6 (Reference 15), by being sealed closed as defined by SRP 6.2.4, item II.6.f (Reference 22) during MODE's 1 through 4 (currently allows the containment purge system to be operable for less than 200 hours per year during MODE's 1 through 4).
- TS 5.5.11 - The allowable methyl iodine penetration for the Auxiliary Building Ventilation charcoal filter will be changed to 5% (currently the allowable methyl iodine penetration for the Auxiliary Building Ventilation charcoal filter is 15%)
- TS 5.5.19 – The dose acceptance criterion to demonstrate compliance with the control room habitability program will be changed from 5 rem whole body or its equivalent to any part of the body for the duration of the accident to 5 rem TEDE.
- TS Bases 3.4.17 – Primary to Secondary leakage (total for all 4 Steam Generators (SGs)) will be limited to a maximum of 0.75 gpm at STP which includes accident induced leakage (note that TS 3.4.13d limits the maximum allowable operational leakage to 150 gpd, per SG)
- TS Bases 3.6.6 - will be updated to require initiation of containment spray in the recirculation mode within 12 minutes of termination of injection spray inclusive of a description of the updated valve alignments.

The related TS Bases, including those that reference 10CFR100 or define recently irradiated fuel, will also be affected.

6. Use of an increased value for control room (CR) unfiltered air inleakage.

7. Credit for the dual ventilation intake design of the CR pressurization air intakes. Based on the availability of redundant PG&E Design Class I radiation monitors at each pressurization intake location, the DCPP design has the capability of initial selection of the cleaner intake, but does not have the capability of automatic selection of the clean intake throughout the event. Based on the CRVS pressurization intake design, and the expectation that the operator will manually make the proper intake selection throughout the event, and per RG 1.194, June 2003, Regulatory Position C.3.3.2.3, (Reference 21), when the CRVS is in Mode 4, the χ/Q values for the more favorable CR intake is reduced by a factor of 4 and utilized to estimate the dose consequences. See Section 5.2 for further detail.

8. To support flexibility in future DCPP fuel management schemes with respect to the potential of having fuel rods that exceed the RG 1.183, Revision 0 linear heat generation rate criteria, and since DCPP falls within, and intends to operate within, the maximum allowable power operating envelop for PWRs shown in Figure 1 of Draft Guide (DG)-1199 (Reference 71), the fuel gap activity fractions used for the DCPP Non-LOCA

events that experience fuel damage (with the exception of the CREA) are based on Table 3 of DG-1199.

9. MSLB / SGTR / LRA / CREA / LOL: Credit for a reduction factor of 5 applied to the calculated ARCON96 χ/Q values for energetic releases from the DCPP Main Steam Safety Valves (MSSVs) and the 10% Atmospheric Dump Valves (ADVs). This approach has been deemed acceptable per Regulatory Position C.6 of RG 1.194 for uncapped and vertically-oriented relief valves. See Section 5.2 for further detail.

10. MSLB / SGTR / LRA / CREA / LOL: Credit for the fact that as a result of the close proximity of the MSSVs/10% ADVs and the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant post-accident plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit. See Section 5.2 for further detail.

11. FHA: Credit for the PG&E Design Class I radiation monitors located at the control room (CR) normal intakes to switch the CRVS from Mode 1 (normal operation) to Mode 4 (pressurized filtered mode) following a FHA in the Fuel Handling Building or Containment. See Section 7.3 for further detail.

12. FHA: Credit for the following administrative controls reflected in plant procedures that ensure the FHB is maintained at a negative pressure relative to atmosphere during movement of irradiated fuel thus ensuring that the environmental releases occur via the Unit vent: See Section 7.3 for further detail.

    - The movable wall is in place and secured
    - No exit door from the FHB is propped open
    - At least one FHBVS exhaust fan is operating. The supply fan flow has been confirmed by design to have less flow than the exhaust fan; thus environmental releases are via the Unit Vent

13. FHA: A minimum decay time of 72 hrs prior to fuel movement. See Section 7.3 for further detail.

14. LOCA: Removal of all the "expected" accident dose consequence assessments that were included in DCPP UFSAR Section 15.5.1 as part of the original license application.

15. LOCA: Credit for containment spray in the recirculation mode - To address the delayed core damage sequence of a post-LOCA AST scenario and support fission product removal from the containment atmosphere, credit is taken for manual initiation of containment spray in the recirculation mode within 12 minutes of termination of containment spray in the injection mode. Containment spray in the recirculation mode is credited until 6.25 hours after accident initiation. See Section 7.2.1 for more detail.

16. LOCA: Updated allowable ESF System leakage values and associated release points. See Section 7.2.3.3 for more detail.

17. LOCA: Updated allowable ESF System leakage into the Miscellaneous Equipment Drain Tank (MEDT) and associated releases / release points. See Section 7.2.3.6 for more detail.

18. LOCA: Inclusion of environmental releases from the Refueling Water Storage Tank (RWST) vent due to sump water back-leakage. See Section 7.2.3.5 for more detail.

19. LOCA: Elimination of the dose contribution due to the containment purge pathway currently included in the dose consequence analysis in support of hydrogen control.

20. LOCA: Inclusion of environmental releases via the 12 inch Containment Vacuum / pressure relief pathway prior to containment isolation. See Section 7.2.3.1 for more detail.

21. LOCA: Defining the portion of Room 506 of the Control Room which serves as a control room foyer adjacent to the Shift Supervisor's office, as a low occupancy area, and conservatively assigning the referenced area with an occupancy factor of less than 5% of the total time spent daily in the control room. See Section 7.2.5.2 for more detail.

22. LOCA/ FHA / MSLB / SGTR / LRA / CREA / LOL: Update of the list of computer codes that support the dose consequence analyses.

23. .Dose Consequences Associated with Other Chapter 15 Accidents:

    The DCPP licensing basis includes dose assessments at offsite locations for several Condition III and Condition IV events which are not addressed in Regulatory Guide 1.183 and have therefore not been re-analyzed with this application. The demonstration of compliance with the accident-specific regulatory limits at the EAB and LPZ for the referenced accidents will be addressed using engineering judgment and by comparison to the accident sequence, predicted fuel damage (if applicable), and resultant dose consequences of the DBAs discussed in Section 1.0 and re-analyzed in support of this application.

    (i) Small Break LOCA (SBLOCA) inside Containment (Condition III event): The possible radiological consequence of a SBLOCA inside containment (defined in UFSAR Chapter 15.3.1 as a break that is large enough to actuate the emergency core cooling system), is expected to be bounded by the "containment release" scenario of the control rod ejection accident (CREA) since the CREA is postulated to result in 10% fuel damage, whereas the SBLOCA has no fuel damage. Refer to Section 7.5 for the CREA.

    (ii) Minor Secondary System Pipe Break (Condition III event): The possible radiological consequence of a minor secondary system line break (defined n UFSAR Section 15.3 as 6 inch diameter or smaller) is expected to be significantly less than a MSLB since in both cases there is no fuel damage and the steam releases following a minor secondary line break is expected to be significantly less than that associated with a MSLB. Refer to Section 7.6 for the MSLB.

(iii) Complete Loss of Forced Reactor Coolant Flow (Condition III event): The possible radiological consequence of a complete loss of forced reactor coolant flow (postulated to result from a simultaneous loss of electrical supplies to all reactor coolant pumps resulting in increased coolant temperature; no predicted fuel damage; dose consequences resulting from the atmospheric steam dumping required for plant cooldown), is expected to be bounded by the conservative Loss-of-Load scenario with a coincident Loss of Offsite power analyzed herein. Refer to Section 7.8 for the LOL event.

(iv) Underfrequency Event (Condition III event): The possible radiological consequence of an underfrequency event (no predicted fuel damage; dose consequences resulting from the atmospheric steam dumping required for plant cooldown), is expected to be bounded by the conservative Loss-of-Load scenario evaluated for DCPP which includes a coincident Loss of Offsite Power. Refer to Section 7.8 for the LOL event.

(v) A Single Rod Cluster Control Assembly Withdrawal (Condition III event): The possible radiological consequence of a single rod cluster control assembly withdrawal will be less than a CREA since the CREA is postulated to result in 10% fuel damage, whereas the condition of one rod cluster control assembly fully withdrawn with the rest of the bank fully inserted, at full power has only 5% fuel damage. Refer to Section 7.5 for the CREA.

(vi) Major Rupture of a Main Feedwater Pipe (Condition IV event): The possible radiological consequence of a main feedwater line break (no predicted fuel damage; dose consequences resulting from airborne activity released to the environment from the break location and via the MSSVs / 10% ADVs) is expected to be bounded by the MSLB since the airborne environmental release via the break point is expected to be less than the MSLB. Per Standard Review Plan 15.2.8, Section III, Item 6 (Reference 32), the evaluation of the radiological consequences of a design basis feedwater line break may be based on a qualitative comparison to the results of the design basis MSLB. Refer to Section 7.6 for the MSLB.

Tank Rupture Events: The tank rupture events represent the accidental release of the radioactivity accumulated in tanks resulting from normal plant operations and are not affected by the accident source term associated with AST. Therefore, the tank rupture events are not reanalyzed in support of this Licensing Application Request.

2.2 Planned Design Modifications

The following design modifications will be implemented prior to implementation of AST:

1. The external west wall of the Control Room has a 6'-8.5" x 10'-1" area where the concrete was replaced by other material as a result of a design change. The affected area is the CR briefing room. PG&E will install shielding material at this location equivalent to that provided by the CR outer walls.

2. The setpoints for the redundant PG&E Design Class I gamma sensitive area radiation monitors 1-RE-25/26, 2-RE-25/26 will be updated. These monitors are designed to automatically isolate the control room normal air intakes and shift to CRVS Mode 4 (filtered / pressurized emergency ventilation) on detection of high radiation at the location of the CR normal intakes. The FHA analyses developed in support of this License Application Request utilize an "analytical limit" of 1 mR/hr for the gamma radiation environment at the CR normal operation air intakes prior to taking credit for CRVS Mode 4 actuation. The actual monitor trip setpoint will include instrument loop uncertainty; i.e., Trip setpoint (or High Alarm Setpoint (HASP) = Analytical limit / (1 + uncertainty %)).

3. The portion of the GE/GW 40-inch Containment Penetration Area Ventilation line that connects to the Plant Vent up to and including the isolating damper solenoid valves, the associated damper actuators and the pressure switches will be re-classified as PG&E Design Class I (Refer to Section 5.2 for detail).

4. Install a high efficiency particulate (HEPA) filter in the TSC normal ventilation system intake.

2.3 Planned Procedural Updates

Provided below are the key plant operating procedures that will be updated prior to implementation of AST. The full set of impacted procedures will be addressed in the AST Engineering Change Package (ECP).

1. Update of Equipment Control Guideline ECG 42.1 to lower the restriction on fuel movement from 100 hrs to 72 hrs post-shutdown.

2. Review / update (as deemed necessary) of the EOPs and operator training procedures to ensure that the requisite steps to select the least contaminated outside air intake, and the provisions for monitoring to ensure the least contaminated intake is in use throughout the event, are provided.

3. Update of Surveillance Test Procedure STP M-57, Control Room Ventilation System Tracer Gas Test Procedure, to include the new CR inleakage test acceptance criteria and the range of CRVS ventilation flows deemed acceptable by the AST dose consequence analyses.

4. Update of the EOPs to include valve alignment information to manually initiate containment spray in the recirculation mode within 12 minutes of termination of injection spray, and to ensure, that at a minimum, spray operation continues until 6.25 hours after accident initiation. An associated Time Critical Operator Action (TCOA) will be implemented upon receipt of NRC approval of this application.

5. Update of the ESF system leak testing procedures (that are part of the Boundary Leakage Program invoked by Technical Specification Section 5.5.2) to establish administrative acceptance criteria to ensure that:

a) The total as-tested leakage from ESF systems that recirculate sump fluid outside containment is ≤ 126 cc/min, and with the following breakdown:

  - In areas covered by the Auxiliary Building ventilation the as-tested leakage is ≤ 120 cc/min
  - In the containment penetration area the as-tested leakage is ≤ 6 cc/min

b) The total as-tested back leakage into the RWST from the containment recirculation sump is ≤ 1 gpm.

c) The total as-tested flow hard piped to the MEDT is ≤ the following values:

  - Leakage from systems carrying non-radioactive fluids ≤ 484 cc/min
  - Leakage from ESF systems that recirculate sump fluids ≤ 950 cc/min

6. Update the TSC administrative procedures to ensure that
   - The nominal normal operation TSC ventilation air intake flowrate is 500 cfm
   - Following a LOCA, the TSC will be manually placed in Mode 4 operation such that filtered pressurization and recirculation can be credited within 2 hours of accident initiation.
   - The nominal post-LOCA TSC ventilation filtered pressurization and recirculation flowrates are 500 cfm, respectively.

7. Review and update, as necessary, administrative procedure OP B-8H (which is a compilation of all the precautions and prerequisites involved with moving irradiated fuel, fuel components and other items over / within the spent fuel pool) to include the requirements and / or restrictions imposed by the FHA dose consequence analyses on the FHB door closure status, the operation of the FHB ventilation system and the ability of the CRVS to actuate Mode 4 operation.

8. Update of DCPP core design procedure TS6.DC3 to include additional verification of core power peaking. Specifically, the limit on the peak rod linear heat generation rate in fuel assemblies during normal operation will be confirmed to remain within the nodal power envelope depicted for PWRs in Figure 1 of DG-1199.

2.4 Dose Acceptance Criteria

PG&E has utilized the following acceptance criteria for the AST DCPP site boundary and control room dose analyses supporting this application:

<u>The acceptance criteria for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Dose</u> are based on 10CFR50.67, and Section 4.4 Table 6 of Regulatory Guide 1.183:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, shall not receive a radiation dose in excess of the accident-specific TEDE value noted in Reference 3, Table 6.

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a radiation dose in excess of the accident-specific TEDE value noted in Reference 3, Table 6.

EAB and LPZ Dose Acceptance Criteria - Condition II and Condition III events:

RG 1.183 (regulatory guidance for accident analyses using AST), does not specifically address Condition II and Condition III scenarios. However, per RG 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10CFR50.67 in all dose consequence analyses. In addition, Section 4.4 of RG 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of RG 1.183, the postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. Thus, the dose consequences at the EAB and LPZ for Condition II and Condition III events will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10CFR50.67.

<u>The acceptance criterion for the Control Room Dose</u> is based on 10 CFR 50.67:

Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

This criteria ensures that the dose criteria of GDC 19, 1999 and NUREG-0737, November 1980, Item III.D.3.4 is met.

<u>The acceptance criterion for the Technical Support Center Dose</u> is based on Section 8.2.1, Item f of NUREG-0737, Supplement 1 (Reference 27), as amended by RG 1.183, Section 1.2.1, which states that a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10CFR50.67 in all dose consequence analyses:

The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

3.0 COMPUTER CODES

The computer codes utilized in support of this application are listed below. These computer programs have been verified and validated under the CB&I S&W Inc. NRC approved Quality Assurance Program, and have been shown to be accurate and acceptable for the use discussed below.

Code Name	Description of Code Use
SCALE 4.3 / SAS2/ ORIGEN-S	SCALE 4.3, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations And Personal Computers," Control Module SAS2 – CB&I S&W Inc. QA Category I Computer Code NU-230, V04, L03. (Reference 60) ORIGEN-S is part of the SCALE 4.3 suite of codes which was developed by Oak Ridge National Laboratory (ORNL) for the NRC to perform standardized computer analyses for licensing evaluations. SAS2 is a control module that provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. SAS2/ORIGEN-S calculates the time-dependent neutron flux and the buildup of fissile trans-uranium nuclides. It properly accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions. It can calculate accurately the neutron-activated products, the actinides and the fission products in a reactor core. SAS2/ORIGEN-S is used to develop the DCPP equilibrium core activity inventory and the decayed fuel inventories after shutdown (for the FHA). SAS2/ORIGEN-S has been used in prior AST licensing applications (e.g., Beaver Valley Power Station [ML032530204], Fort Calhoun Power Station [ML013030027]) to develop the core inventory, and its results accepted by the NRC.
ACTIVITY2	ACTIVITY2, "Fission Products in a Nuclear Reactor" – CB&I S&W Inc. Proprietary QA Category I Computer Code NU-014, V01, L03. (Reference 61) ACTIVITY2 calculates the concentration of fission products in the fuel, coolant, waste gas decay tanks, ion exchangers, miscellaneous tanks, and release lines to the atmosphere for a pressurized water reactor system. The program uses a library of properties of more than 100 significant fission products and may be modified to include as many as 200 nuclides. The program output presents the activity and energy spectrum at the selected part of the system for any specified operating time. ACTIVITY2 is used to develop the DCPP reactor coolant activity inventory (design and as limited by the plant Technical Specifications) ACTIVITY2 has been used in prior AST licensing applications (e.g., Beaver Valley Power Station [ML032530204], Fort Calhoun Power Station [ML013030027]) to develop the primary coolant inventory, and its results accepted by the NRC.

Code Name	Description of Code Use
IONEXCHANGER	IONEXCHANGER, - CB&I S&W Inc. Proprietary QA Category I Computer Code NU-009, Ver. 01, Lev. 03. (Reference 62) IONEXCHANGER calculates the activity of nuclides in an ion exchanger or tank of a nuclear reactor plant by solving the appropriate growth-decay-purification equations. Based on a known feed rate of primary coolant or other fluid with known radionuclide activities, it calculates the activity of each nuclide and its products in the ion exchanger or tank at some later time. The program also calculates the specific gamma activity for each of the seven fixed energy groups. IONEXCHANGER is used to develop the DCPP secondary coolant activity inventory (design and as limited by the plant Technical Specifications) IONEXCHANGER has been used in prior AST licensing applications (e.g., Beaver Valley Power Station [ML032530204], Fort Calhoun Power Station [ML013030027]) to develop the secondary coolant inventory, and its results accepted by the NRC.
Atmospheric Dispersion Factors	EN-113, "Atmospheric Dispersion Factors" – CB&I S&W Inc. Proprietary QA Category I Computer Code EN-113, V06, L08. (Reference 63) EN-113 calculates χ/Q values at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) following the methodology and logic outlined in NRC Regulatory Guide 1.145. The program can handle single or multiple release points for a specified time period and set of site-specific and plant-specific parameters. A release point can be identified as either of two types of release (i.e., ground or elevated), time periods for which sliding averages are calculated (i.e., 1 to 624 hours and/or annual average), applicable short-term building wake effect, meandering plume, long-term building height wake effect, and a wind speed value to be assigned to calm conditions. Downwind distances can be assigned for each of the sixteen 22.5-degree sectors for two irregular boundaries and for ten additional concentric boundaries used only in the annual average calculation. EN-113 performs the same calculations as the NRC PAVAN code except that EN-113 calculates χ/Q values for the various averaging periods directly using hourly meteorological data whereas PAVAN uses a joint frequency distribution of wind speed, wind direction, and stability class. EN-113 is used to develop the DCPP site boundary atmospheric dispersion factors. EN-113 has been used in prior AST licensing applications (e.g., Fort Calhoun Power Station [ML013030027]), and its results accepted by the NRC.
ARCON96	ARCON96, "Atmospheric Relative Concentrations in Building Wakes" – CB&I S&W Inc. QA Category I Computer Code EN-292, V00, L00. (Reference 64) ARCON96 was developed by Pacific Northwest National Laboratory (PNNL) for the Nuclear Regulatory Commission (NRC) to calculate relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point. ARCON96 has the ability to evaluate ground-level, vent, and elevated stack releases; it implements a straight-line Gaussian dispersion model with dispersion coefficients that are modified to account for low wind meander and building wake effects. The methodology is also able to evaluate diffuse and area source releases using the virtual point source technique, wherein initial

Code Name	Description of Code Use
	values of the dispersion coefficients are assigned based on the size of the diffuse or area source. Hourly, normalized concentrations (χ/Q) are calculated from hourly meteorological data. The hourly values are averaged to form χ/Qs for periods ranging from 2 to 720 hours in duration. The calculated values for each period are used to form cumulative frequency distributions. ARCON96 is used to develop the DCPP on-site control room and technical support center atmospheric dispersion factors. ARCON96 has been used extensively by the nuclear power industry in prior AST licensing applications and its results accepted by the NRC.
SWNAUA	SWNAUA, "Aerosol Behavior in Condensing Atmosphere", CB&I S&W Inc. Proprietary QA Category I Computer Code NU-185, V02, L00. (Reference 65) SWNAUA is a derivative of industry computer code NAUA/Mod 4 which was originally developed in Germany and was based on experimental data. NAUA/Mod 4 addressed particulate aerosol transport and removal following a LOCA at an LWR. It developed removal coefficients to address physical phenomena such as gravitational settling (also called gravitational sedimentation), diffusion, particle growth due to agglomeration, etc. using time-dependent airborne aerosol mass. NAUA4 (included in the NRC Source Term Code Package) was used by NRC during the initial evaluations of post-TMI data. S&W modified NAUA/Mod 4 to include spray removal and diffusiophoretic effects suitable for design basis accident analyses. A version of SWNAUA (SWNAUA-HYGRO) was proven to be the most reliable of more than a dozen international entries, in making predictions of aerosol removal for the LWR Aerosol Containment Experiments (LACE) series. SWNAUA is used to develop the time dependent post LOCA particulate aerosol removal coefficients in the sprayed and unsprayed regions of DCPP containment. SWNAUA has been used in prior AST applications (e.g., the design certification of CE System 80+, and for operating nuclear plants Beaver Valley Power Station [ML032530204] and Fort Calhoun Station [ML013030027]) and its results accepted by the NRC.
RADTRAD 3.03	RADTRAD 3.03 "A Simplified Model for RADionuclide Transport and Removal And Dose Estimates" – CB&I S&W Inc. QA Category I computer code No. NU-232, Version 3.03, Level (NA). (Reference 66) RADTRAD 3.03 is a NRC sponsored program, developed by Sandia National Labs (SNL). It can be used to calculate radiological doses to the public, plant operators and emergency personnel due to environmental releases that resulting from postulated design basis accidents at light water reactor (LWR) power plants The RADTRAD 3.03 (GUI Interface Mode) includes models for a variety of processes that can attenuate and/or transport radionuclides. It can model sprays and natural deposition that reduce the quantity of radionuclides suspended in the containment or other compartments. It can model the flow of radionuclides between compartments within a building, from buildings into the environment, and from the environment into a Control Room (CR)). These flows can be through filters, piping, or simply due to air leakage. RADTRAD 3.03 can also model radioactive decay and in-growth of daughters. Ultimately the program

Code Name	Description of Code Use
	calculates the Thyroid and TEDE dose (rem) to the public located offsite and to onsite personnel located in the CR due to inhalation and submersion in airborne radioactivity based on user specified, fuel inventory, nuclear data, dispersion coefficients, and dose conversion factors. RADTRAD is used to develop the TEDE dose to the public located offsite and to onsite personnel located in the CR due to inhalation and submersion in airborne radioactivity following the DCPP accidents listed in Chapter 1.0. RADTRAD has been used extensively by the nuclear power industry in prior AST licensing applications and its results accepted by the NRC.
PERC2	PERC2, "Passive Evolutionary Regulatory Consequence Code" – CB&I S&W Inc. Proprietary QA Category I Computer Code, NU-226, V00, L02. (Reference 67) PERC2 is a multi-region activity transport and radiological dose consequence program. It includes the following major features: • Provision of time-dependent releases from the reactor coolant system to the containment atmosphere. • Provision for airborne radionuclides for both TID and AST release assumptions, including daughter in growth. • Provision for calculating the CEDE to individual organs as well as EDE from inhalation, DDE and beta from submersion, and TEDE. • Provisions for tracking time-dependent inventories of all radionuclides in all control regions of the plant model. • Provision for calculating instantaneous and integrated gamma radiation source strengths as well as activities for the inventoried radionuclides to permit direct assessment of the dose from contained / or external sources for equipment qualification, vital area access and CR/TSC and EAB direct shine dose estimates. PERC2 is used to calculate the accident energy release rates and integrated gamma energy releases versus time for the various DCPP post-LOCA external and contained radiation sources. This source term information is input into SW_QADCGGP to develop the direct shine dose to the CR. PERC2 is also used to develop the decay heat in the RWST and MEDT and develop the TEDE dose to personnel located in the technical support center due to inhalation and submersion in airborne radioactivity following LOCA. PERC2 has been used in prior AST licensing applications (e.g., Beaver Valley Power Station [ML032530204], Fort Calhoun Power Station [ML013030027]), and its results accepted by NRC.
SW-QADCGGP	SW-QADCGGP, "A Combinatorial Geometry Version of QAD-5A" – CB&I S&W Inc. Proprietary QA Category I Computer Code, NU-222, V00, L02. (Reference 68) SW-QADCGGP is a variant of the QAD point kernel shielding program originally written at the Los Alamos Scientific Laboratory by R. E. Malenfant. The QADCGGP version implements combinatorial geometry and the geometric progression build-up factor algorithm. The SW-QADCGGP implements a graphical indication of the status of the computation process.

Code Name	Description of Code Use
	SW-QADCGGP is used to develop the direct shine dose to the DCPP CR, TSC and EAB. SW-QADCGGP has been used in prior AST licensing applications (e.g., Beaver Valley Power Station [ML032530204], Fort Calhoun Power Station [ML013030027]), and its results accepted by NRC.
GOTHIC	GOTHIC, "Generation of Thermal-Hydraulic Information for Containments", CB&I S&W Inc. QA Category I computer code No. ME-376, Version 8.0, Lev (NA). (Reference 69) GOTHIC is developed and maintained by Numerical Applications Incorporated (NAI) and is an integrated, general purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. GOTHIC solves the conservation equations for mass, momentum and energy for multicomponent, multi-phase flow in lumped parameter and/or multi-dimensional geometries. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal non equilibrium between phases and unequal phase velocities, including countercurrent flow. Other phenomena include models for commonly available safety equipment, heat transfer to structures, hydrogen burn and isotope transport. GOTHIC is used to estimate the containment and sump pressure and temperature response with recirculation spray, the temperature transient in the DCPP RWST / MEDT gas and liquid due to incoming sump water leakage / inflow / decay heat from the RWST / MEDT fission product inventory, and the volumetric release fraction transient from the RWST / MEDT gas space to the environment. GOTHIC has been used in prior AST licensing applications for this purpose (Beaver Valley Power Station [ML072780163]), and its results accepted by NRC.

4.0 RADIATION SOURCE TERMS

4.1 Core Activity Inventory

The guidance provided in Section 3.1 of RG 1.183 indicates that the inventory of fission products in the reactor core available for release to the containment following an accident should reflect maximum full power operation of the core with the current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty in the 10CFR50 Appendix K analysis (typically 1.02).

In accordance with the above guidance, the isotopic inventory of fission products in the DCPP reactor core developed in support of implementation of AST is conservatively based on plant operation at 105% of the current licensed rated thermal power of 3411 MWth, with current licensed values of fuel enrichment and fuel burnup.

The DCPP equilibrium core inventory is calculated using computer code ORIGEN-S. The calculation is performed by utilizing Control Module SAS2 of the SCALE 4.3 computer code package. The SAS2 control module provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. It calculates the time-dependent neutron flux and the buildup of fissile trans-uranium nuclides. It accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions with materials in the core. It calculates the neutron-activated products, the actinides, and the fission products in a reactor core.

The DCPP reactor core consists of 193 fuel assemblies with various Uranium-235 enrichments. Per DCPP core-reload design documentation, the peak rod burnup limit at the end of cycle is not allowed to exceed 62,000 MWD/MTU. The current licensed maximum value for fuel enrichment is 5.0%. To account for variation of U-235 enrichment in fresh fuel, the radionuclide inventories were calculated for a 4.2% average enriched core (representing minimum enrichment at DCPP), and 5% average enriched core (representing maximum enrichment). The higher activity for each isotope from the above two enrichment cases is chosen to represent the inventory of that isotope in the equilibrium core.

The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with three different burnups, i.e., approximately 1/3 of the core is subjected to one fuel cycle, 1/3 of the core to two fuel cycles and 1/3 of the core to three fuel cycles. This approach has been demonstrated to develop an isotopic core inventory that is a reasonable and conservative approximation of a core inventory developed using DCPP specific fuel management history data.

A 19 month fuel cycle length was utilized in the analysis. The 19-month average fuel cycle is an artifact of the current DCPP fuel management scheme which specifies 3 fuel cycles every 5 years and refueling outages in Spring or Fall. Minor variations in fuel irradiation time and duration of refueling outages will have a slight impact on the estimated inventory of long-lived isotopes in the core. However, these inventory changes will have an insignificant impact on the radiological consequences of postulated accidents. Regardless, a 4% margin has been

included in the final isotopic radioactive inventories in support of bounding analyses and to address minor changes in future fuel management schemes.

In summary, the equilibrium isotopic core average inventory is based on

- A power level of 3580 MWth inclusive of power uncertainty
- A range of enrichment of 4.2 to 5.0 w % U-235. Use of a few assemblies with lower enrichment is a common industry practice when replacing assemblies previously irradiated but proven unsuitable for continued irradiation. As these assemblies are designed to replace higher enrichment assemblies with ones of similar reactivity for the remainder of the fuel cycle, their inventory is enveloped by the isotopic core average inventory developed to support the dose consequence analyses
- A maximum core average burnup of 50 GWD/MTU

The core inventory developed by ORIGEN-S using the above methodology includes over 800 isotopes. The DCPP equilibrium core fission product inventory of dose significant isotopes relative to LWR accidents is presented in Table 4.1-1.

4.2 Coolant Activity Inventory

Design Basis Primary and Secondary Coolant Activity Concentrations

CB&I S&W Inc. computer code ACTIVITY2, is used to calculate the design basis primary coolant activity concentrations for both DCPP Units 1 and 2 based on the core inventory developed using ORIGEN-S and discussed in Section 4.1. The source terms for the primary coolant fission product activity include leakage from 1% fuel defects and the decay of parent and second parent isotopes. The depletion terms of the primary coolant fission product activity include radioactive decay, purification of the letdown flow and neutron absorption when the coolant passes the reactor core. The nuclear library includes 3rd order decay chains of approximately 200 isotopes.

CB&I S&W Inc. computer code IONEXCHANGER, is used to calculate the design basis halogen and remainder activity concentrations in the secondary side liquid. The source terms for the secondary side activity include the primary-to-secondary leakage in steam generators and the decay products of parent and second parent isotopes. The depletion terms of the secondary side liquid activity include radioactive decay, and purification due to the steam generator blowdown flow, and continuous condensate polishing.

The design basis noble gas concentrations in the secondary steam are calculated by dividing the appearance rate (µCi/sec) by the steam flow rate (gm/sec). The noble gas appearance rate in the steam generator steam space includes the primary-to-secondary leak contribution and the noble gas generation due to decay of halogens in the SG liquid. The activity concentrations of the other isotopes in the steam are determined by the SG liquid concentrations and the partition coefficients recommended in NUREG-0017, Revision 1. (Reference 25)

Technical Specification Primary and Secondary Coolant Activity Concentrations

DCPP Technical Specifications Limiting Condition for Operation (LCO) 3.4.16 limits the specific activity for iodines and noble gases in the primary coolant to 1 µCi/gm Dose Equivalent (DE) I-131, and 600 µCi/gm DE Xe-133 (corresponds to ~1% fuel defects), respectively. Technical Specifications LCO 3.7.18 limits the specific activity for iodines in the secondary liquid to 0.1 µCi/gm DE I-131.

In accordance with current licensing basis, the primary coolant technical specification activities for iodines are based on 1.0 µCi/gm DE I-131. As discussed earlier, with this application, PG&E proposes to reduce the TS LCO limit for the noble gases to 270 µCi/gm DE Xe-133 (corresponds to ~ 0.5 % fuel defects).

The Technical Specification (TS) based primary coolant isotopic activity reflects the following:

- Isotopic compositions based on the design basis primary coolant equilibrium concentrations at 1% fuel defects.
- Iodine concentrations based on the thyroid inhalation weighting factors* for I-131, I-132, I-133, I-134, and I-135 obtained from Federal Guidance Report 11 (Reference 19).
- Noble gas concentrations based on the submersion weighting factors for Xe-133, Xe-133m, Xe-135m, Xe-135, Xe-138, Kr-85m, Kr-87 and Kr-88 obtained from Federal Guidance Report 12 (Reference 20)

* The isotopic iodine concentrations in the primary and secondary coolant allowable by the Plant Technical Specifications are based on committed *thyroid* dose equivalent conversion factors from Table 2.1 of Federal Guidance 11. This selection is made in recognition of the fact that available physical data with respect to radiation damage resulting from inhalation of radioactive iodine is associated with a specific organ, i.e., the thyroid.

It is noted that use of the committed *effective* dose equivalent conversion factors from Table 2.1 of Federal Guidance 11 would predict slightly lower Technical Specification primary and secondary coolant iodine concentrations (by ~ 2%), which, when used in the accident analyses to estimate the releases, would result in slightly lower dose consequences (~2%).

It is acknowledged that defining the dose equivalent I-131 based on the committed effective dose has the advantage of being consistent with the post-accident dose consequences since with the implementation of AST, the dose is estimated in terms of TEDE. However, the approach used in defining DE I-131 based on the thyroid dose conversion factors is more conservative, and believed to be more appropriate since the thyroid dose is a more precisely defined physical quantity for the radio-toxicity of iodines.

In accordance with the requirements of Item 10 of RIS 2006-04 (Reference 70), the TS definition for Dose Equivalent I-131 will be updated to reflect the use of the committed thyroid dose equivalent conversion factors.

The Technical Specification 1 µCi/gm DE I-131 concentrations per nuclide in the primary coolant are calculated with the following equation:

$$DEI_{131}(i)\,(\mu Ci/gm) = \frac{C(i) \times CT_{tot}}{\sum\{F(i) \times C(i)\}}$$

Where:

$F(i)$ =	$DCF(i) / DCF_{I\text{-}131}$
$DCF(i)$=	FGR-11 Table 2-1 Thyroid Dose Conversion Factor per Nuclide (Rem/Ci)
$C(i)$ =	design basis primary coolant equilibrium iodine concentration per nuclide (µCi/gm)
CT_{tot} =	primary coolant total (DE I-131) technical specification iodine concentration (µCi/gm).

The CT_{tot} for the pre-accident iodine spike is 60 µCi/gm (transient TS limit for full power operation), or 60 times the primary coolant total iodine technical specification concentration.

The accident initiated iodine spike activities are based on an accident dependent multiplier times the equilibrium iodine appearance rate. The equilibrium appearance rates are conservatively calculated based on the technical specification reactor coolant activities, along with the maximum design letdown rate, maximum technical specification based allowed primary coolant leakage, and an assumed ion-exchanger iodine removal efficiency of 100%.

The TS secondary liquid iodine concentration is determined using methodology similar to that described above for the primary coolant where CT_{tot} is 0.1 µCi/gm DE I-131, and C(i) is the design basis secondary coolant equilibrium concentrations per nuclide.

The TS noble gas concentrations for the primary coolant are based on 270 µci/gm DE Xe-133. The DE Xe-133 for noble gases is calculated as follows:

$$DE\ X_{133} = \Sigma\{F(i) \times C(i)\}$$

Where:

F(i) = DCF(i) / DCF Xe-133

DCF(i) = EPA FGR-12 (1993), Table III.1, Dose Coefficient per Nuclide [(rem-m^3)/(Ci-sec)]

C(i) = design basis primary coolant equilibrium noble gas concentration per nuclide (µCi/gm)

The noble gas and halogen primary and secondary coolant Technical Specification Activity Concentrations for DCPP Units 1 and 2 are presented in Table 4.2-1. The pre-accident iodine spike concentrations and the equilibrium iodine appearance rates (utilized to develop accident initiated iodine spike values), are presented in Table 4.2-2.

4.3 Gap Fractions for Non-LOCA Events

RG 1.183, Rev 0, Table 3 provides the gap fractions for Non-LOCA events that are postulated to result in fuel damage. The referenced gap fractions are contingent upon meeting Note 11 of Table 3 of RG 1.183. Note 11 indicates that the release fractions listed in Table 3 are "acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU." As documented in NRC communications with other licensees, (Millstone, ML041320350), the burnup criterion associated with the maximum allowable linear heat generation rate is applicable to the peak rod average burnup in

any assembly and is not limited to assemblies with an average burnup that exceeds 54 GWD/MTU.

Diablo Canyon has three design basis non-LOCA accidents that are postulated to result in fuel damage, i.e., the Locked Rotor Accident (LRA), the Fuel Handling Accident (FHA) and the Control Rod Ejection Accident (CREA).

To support flexibility in future DCPP fuel management schemes and establish dose consequences that take into consideration fuel rods that may exceed the RG 1.183 R0, Table 3, Note 11 linear heat generation criteria, PG&E has elected to use the fuel gap fractions provided in Table 3 of DG-1199 (Reference 71) for all DCPP Non-LOCA events that are postulated to result in fuel damage with the exception of the CREA. This approach has been deemed acceptable (i.e., in lieu of developing DCPP specific fission gas release calculations using NRC approved methods and bounding power history to establish the gap fractions), since DCPP falls within, and intends to operate within, the maximum allowable power operating envelop for PWRs shown in Figure 1 of DG-1199.

In summary, the fuel gap activity fractions used to assess the dose consequences of the FHA and LRA are as follows:

Nuclide Group	FHA /LRA (based on DG-1199)
I-131	0.08
I-132	0.23
Kr-85	0.35
Other Noble Gases	0.04
Other Halogens	0.05
Alkali Metals	0.46

In accordance with RG 1.183 (Appendix H, and Note 11 of Table 3), the gap fraction associated with the CREA is as follows:

Noble Gases:	10%
Halogens:	10%

Refer to the following Tables for the isotopic activity in the gap for Non-LOCA events

Table 4.3-1 - LRA and CREA
Table 7.3-2 - FHA

Table 4.1-1
DCPP Equilibrium Core Inventory (Power Level: 3580 MWth)
Dose Significant Isotopes including the Parent, Grandparent and 2nd Parent Isotopes

ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)
AG-110	2.67E+07	IN-125	8.46E+05	**SB-125**	9.63E+05
AG-110M	6.92E+05	IN-127	1.86E+06	**SB-127**	9.14E+06
AG-111	7.09E+06	IN-129	3.55E+06	**SB-129**	3.25E+07
AG-111M	7.09E+06	IN-131	1.09E+06	**SB-130**	1.08E+07
AG-112	3.16E+06	IN-132	2.85E+05	**SB-130M**	4.38E+07
AG-115	6.21E+05	**KR-83M**	1.14E+07	**SB-131**	7.67E+07
AG-115M	2.60E+05	**KR-85**	1.11E+06	**SB-132**	4.70E+07
AM-239	4.90E-01	**KR-85M**	2.33E+07	**SB-132M**	4.37E+07
AM-241	1.32E+04	**KR-87**	4.65E+07	**SB-133**	6.32E+07
AM-242	9.40E+06	**KR-88**	6.43E+07	SB-134	1.14E+07
AM-242M	8.54E+02	**KR-89**	7.94E+07	SB-135	5.46E+06
AM-243	5.28E+03	**KR-90**	8.48E+07	SB-136	8.63E+05
AM-244	3.79E+07	KR-91	5.83E+07	**SE-83**	5.38E+06
AM-245	1.12E-03	KR-92	3.12E+07	SE-83M	5.65E+06
AS-76	3.05E+03	KR-93	1.07E+07	SE-84	2.04E+07
AS-83	7.02E+06	KR-94	5.00E+06	SE-85	9.54E+06
BA-137M	1.30E+07	**LA-140**	1.85E+08	SE-87	1.32E+07
BA-139	1.76E+08	**LA-141**	1.61E+08	SE-88	7.15E+06
BA-140	1.78E+08	**LA-142**	1.57E+08	SE-89	2.49E+06
BA-141	1.59E+08	**LA-143**	1.48E+08	**SM-153**	6.04E+07
BA-142	1.51E+08	LA-144	1.31E+08	SM-155	4.30E+06
BA-143	1.29E+08	**MO-99**	1.84E+08	SM-156	2.66E+06
BA-144	9.93E+07	**MO-101**	1.69E+08	SM-157	1.70E+06
BR-82	4.44E+05	MO-103	1.62E+08	**SN-121**	8.43E+05
BR-82M	3.88E+05	MO-104	1.33E+08	**SN-123**	6.43E+04
BR-83	1.13E+07	MO-105	9.97E+07	**SN-125**	5.25E+05
BR-84	2.10E+07	MO-106	5.86E+07	SN-125M	1.58E+06
BR-85	2.31E+07	NB-101	1.59E+08	**SN-127**	3.69E+06
BR-87	3.67E+07	NB-104	5.10E+07	SN-127M	4.95E+06
BR-88	3.52E+07	**NB-95**	1.66E+08	SN-129	1.28E+07
BR-89	2.45E+07	**NB-95M**	1.89E+06	SN-129M	1.17E+07
BR-90	1.35E+07	**NB-97**	1.59E+08	SN-130	3.28E+07
CD-115	9.17E+05	**NB-97M**	1.50E+08	SN-131	2.83E+07
CD-115M	4.43E+04	NB-99	1.07E+08	SN-132	2.28E+07

Table 4.1-1
DCPP Equilibrium Core Inventory (Power Level: 3580 MWth)
Dose Significant Isotopes including the Parent, Grandparent and 2nd Parent Isotopes

ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)
CD-121	7.67E+05	NB-99M	7.35E+07	SN-133	6.21E+06
CE-141	1.63E+08	**ND-147**	6.59E+07	SN-134	1.07E+06
CE-143	1.50E+08	ND-149	3.90E+07	**SR-89**	9.05E+07
CE-144	1.26E+08	ND-151	2.08E+07	**SR-90**	9.67E+06
CE-147	6.18E+07	NP-238	6.20E+07	**SR-91**	1.13E+08
CF-249	2.46E-02	**NP-239**	2.16E+09	**SR-92**	1.22E+08
CM-241	3.54E+00	NP-240	6.23E+06	**SR-93**	1.39E+08
CM-242	5.88E+06	**PD-109**	4.73E+07	**SR-94**	1.39E+08
CM-244	1.31E+06	PD-109M	3.12E+05	SR-95	1.25E+08
CM-245	1.26E+02	PD-111	7.09E+06	SR-97	4.68E+07
CO-58**	0.00E+00	PD-112	3.14E+06	**TB-160**	1.87E+05
CO-60**	0.00E+00	PD-115	7.84E+05	**TC-99M**	1.63E+08
CS-132	5.75E+03	**PM-147**	1.68E+07	**TC-101**	1.69E+08
CS-134	2.41E+07	**PM-148**	1.88E+07	TC-103	1.65E+08
CS-134M	5.63E+06	**PM-148M**	2.83E+06	**TC-104**	1.40E+08
CS-136	7.01E+06	**PM-149**	6.43E+07	**TC-105**	1.18E+08
CS-137	1.37E+07	**PM-151**	2.10E+07	TC-106	8.80E+07
CS-138	1.85E+08	PM-153	9.77E+06	**TE-127**	9.03E+06
CS-139	1.72E+08	**PR-142**	9.47E+06	**TE-127M**	1.52E+06
CS-140	1.54E+08	**PR-143**	1.47E+08	**TE-129**	3.10E+07
CS-141	1.17E+08	**PR-144**	1.27E+08	**TE-129M**	6.30E+06
CS-142	6.80E+07	PR-144M	1.76E+06	**TE-131**	8.28E+07
CS-143	3.41E+07	PR-147	6.52E+07	**TE-131M**	2.04E+07
DY-166	4.91E+02	PR-149	3.57E+07	**TE-132**	1.41E+08
EU-154	9.00E+05	PR-151	1.23E+07	**TE-133**	1.09E+08
EU-155	3.83E+05	**PU-238**	5.22E+05	**TE-133M**	8.93E+07
EU-156	3.90E+07	**PU-239**	3.06E+04	**TE-134**	1.75E+08
EU-157	4.12E+06	**PU-240**	4.87E+04	TE-135	9.68E+07
EU-158	1.01E+06	**PU-241**	1.36E+07	TE-136	4.29E+07
EU-159	5.15E+05	**PU-242**	3.34E+02	TE-137	1.45E+07
GA-72	1.71E+03	PU-243	7.36E+07	TE-138	3.65E+06
GA-77	1.66E+05	RA-224	5.16E-01	**TH-228**	5.14E-01
GD-159	8.91E+05	**RB-86**	2.50E+05	U-239	2.17E+09
GE-77	6.48E+04	RB-86M	2.07E+04	**XE-131M**	1.42E+06

Table 4.1-1
DCPP Equilibrium Core Inventory (Power Level: 3580 MWth)
Dose Significant Isotopes including the Parent, Grandparent and 2nd Parent Isotopes

ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)	ISOTOPE*	ACTIVITY (CURIES)
GE-77M	1.70E+05	**RB-88**	6.60E+07	**XE-133**	2.01E+08
GE-83	1.24E+06	**RB-89**	8.57E+07	**XE-133M**	6.42E+06
H-3	6.10E+04	**RB-90**	7.86E+07	**XE-135**	4.92E+07
HO-166	2.58E+04	**RB-90M**	2.53E+07	**XE-135M**	4.30E+07
I-129	4.00E+00	RB-91	1.05E+08	**XE-137**	1.84E+08
I-130	3.58E+06	RB-92	9.35E+07	**XE-138**	1.70E+08
I-130M	1.92E+06	RB-93	7.89E+07	XE-139	1.25E+08
I-131	9.90E+07	RB-94	4.13E+07	XE-140	8.66E+07
I-132	1.44E+08	RB-95	2.01E+07	XE-142	1.33E+07
I-133	2.01E+08	**RH-103M**	1.66E+08	**Y-90**	1.02E+07
I-134	2.22E+08	**RH-105**	1.08E+08	Y-90M	7.71E+02
I-134M	2.07E+07	**RH-105M**	3.43E+07	**Y-91**	1.19E+08
I-135	1.92E+08	**RH-106**	7.53E+07	**Y-91M**	6.57E+07
I-136	8.73E+07	RH-109	3.65E+07	**Y-92**	1.23E+08
I-137	9.40E+07	RN-220	5.16E-01	**Y-93**	9.41E+07
I-138	4.80E+07	**RU-103**	1.66E+08	**Y-94**	1.50E+08
I-139	2.22E+07	**RU-105**	1.21E+08	**Y-95**	1.57E+08
I-140	6.06E+06	**RU-106**	6.68E+07	Y-97	1.26E+08
IN-115M	9.17E+05	RU-109	3.16E+07	ZN-72	1.71E+03
IN-121	7.55E+04	**SB-122**	1.57E+05	ZR-101	9.55E+07
IN-121M	7.82E+05	SB-122M	1.57E+04	**ZR-95**	1.65E+08
IN-123	6.87E+05	**SB-124**	1.21E+05	**ZR-97**	1.58E+08
		SB-124M	2.34E+03	ZR-99	1.66E+08

Note:

* Isotopes in **Bold** Font are dose-significant for inhalation, submersion and direct shine. The parent, grandparent and second parent of the isotopes in **Bold** Font are also required to address daughter product in-growth.

The group of isotopes needed to determine the "submersion and inhalation" dose in the Control Room and at the Site Boundary is typically a subset of the isotopes listed above in **bold** font, and represent a small group of reasonably long half-life isotopes with significant inhalation dose conversion factors which dominate the TEDE dose.

To determine the TEDE resulting from inhalation and submersion following a LOCA, the DCPP LOCA dose consequence analysis uses the default group of 60 isotopes provided with computer code RADTRAD 3.03 plus 13 additional nuclides that were deemed to be dose significant (i.e., Br-82, Br-84, Rb-88, Rb-89, Te-133, Te-133m, Te-134, I-130, Xe-131m, Xe-133m, Xe-138, Cs-138 and Np-238).

** Co-58 / Co-60 are activation products that are developed external to the core and typically do not appear in the equilibrium core inventory.

Table 4.2-1 Primary and Secondary Coolant Technical Specification Activity Concentrations		
Nuclide	**Primary Coolant (µCi/gm)**	**Secondary Coolant (µCi/gm)**
Kr-83M	1.87E-01	-----
Kr-85M	6.60E-01	-----
Kr-85	5.60E+00	-----
Kr-87	4.41E-01	-----
Kr-88	1.22E+00	-----
Xe-131M	1.88E+00	-----
Xe-133M	1.92E+00	-----
Xe-133	1.29E+02	-----
Xe-135M	4.07E-01	-----
Xe-135	3.76E+00	-----
I-131	7.87E-01	8.06E-02
I-132	3.00E-01	1.94E-02
I-133	1.16E+00	1.08E-01
I-134	1.67E-01	4.78E-03
I-135	6.68E-01	5.09E-02

Table 4.2-2 Primary Coolant Pre-Accident Iodine Spike Concentrations & Equilibrium Iodine Appearance Rates		
Nuclide	Pre-Accident Spike RCS Concentrations (60 µCi/gm DE I-131) (µCi/gm)	Equilibrium Iodine Activity Appearance Rates into RCS (µCi/sec)
I-131	47.2	7.18E+03
I-132	17.9	7.78E+03
I-133	69.5	1.25E+04
I-134	10.0	8.91E+03
I135	40.1	9.91E+03

Table 4.3-1
Isotopic Gap Activity
Locked Rotor Accident / Control Rod Ejection Accident

Nuclide	Core Activity (Ci)	Fraction of Core Activity in Gap LRA	Core Gap Activity w/o Peaking Factor (Ci) LRA	Fraction of Core Activity in Gap CREA	Core Gap Activity w/o Peaking Factor (Ci) CREA
KR-85	1.11E+06	0.35	3.89E+05	0.10	1.11E+05
KR-85M	2.33E+07	0.04	9.32E+05	0.10	2.33E+06
KR-87	4.65E+07	0.04	1.86E+06	0.10	4.65E+06
KR-88	6.43E+07	0.04	2.57E+06	0.10	6.43E+06
Xe-131M	1.42E+06	0.04	5.68E+04	0.10	1.42E+05
Xe-133M	6.42E+06	0.04	2.57E+05	0.10	6.42E+05
XE-133	2.01E+08	0.04	8.04E+06	0.10	2.01E+07
XE-135	4.92E+07	0.04	1.97E+06	0.10	4.92E+06
Xe-138	1.70E+08	0.04	6.80E+06	0.10	1.70E+07
I-130	3.58E+06	0.05	1.79E+05	0.10	3.58E+05
I-131	9.90E+07	0.08	7.92E+06	0.10	9.90E+06
I-132	1.44E+08	0.23	3.31E+07	0.10	1.44E+07
I-133	2.01E+08	0.05	1.01E+07	0.10	2.01E+07
I-134	2.22E+08	0.05	1.11E+07	0.10	2.22E+07
I-135	1.92E+08	0.05	9.60E+06	0.10	1.92E+07
BR-82	4.44E+05	0.05	2.22E+04	0.10	4.44E+04
BR-84	2.10E+07	0.05	1.05E+06	0.10	2.10E+06
CS-134	2.41E+07	0.46	1.11E+07	-	-
CS-136	7.01E+06	0.46	3.22E+06	-	-
CS-137	1.37E+07	0.46	6.30E+06	-	-
CS-138	1.85E+08	0.46	8.51E+07	-	-
RB-86	2.50E+05	0.46	1.15E+05	-	-
Rb-88	6.60E+07	0.46	3.04E+07	-	-
Rb-89	8.57E+07	0.46	3.94E+07	-	-

Note: Values reported reflect the core isotopic gap activity assumed for the LRA and CREA. These values have to be adjusted for a) the failed fuel percentage (10%) and b) peaking factor (1.65), prior to assessing the associated dose consequences.

For the isotopic gap activity associated with the FHA refer to Table 7.3-2.

5.0 ACCIDENT ATMOSPHERIC DISPERSION FACTORS (χ/Q)

The DCPP meteorological measurement program is described in DCPP UFSAR Section 2.3.3. The meteorological program was designed to meet the requirements of Safety Guide 23, February 1972 (Reference 59). The program consists of monitoring wind speed, wind direction, ambient temperature, and precipitation. Operation of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere.

5.1 Exclusion Area Boundary and Low Population Zone Atmospheric Dispersion Factors

Atmospheric dispersion factors (i.e., χ/Qs) are calculated at the DCPP EAB and LPZ for post-accident environmental releases originating from DCPP Units 1 and 2.

The applicable methodology is identified in RG 1.145 for ground level releases (Reference 5). The methodology is implemented using CB&I S&W Inc. computer program "Atmospheric Dispersion Factors" using a continuous temporally representative 5-year period of hourly meteorological data from the DCPP onsite meteorological tower (i.e., January 1, 2007 through December 31, 2011).

The Regulatory Guide 1.145 methodology for ground level sources is as follows:

$$\chi/Q_1 = \{(\bar{u})[(\pi)(\sigma_y)(\sigma_z) + (A/2)]\}^{-1}$$
$$\chi/Q_2 = [(u)(3\pi)(\sigma_y)(\sigma_z)]^{-1}$$
$$\chi/Q_3 = [(u)(\pi)(\Sigma_y)(\sigma_z)]^{-1}$$

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 5.

Per the guidance provided in RG 1.145, 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 a DCPP site boundary drawing (Figure 5.1-1), taking into consideration a 45-degree azimuth sector centered on each 22.5° azimuth sector as described in RG 1.145, 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. The release point locations are listed in Table 5.2-1.

As shown in UFSAR Figure 2.1-2, on land, the DCPP EAB is marked by a farm type fence. As demonstrated by Table 5.1-2, the location of the "as modeled" continuous EAB over the ocean is based on an extension of the arc (shown in UFSAR Figure 2.1-2 in the northern direction) over the water until it once again touches the shore line, at which point it follows the shoreline till it intersects with the farm type fence that defines the EAB on land in the southern direction.

The above approach in defining the EAB over the water is conservative since it remains well within the 2000-yard radius offshore security zone maintained by the U.S. coast guard in the south-southeast clockwise through the west-northwest direction sectors.

An LPZ distance of 6 miles (9,654 meters) is used in the analysis. The use of one LPZ distance in all 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.

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 is identified in RG 1.111, Regulatory Position C.1.c.(Reference 26) 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 enhancement of vertical turbulence (sector averaging eliminates the horizontal component) is only a function of building height, which has been credited for the containment (i.e., 66.5 meters is based on the height of the containment structure above grade (i.e., 218 feet).

The building wake effect cross sectional area of 2745 m^2 is based on the geometry of the containment building (CB). The CB area that has an effect on the dispersion of the releases is its entire cross-sectional area. The containment has a radius of 73 ft and is shaped like a cylinder from the base at El 85 ft to an elevation of 230 ft. It is essentially a hemisphere shape from El 230 ft to 303 ft. The cross-sectional area is calculated as the sum of the cylinder and hemisphere:

Cylinder: (230 ft - 85 ft) x (2 x 73 ft) = 21,170 ft^2

Hemisphere: [π x (303 ft- 230 ft)^2] / 2 = 8,370.8 ft^2 (conservative approximation)

Total containment cross-sectional area = 21,170 ft^2 + 8,370.8 ft^2 = 29,540.8 ft^2 or 2,744.5 m^2.

The following conservative assumptions are made for these calculations:

- All releases are treated as point sources
- All releases are treated as ground-level as there are no release conditions that are high enough 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
- The terrain adjustment factor (TAF) used in the calculation of the annual average χ/Q values for the EAB and LPZ models are 4.0 and 1.25, respectively, and are based on the default open TAF values presented as a function of distance in Figure 4.2 of NUREG/CR 2858. (Reference 72)
- Radioactive decay or plume depletion due to deposition is not considered

Table 5.1-2 provides the sixteen sector-specific 0.5% 0-2 hour X/Q values from the Unit 1 EAB, Unit 2 EAB and LPZ model runs, the sixteen sector-dependent annual average X/Q values from the U1/U2 (LPZ) model run, and the sector-specific distances to the EAB relative to both units.

Table 5.1-3 provides the 5-percent overall site 0-2 hour X/Q values for the U1 (EAB) and the U2 (EAB), and the intermediate, short-term X/Q values (i.e., 2-8 hours, 8-24 hours, 1-4 days, and 4-30 days) from the U1/U2 (LPZ)

The highest EAB & LPZ χ/Q values from among all 22.5° downwind sectors for each release/receptor combination and accident period are summarized in Table 5.1-1. 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.

5.2 On-Site Atmospheric Dispersion Factors

Regulatory Guide 1.194, June 2003 (Reference 21), Regulatory Position 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 are used to determine short-term on-site χ/Q values in support of design basis radiological habitability assessments.

In accordance with Regulatory Position C.2 of RG 1.194, the control room and technical support center χ/Q values for radiological releases from DCPP Unit 1 and Unit 2 were calculated using the NRC "**A**tmospheric **R**elative **CON**centrations in Building Wakes" (ARCON96) methodology as documented in NUREG/CR-6331, Revision 1 (Reference 6). 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., the distance and direction from the control room air intake to the release point, and intake height).

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) was used for the ARCON96 runs. Each hour of data, at a minimum, had 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 Safety Guide 23, (Reference 59) guidance.

The ARCON96 modeling utilized to establish the Unit 1 and Unit 2 χ/Q values follows the ground level release guidance of Regulatory Position C.3 of RG 1.194 (Reference 21) relative to

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

The cross-sectional areas of the Containment Buildings, Refueling Water Storage Tanks, and Fuel Handling Buildings for DCPP Units 1 and 2 as input to the ARCON96 dispersion modeling analysis are based on the geometry of the site buildings as discussed below:

Containment Building (CB): The CB area that has an effect on the dispersion of the releases is its entire cross-sectional area. The containment has a radius of 73 ft and is shaped like a cylinder from the base at El 85 ft to an elevation of 230 ft. It is a hemisphere shape from El 230 ft to 303 ft. The cross-sectional area is the sum of the cylinder and hemisphere. As discussed earlier the total containment cross-sectional area is estimated to be = 2,744.5 m^2.

Fuel Handling Building (FHB): The highest elevation of the FHB is 190 ft. The elevation of the containment penetration area leakage releases is 140 ft. The FHB elevation relative to these release points is 190 ft – 140 ft = 50 ft. The diagonal length of the FHB is measured to be approximately 114 ft. This width is conservative in that the FHB is contained within a larger structure that could have been considered in this calculation. Therefore, the conservative cross-sectional area is given by:

(50 ft) x (114.2 ft) = 5,710 ft^2 / (10.76365 ft^2 / m^2) = 530.4 m^2.

Refueling Water Storage Tank (RWST): The RWST roof elevation is 173 ft and the grade elevation at the RWST is 115 ft. The RWST height relative to local grade is 173 ft – 115 ft = 58 ft. The diameter of the RWST is approximately 40 ft. Therefore, the cross-sectional area of the RWST is given by:

(58 ft) x (40 ft) = 2,320 ft^2 / 10.76365 ft^2 / m^2) = 215.5 m^2.

All releases were assumed to be ground-level as none of the release points at DCPP meet the definition of an elevated release as indicated in Regulatory Position C.3.2.2 of RG 1.194 (i.e., do not meet the requirement to be 2.5 times the height of plant buildings).

Only the containment building edge releases were treated as diffuse sources as the releases occur from the entire surface of the building. In these cases, initial values of the diffusion coefficients (σ_y, σ_z) were determined in accordance with the guidance provided in RG 1.194, June 2003, Regulatory Position C.3.2.4. Release and receptor locations are applied in accordance with the guidance provided in RG 1.194, Regulatory Position C.3.4 for building geometry and line-of-site distances (refer to Appendix A).

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 meter and
- Sector averaging constant = 4.3 (dimensionless).

The following assumptions are made for these 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 source/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 conservatively 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
- 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 inleakage 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 DCPP control room and technical support center χ/Q analyses, as applicable

- <u>Control room receptors within 10-meters of release</u>: In accordance with RG 1.194, Regulatory Position C.3.4, the ARCON96 methodology is not recommended for use at distances less than about 10 meters. Based on engineering judgment, the ARCON96 methodology has been applied for 2 cases when the distance from the release to the receptor is slightly less than 10 meters. Use of ARCON96 methodology for release point-receptors distances less than 10 meters (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 and not the normal atmospheric dispersion coefficients. Note that the χ/Q values for the above 2 cases were developed to establish the bounding χ/Q values; the

referenced χ/Q's were not the bounding values and therefore not used in the dose consequence analyses.

- Control room receptors at 1.5 meters from release: Since the Unit 1 and Unit 2 MSSVs, 10% ADVs, and MSL break 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 RG 1.194. Thus no χ/Qs are developed for the above release point / receptor combinations. The bullet below provides a discussion of the effect of atmospheric dispersion on releases from the MSSVs/10% ADVs with respect to the affected unit's control room normal intake. Atmospheric dispersion is not credited when determining the CR operator dose due to releases out of the MSL break location in the faulted steam generator. For further detail see Section 7.6.

- Energetic releases: The 95th-percentile high wind speed values for the 10-m and 76-m levels are 11.0 and 12.1 m/sec, respectively. The ratios of the vertical exit velocity of the releases from the MSSVs and 10% ADVs to the wind speed are 8.6 (i.e., 94.9 m/sec to 11.0 m/sec) and 7.8 (i.e., 94.9 m/sec to 12.1 m/sec) for the 10-m and 76-m tower levels, respectively, over the 5-year meteorological data base.. The large vertical velocities of the MSSV and 10% ADV releases (ranging from 98.9 - 94.9 m/sec), and orientation relative to the CR intakes, 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% ADV releases to Unit 1 CR normal intake) is not contaminated. Moreover, since the horizontal distance is only 1.5 meters, 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% ADV releases (i.e., 27.1 and 26.5 meters, respectively) relative to the height of the CR normal intakes (i.e., 22 meters). Plume rise calculations indicate that the MSSV and ADV release heights will be enhanced by 2 meters at the 95th percentile wind speed of 11.0 m/sec and 12.1 m/sec for the 10-m and 76-m tower levels, respectively, 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 CR intake of the opposite unit for releases from the MSSVs / 10% ADVs as the worst case CR normal intake location.

- Vertically-oriented energetic releases: Regulatory Position C.6 of NRC Regulatory Guide 1.194 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 until initiation of shutdown cooling at 10.73 hours of the accident, the reduction factor of 5 is applicable to the DCPP MSSV and 10% ADV 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 T= 8 to 10.73 hrs, continued use of the 2-8 hr χ/Q (with the factor of 5 reduction) is acceptable and conservative.

- Dual Intakes: The Unit 1 and Unit 2 control room emergency air intakes (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 emergency air intakes are never within the same wind direction window; defined as a wedge centered on the line of sight between the source and the receptor with the vertex located on 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 RG 1.194.

- Redundant radiation monitors: Per RG 1.194, Regulatory Position C.3.3.2.3, based on the dual intake design of the control room pressurization intakes, and as discussed later in Section 7.1, 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), the χ/Q values applicable to the more favorable CR 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 I Plant Vent: It has been determined that there are several PG&E Design Class II lines that connect to the PG&E Design Class I Plant Vent; specifically, a) the 40 inch containment penetration area (GE/GW) HVAC ventilation line, b) the 2-inch gas decay tank vent line, and c) the 16 inch gland steam condenser (GSC) / steam jet air ejector (SJAE) exhaust header. In addition, it has been determined that the Plant Vent Expansion Joint may experience a tear during a seismic event, however, the plant vent will remain intact and functional.

    a) The 40-inch Containment Penetration Area Ventilation line and the 2-inch gas decay tank vent line were originally designed as PG&E Design Class I, but were subsequently declassified to PG&E Design Class II. As noted in Section 2.2, the portion of the 40-inch Containment Penetration Area Ventilation line that connects to the Plant Vent will be re-classified as PG&E Design Class I.

    b) The 2-inch gas decay tank vent header connects to the plant vent at El 137'-6" on the North-East side / South–East side of the Unit 1 and Unit 2 containments, respectively. However, re-classification of the 2-inch gas decay tank vent line is not deemed to be the best solution. Returning the piping to Design Class I is not expected to result in any physical changes to the piping. It would, however result in the need for development of some additional documentation, and increased inspections of Class I piping, hangers and testing which would result in increased dose and burden to plant maintenance, engineering and in-service inspection personnel.

c) The GSC / SJAE 16 inch exhaust header 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 most likely occur at the interface of this line and the plant vent.

d) 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 potential release locations at the *interface point* of the GSC / SJAE 16 inch line or the 2 inch gas decay tank vent header with the plant vent, or at the plant vent expansion joint, indicates that the χ/Q values developed for the plant vent are either conservative or representative of these potential release locations. Thus the dose impact of a break at these locations is bounded by the dose consequence analyses.

PG&E has also evaluated the cumulative dose impact in the control room due to back flow from the Plant Vent out of a potential break in a) the 2-inch gas decay tank vent header and b) in the GSC / SJAE 16 inch exhaust header (or any other PG&E Design Class II piping connected to it), at a location *other than the point of interface* with the Plant Vent, and demonstrated that the associated dose contribution is insignificant.

The configuration of release point and receptor information used in the χ/Q calculations including all significant release and receptor location permutations, release and receptor heights above plant grade, cross-sectional building areas encountered by the release on its path to the receptor, and line-of-sight distance between release and receptor used in the ARCON96 calculations are summarized in Appendix A of this report. A drawing showing the locations of the release points and receptors is also provided in Appendix A (See Figure A-1). Also provided in Appendix A is the hourly 5-year on-site meteorological data from the DCPP onsite meteorological tower (i.e., January 1, 2007 through December 31, 2011) which was used in the ARCON96 analyses.

Table 5.2-1 provides the release point / receptor combinations evaluated. Tables 5.2-2 and 5.2-3 provide the control room χ/Q values for the individual release point-receptor combinations for Unit 1 and Unit 2, respectively.

Note that the specific CR χ/Q values used in each of the accident analyses are presented in the accident-specific Tables presented in Chapter 7. The χ/Q values selected for use in the Chapter 7 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 to identify the bounding χ/Q values and reflect the allowable adjustments / reductions in the values as discussed earlier in this section and summarized in the notes of Tables 5.2-2 and 5.2-3.

Table 5.2-4 presents the χ/Q values for the individual post-accident 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). In the interest of model simplification, the Unit 1 & 2 MSLB locations were also used to represent the Unit 1 and Unit 2 Main Steam Safety Valves (MSSVs) and 10 percent ADVs. This approach is acceptable since these release points are essentially co-located, but the MSLB location releases have the lowest elevation and is therefore the closest to the TSC. The Unit 1 and Unit 2 control room pressurization air intakes also serve the TSC during the post-accident pressurization mode. Thus, the χ/Qs presented in Tables 5.2-2 and 5.2-3 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.

Presence of the Simulator Building in the vicinity of the DCPP Onsite Meteorological Tower

PG&E conducted an analysis to determine whether the presence of the Simulator Building in the vicinity of the DCPP onsite meteorological tower is meaningfully affecting the spatial representativeness of the meteorological data used for atmospheric dispersion calculations. This was done in response to an NRC Request for Additional Information (RAI) dated July 6, 2010 concerning the distance of the location of the onsite meteorological tower being less than ten times the height of the Simulator Building (Regulatory Position C.3 of RG 1.23, Revision 1, Reference 28) and is documented here for completeness. The distance from the Simulator Building to the meteorological tower is just over seven times the height of the Simulator Building.

The assessment of the presence of the Simulator Building on the spatial representativeness of the meteorological data involved several steps.

- The first step established the Simulator Building region of influence (ROI) in terms of distance and upwind fetch azimuth, on the DCPP meteorological tower.
- The second step determined the source-receptor cases that are within the Simulator Building ROI which may be potentially affected by the aerodynamic effects of the Simulator Building on the local wind field.
- The third step involved an analysis comparing pre-building and post-building meteorological data base sigma theta averages based on work performed by Bellinger (Reference 29) and Call (Reference 30).
- The fourth step involved calculating and comparing the χ/Q values using ARCON96 (for the control room) and Regulatory Guide 1.145 methodology (for the EAB and LPZ) for the pre-Simulator Building and post-Simulator Building meteorological data bases for the source-receptor cases within the Simulator Building ROI.
- The fifth step involved calculating and comparing the χ/Q values using ARCON96 (for the control room) and Regulatory Guide 1.145 methodology (for the EAB and LPZ) for the pre-Simulator Building and post-Simulator Building meteorological data bases for the source-receptor cases that were outside the Simulator Building ROI.

These analyses were examined to determine whether any differences in sigma theta averages and calculated χ/Q values were systematic, as well as for excessively large and outside-the range-of-expected stochastic climatic variations.

A comparison of the χ/Q values *within the Simulator Building ROI* and relevant to the FHA and MSSV / 10% ADV / MSL Break releases, indicated that χ/Q values calculated with the pre-Simulator Building onsite meteorological data (1974 to 1978) are always slightly smaller than those calculated with the 2007-2011 onsite meteorological data for all relevant time periods. A comparison of the χ/Q values generated by the two different meteorological data bases for the LOCA releases, indicated that for the 0-2 hour period, all but one of the 16 cases showed a small decrease in χ/Q values when calculated with the pre-Simulator Building onsite meteorological data. For the longer post-LOCA time periods, the calculated χ/Qs demonstrated, for the most part, a combination of small decreases and small increases for the pre-Simulator Building onsite meteorological data versus the post-Simulator Building onsite meteorological data.

In order to better determine if the differences in χ/Q values developed using the pre-Simulator Building and post-Simulator Building meteorological data bases are primarily the result of expected climatological differences, the LOCA releases for those source-receptor combinations that were *outside the Simulator Building ROI* were also examined. The magnitude of the variation in the χ/Q values were found, for the most part, to be similar to that for the χ/Q values within the Simulator Building ROI, and the results reflected a similar combination of increases and decreases.

It was concluded that the differences in χ/Q values most likely reflect the climatological differences between the 1974-1978 and 2007-2011 data bases, as opposed to the limited aerodynamic effect of the Simulator Building on the wind flow at the meteorological tower location. If there is an effect of the Simulator Building on the meteorological data, it is overshadowed by climatological differences in the data bases.

It was therefore judged that the Simulator Building has little to no influence on the DCPP meteorological tower data as it is applied to Gaussian atmospheric dispersion modeling. Moreover, the heuristic uncertainties in Gaussian modeling, which are more than adequately compensated for by conservative assumptions of point releases, centerline calculations and 95% meteorological conditions, are larger than the differences in comparative χ/Q values from the meteorological databases.

Table 5.1-1 Site Boundary Atmospheric Dispersion Factors					
	χ/Q (sec/m^3)				
Receptor	**0 - 2 hours**	**2 - 8 hours**	**8 - 24 hours**	**1 - 4 days**	**4 - 30 days**
Unit 1 EAB	2.50E-04	-	-	-	-
Unit 2 EAB	2.17E-04	-	-	-	-
Unit 1/2 LPZ	2.00E-05	8.94E-06	6.14E-06	2.72E-06	8.48E-07

<u>Note</u>

1 An EAB χ/Q value of 2.5E-4 sec/m^3 is used for all release points.
2. The 0.5% sector dependent χ/Q values are presented in Table 5.1-2, with the maximum value applicable to all sectors being reported in Table 5.1-1 and used to establish dose consequences. The worst case downwind sector for the 0-2 hour period for all receptors is northwest. For Unit 1/2 LPZ the worst case sector for periods 2-8 hours and longer is southeast.

Table 5.1-2 EAB/LPZ Sector Dependent Distances & Atmospheric Dispersion Factors						
Downwind Sector	**Unit 1 EAB**		**Unit 2 EAB**		**Unit 1/2 LPZ***	
	0-2hr X/Q (sec/m^3)	**Distance (m)**	**0-2 hr X/Q (sec/m^3)**	**Distance (m)**	**0-2 hr X/Q (sec/m^3)**	**Annual Average X/Q (sec/m^3)**
S	7.77E-05	830	9.46E-05	730	4.73E-06	3.85E-08
SSW	8.39E-05	830	1.02E-04	730	4.99E-06	3.92E-08
SW	1.12E-04	780	1.22E-04	740	6.20E-06	4.69E-08
WSW	1.04E-04	780	1.04E-04	780	5.80E-06	3.66E-08
W	1.47E-04	750	1.38E-04	780	8.17E-06	4.43E-08
WNW	2.02E-04	750	1.89E-04	780	1.38E-05	7.81E-08
NW	2.50E-04	750	2.17E-04	830	2.00E-05	1.31E-07
NNW	2.17E-04	750	1.88E-04	830	1.49E-05	8.83E-08
N	1.46E-04	730	1.19E-04	830	7.19E-06	4.70E-08
NNE	1.16E-04	730	9.53E-05	830	4.76E-06	3.03E-08
NE	9.99E-05	740	8.51E-05	820	4.28E-06	2.60E-08
ENE	9.25E-05	740	7.88E-05	820	3.89E-06	2.50E-08
E	8.75E-05	890	9.00E-05	870	4.99E-06	3.35E-08
ESE	1.52E-04	890	1.56E-04	870	1.33E-05	1.02E-07
SE	1.92E-04	920	2.09E-04	850	1.89E-05	2.03E-07
SSE	1.12E-04	830	1.29E-04	730	6.75E-06	6.44E-08

* LPZ distance (all sectors) = 9650 m (6 miles)

TABLE 5.1-3 EAB/ LPZ 5-Percent Overall Site Atmospheric Dispersion Factors					
	X/Q (sec/m^3)				
	0 - 2 hrs	**2 - 8 hrs**	**8 - 24 hrs**	**1 - 4 days**	**4 - 30 days**
Unit 1 EAB	1.89E-04	-	-	-	-
Unit 2 EAB	1.88E-04	-	-	-	-
Unit 1/Unit 2 LPZ	1.46E-05	7.20E-06	5.06E-06	2.35E-06	7.81E-07

TABLE 5.2-1[1]
DCPP On-Site Atmospheric Dispersion Factor Evaluation
Post-Accident Release Point / Receptor Combinations

Release Points	On-Site Receptors
1. Unit 1 Containment Building Edge	1. Unit 1 Control Room Normal Intake
2. Unit 2 Containment Building Edge	2. Unit 2 Control Room Normal Intake
3. Unit 1 Plant Vent	3. Unit 1 Control Room Emergency Intake
4. Unit 2 Plant Vent	4. Unit 2 Control Room Emergency Intake
5. Unit 1 Refueling Water Storage Tank (RWST) Vent[2]	5. Control Room Center (i.e., In-leakage)
6. Unit 2 RWST Vent[2]	6. TSC Normal Intake[3]
7. Unit 1 Containment Penetration (GE Area)	7. TSC Center (i.e., In-leakage)[3]
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[4]	
18. Unit 1 10% Atmospheric Dump Valves [4]	
19. Unit 1 Main Steam Line Break Location	
20. Unit 2 Main Steam Line Break Location	

Notes:

1. See Appendix A, Table A-1 and Figure A-1 for Release Point / Receptor Locations, including all input data used in the ARCON96 calculations.
2. χ/Q s 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.
3. The Unit 1 & 2 MSL break locations are also used to represent the Unit 1 and Unit 2 MSSVs and 10% ADVs. This approach is acceptable since these release points are essentially co-located, but the MSLB location releases have the lowest elevation and is therefore the closest to the TSC.

TABLE 5.2-2[5]
DCPP Unit 1 Control Room Atmospheric Dispersion Factors (sec/m^3)

Source and Receptor		0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
Unit 1 Containment Edge to Unit 1 Control Room (CR) Normal Intake		1.44E-03	7.03E-04	3.00E-04	3.06E-04	3.04E-04
Unit 1 Containment Edge to Unit 2 CR Normal Intake		6.41E-04	3.51E-04	1.49E-04	1.49E-04	1.36E-04
Unit 1 Containment Edge to Unit 1 CR Emergency Intake[4]		4.09E-04	2.31E-04	9.54E-05	8.61E-05	7.04E-05
Unit 1 Containment Edge to Unit 2 CR Emergency Intake[4]		1.57E-04	7.82E-05	2.57E-05	2.71E-05	2.32E-05
Unit 1 Containment Edge to CR Center		9.21E-04	4.38E-04	1.77E-04	1.80E-04	1.67E-04
Unit 1 Plant Vent to Unit 1 CR Normal Intake		1.67E-03	1.22E-03	4.93E-04	4.89E-04	4.36E-04
Unit 1 Plant Vent to Unit 2 CR Normal Intake		9.08E-04	6.53E-04	2.69E-04	2.62E-04	2.38E-04
Unit 1 Plant Vent to Unit 1 CR Emergency Intake[4]		5.56E-04	3.33E-04	1.29E-04	1.11E-04	8.34E-05
Unit 1 Plant Vent to Unit 2 CR Emergency Intake[4]		2.22E-04	1.47E-04	5.44E-05	5.52E-05	4.45E-05
Unit 1 Plant Vent to CR Center		1.25E-03	9.08E-04	3.61E-04	3.65E-04	3.17E-04
Unit 1 Containment Penetration (GE Area) to Unit 1 CR Normal Intake		6.59E-03	2.81E-03	1.16E-03	1.07E-03	8.31E-04
Unit 1 Containment Penetration (GE Area) to Unit 2 CR Normal Intake		2.07E-03	1.13E-03	3.73E-04	3.78E-04	3.05E-04
Unit 1 Containment Penetration (GE Area) to Unit 1 CR Emergency Intake[4]		3.67E-04	2.31E-04	9.02E-05	8.38E-05	6.42E-05
Unit 1 Containment Penetration (GE Area) to Unit 2 CR Emergency Intake[4]		2.39E-04	1.20E-04	4.27E-05	4.22E-05	3.39E-05
Unit 1 Containment Penetration (GE Area) to CR Center		3.01E-03	1.33E-03	5.43E-04	4.93E-04	4.01E-04
Unit 1 Containment Penetration (GW/FW Area) to Unit 1 CR Normal Intake		4.86E-03	3.43E-03	1.35E-03	1.37E-03	1.25E-03
Unit 1 Containment Penetration (GW/FW Area) to Unit 2 CR Normal Intake		1.35E-03	9.79E-04	3.86E-04	3.84E-04	3.46E-04
Unit 1 Containment Penetration (GW/FW Area) to Unit 1 CR Emergency Intake[4]		8.05E-04	5.32E-04	2.12E-04	1.86E-04	1.40E-04
Unit 1 Containment Penetration (GW/FW Area) to Unit 2 CR Emergency Intake[4]		2.40E-04	1.53E-04	4.83E-05	5.20E-05	4.41E-05
Unit 1 Containment Penetration (GW/FW Area) to CR Center		2.55E-03	1.80E-03	7.17E-04	7.13E-04	6.50E-04
Unit 1 RWST Vent to Unit 1 CR Emergency Intake[4]		3.24E-04	1.83E-04	6.94E-05	6.82E-05	5.57E-05
Unit 1 RWST Vent to Unit 2 CR Emergency Intake[4]		1.88E-04	9.18E-05	3.40E-05	3.28E-05	2.69E-05
Unit 1 RWST Vent to CR Center		1.01E-03	4.26E-04	1.85E-04	1.62E-04	1.31E-04

TABLE 5.2-2[5] (Continued)
DCPP Unit 1 Control Room Atmospheric Dispersion Factors (sec/m^3)

Source 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]		4.05E-03	2.65E-03	1.02E-03	1.02E-03	8.95E-04
Unit 1 MSSVs to Unit 1 CR Emergency Intake [3,4]		4.52E-04	2.86E-04	1.14E-04	1.01E-04	7.82E-05
Unit 1 MSSVs to Unit 2 CR Emergency Intake[3,4]		2.75E-04	1.49E-04	4.79E-05	5.02E-05	4.14E-05
Unit 1 MSSVs to CR Center[3]		1.23E-02	7.28E-03	2.21E-03	2.43E-03	2.06E-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]		4.06E-03	2.66E-03	1.03E-03	1.02E-03	9.03E-04
Unit 1 10% ADVs to Unit 1 CR Emergency Intake[3,4]		4.52E-04	2.86E-04	1.14E-04	1.01E-04	7.82E-05
Unit 1 10% ADVs to Unit 2 CR Emergency Intake[3,4]		2.75E-04	1.50E-04	4.82E-05	5.03E-05	4.15E-05
Unit 1 10% ADVs to CR Center[3]		1.23E-02	7.34E-03	2.22E-03	2.45E-03	2.08E-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		4.07E-03	2.86E-03	1.11E-03	1.10E-03	9.70E-04
Unit 1 MSL Break Location to Unit 1 CR Emergency Intake[4]		4.30E-04	2.89E-04	1.14E-04	1.00E-04	7.63E-05
Unit 1 MSL Break Location to Unit 2 CR Emergency Intake[4]		2.74E-04	1.54E-04	4.98E-05	5.12E-05	4.20E-05
Unit 1 MSL Break Location to CR Center		1.14E-02	7.05E-03	2.19E-03	2.37E-03	1.98E-03
Unit 1 FHB to Unit 1 CR Normal Intake		6.68E-03	-	-	-	-
Unit 1 FHB to Unit 2 CR Normal Intake		2.69E-03	-	-	-	-
Unit 1 FHB to Unit 1 CR Emergency Intake[4]		3.28E-04	-	-	-	-
Unit 1 FHB to Unit 2 CR Emergency Intake[4]		2.39E-04	-	-	-	-
Unit 1 FHB to CR Center		3.54E-03	-	-	-	-
Unit 1 Equipment Hatch to Unit 1 CR Normal Intake		2.43E-02	-	-	-	-
Unit 1 Equipment Hatch to Unit 2 CR Normal Intake		2.67E-03	-	-	-	-
Unit 1 Equipment Hatch to Unit 1 CR Emergency Intake[4]		4.32E-04	-	-	-	-
Unit 1 Equipment Hatch to Unit 2 CR Emergency Intake[4]		2.45E-04	-	-	-	-
Unit 1 Equipment Hatch to CR Center		5.06E-03	-	-	-	-

Notes:

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 CR 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 normal operation CR intake of the affected unit.
3. For releases from the MSSVs and 10% ADVs (which are uncapped / vertically oriented and 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 until t=10.73 hrs. Since χ/Q values are averaged over the identified period (i.e., 0-2 hrs, 2-8 hrs, 8-24 hrs, etc), and the vertical velocity has been estimated only up to 10.73 hrs, application of the factor of 5 reduction is not appropriate for χ/Q values applicable to averaging periods beyond the 2-8 hrs averaging period. For assessment of an environmental release between T= 8 to 10.73 hrs, continued use of the 2-8 hr χ/Q (with the factor of 5 reduction) is acceptable and conservative.
4. The more favorable χ/Q value presented above for the CR 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 5.2-3[5]
DCPP Unit 2 Control Room Atmospheric Dispersion Factors (sec/m^3)

Source 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		1.99E-03	9.59E-04	4.60E-04	4.04E-04	3.20E-04
Unit 2 Containment Edge to Unit 1 CR Normal Intake		6.89E-04	3.85E-04	1.66E-04	1.41E-04	1.08E-04
Unit 2 Containment Edge to Unit 1 CR Emergency Intake[4]		1.66E-04	1.05E-04	4.19E-05	3.73E-05	2.93E-05
Unit 2 Containment Edge to Unit 2 CR Emergency Intake[4]		3.78E-04	1.47E-04	5.99E-05	5.87E-05	4.90E-05
Unit 2 Containment Edge to CR Center		1.09E-03	5.49E-04	2.47E-04	2.12E-04	1.70E-04
Unit 2 Plant Vent to Unit 2 CR Normal Intake		1.49E-03	9.29E-04	3.80E-04	3.16E-04	2.21E-04
Unit 2 Plant Vent to Unit 1 CR Normal Intake		7.79E-04	4.80E-04	1.98E-04	1.65E-04	1.15E-04
Unit 2 Plant Vent to Unit 1 CR Emergency Intake[4]		2.02E-04	1.27E-04	5.11E-05	4.20E-05	3.15E-05
Unit 2 Plant Vent to Unit 2 CR Emergency Intake[4]		5.61E-04	2.91E-04	1.16E-04	1.02E-04	8.03E-05
Unit 2 Plant Vent to CR Center		1.11E-03	6.96E-04	2.82E-04	2.35E-04	1.66E-04
Unit 2 Containment Penetration (GE Area) to Unit 2 CR Normal Intake		6.60E-03	3.01E-03	1.17E-03	1.20E-03	1.01E-03
Unit 2 Containment Penetration (GE Area) to Unit 1 CR Normal Intake		2.08E-03	1.38E-03	5.62E-04	4.76E-04	3.59E-04
Unit 2 Containment Penetration (GE Area) to Unit 1 CR Emergency Intake[4]		2.26E-04	1.57E-04	6.15E-05	5.47E-05	4.08E-05
Unit 2 Containment Penetration (GE Area) to Unit 2 CR Emergency Intake[4]		3.74E-04	1.67E-04	6.72E-05	6.14E-05	5.08E-05
Unit 2 Containment Penetration (GE Area) to CR Center		3.09E-03	1.83E-03	7.22E-04	6.74E-04	5.35E-04
Unit 2 Containment Penetration (GW/FW Area) to Unit 2 CR Normal Intake		3.45E-03	1.14E-03	4.70E-04	4.42E-04	2.93E-04
Unit 2 Containment Penetration (GW/FW Area) to Unit 1 CR Normal Intake		1.20E-03	6.21E-04	2.49E-04	2.09E-04	1.41E-04
Unit 2 Containment Penetration (GW/FW Area) to Unit 1 CR Emergency Intake[4]		2.26E-04	1.59E-04	6.50E-05	5.36E-05	3.96E-05
Unit 2 Containment Penetration (GW/FW Area)to Unit 2 CR Emergency Intake[4]		8.08E-04	4.07E-04	1.43E-04	1.42E-04	1.14E-04
Unit 2 Containment Penetration (GW/FW Area)to CR Center		2.19E-03	1.16E-03	4.56E-04	3.83E-04	2.58E-04
Unit 2 RWST Vent to Unit 1 CR Emergency Intake[4]		1.90E-04	1.29E-04	5.00E-05	4.57E-05	3.49E-05
Unit 2 RWST Vent to Unit 2 CR Emergency Intake[4]		3.17E-04	1.40E-04	5.64E-05	5.12E-05	4.16E-05
Unit 2 RWST Vent to CR Center		1.05E-03	5.55E-04	2.12E-04	2.12E-04	1.72E-04

TABLE 5.2-3[5] (Continued)
DCPP Unit 2 Control Room Atmospheric Dispersion Factors (sec/m^3)

Source 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]		3.80E-03	2.36E-03	9.80E-04	8.00E-04	5.99E-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]		2.79E-04	1.87E-04	7.33E-05	6.50E-05	4.89E-05
Unit 2 MSSVs to Unit 2 CR Emergency Intake[3,4]		4.39E-04	2.14E-04	7.68E-05	7.54E-05	6.09E-05
Unit 2 MSSVs to CR Center[3]		1.19E-02	7.90E-03	3.22E-03	2.68E-03	2.05E-03
Unit 2 10% ADVs to Unit 1 CR Normal Intake[3]		3.82E-03	2.36E-03	9.86E-04	8.01E-04	6.01E-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]		2.77E-04	1.88E-04	7.35E-05	6.49E-05	4.89E-05
Unit 2 10% ADVs to Unit 2 CR Emergency Intake[3,4]		4.39E-04	2.14E-04	7.68E-05	7.54E-05	6.09E-05
Unit 2 10% ADVs to CR Center[3]		1.19E-02	7.94E-03	3.23E-03	2.70E-03	2.05E-03
Unit 2 MSL Break Location to Unit 1 CR Normal Intake		3.75E-03	2.37E-03	1.00E-03	7.93E-04	5.81E-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]		2.72E-04	1.88E-04	7.40E-05	6.42E-05	4.80E-05
Unit 2 MSL Break Location to Unit 2 CR Emergency Intake[4]		4.29E-04	2.19E-04	7.73E-05	7.57E-05	6.11E-05
Unit 2 MSL Break Location to CR Center		1.08E-02	7.22E-03	3.00E-03	2.44E-03	1.83E-03
Unit 2 FHB to Unit 1 CR Normal Intake		2.68E-03	-	-	-	-
Unit 2 FHB to Unit 2 CR Normal Intake		6.68E-03	-	-	-	-
Unit 2 FHB to Unit 1 CR Emergency Intake[4]		2.45E-04	-	-	-	-
Unit 2 FHB to Unit 2 CR Emergency Intake[4]		3.23E-04	-	-	-	-
Unit 2 FHB to CR Center		3.61E-03	-	-	-	-
Unit 2 Equipment Hatch to Unit 1 CR Normal Intake		2.47E-03	-	-	-	-
Unit 2 Equipment Hatch to Unit 2 CR Normal Intake		2.48E-02	-	-	-	-
Unit 2 Equipment Hatch to Unit 1 CR Emergency Intake[4]		2.46E-04	-	-	-	-
Unit 2 Equipment Hatch to Unit 2 CR Emergency Intake[4]		4.26E-04	-	-	-	-
Unit 2 Equipment Hatch to CR Center		5.09E-03	-	-	-	-

Notes:

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 CR 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 normal operation CR intake of the affected unit.

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

4 The more favorable χ/Q value presented above for the CR 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 5.2-4
DCPP Units 1 and 2 Technical Support Center Atmospheric Dispersion Factors (sec/m³)

Source and Receptor		0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
UNIT 1						
Unit 1 Containment Edge to TSC Normal Intake		2.45E-04	1.16E-04	4.08E-05	4.17E-05	3.48E-05
Unit 1 Containment Edge to TSC Center		2.74E-04	1.31E-04	4.80E-05	4.70E-05	4.00E-05
Unit 1 Plant Vent to TSC Normal Intake		3.04E-04	1.76E-04	6.82E-05	6.21E-05	5.20E-05
Unit 1 Plant Vent to TSC Center		3.41E-04	1.94E-04	7.63E-05	6.61E-05	5.62E-05
Unit 1 RWST Vent to TSC Normal Intake		2.48E-04	1.15E-04	4.52E-05	4.11E-05	3.40E-05
Unit 1 RWST Vent to TSC Center		2.76E-04	1.23E-04	5.00E-05	4.53E-05	3.65E-05
Unit 1 Containment Penetration (GE Area) to TSC Normal Intake		3.51E-04	1.61E-04	6.43E-05	5.89E-05	4.83E-05
Unit 1 Containment Penetration (GE Area) to TSC.Center		4.05E-04	1.80E-04	7.26E-05	6.60E-05	5.37E-05
Unit 1 Containment Penetration (GW/FW Area) to TSC Normal Intake		4.44E-04	2.48E-04	8.04E-05	8.31E-05	6.68E-05
Unit 1 Containment Penetration (GW/FW Area) to TSC Center		5.61E-04	2.93E-04	1.00E-04	9.86E-05	8.16E-05
Unit 1 MSL Break Location to TSC Normal Intake[1,2]		5.05E-04	2.34E-04	8.95E-05	8.50E-05	6.94E-05
Unit 1 MSL Break Location to TSC Center[1,2]		6.03E-04	2.70E-04	1.07E-04	1.00E-04	8.16E-05
Unit 1 FHB to TSC Normal Intake		3.77E-04	1.68E-04	6.74E-05	6.09E-05	5.06E-05
Unit 1 FHB to TSC Center		4.21E-04	1.87E-04	7.84E-05	6.91E-05	5.57E-05
Unit 1 EH to TSC Normal Intake		4.19E-04	1.93E-04	7.41E-05	7.03E-05	5.76E-05
Unit 1 EH to TSC Center		4.93E-04	2.16E-04	8.73E-05	8.03E-05	6.55E-05
UNIT 2						
Unit 2 Containment Edge to TSC Normal Intake		5.31E-04	1.97E-04	8.36E-05	8.25E-05	6.72E-05
Unit 2 Containment Edge to TSC Center		5.39E-04	2.01E-04	8.73E-05	8.78E-05	6.84E-05
Unit 2 Plant Vent to TSC Normal Intake		5.47E-04	2.27E-04	1.03E-04	8.46E-05	6.68E-05
Unit 2 Plant Vent to TSC Center		5.41E-04	2.09E-04	9.67E-05	7.95E-05	6.43E-05

TABLE 5.2-4
DCPP Units 1 and 2 Technical Support Center Atmospheric Dispersion Factors (sec/m^3)

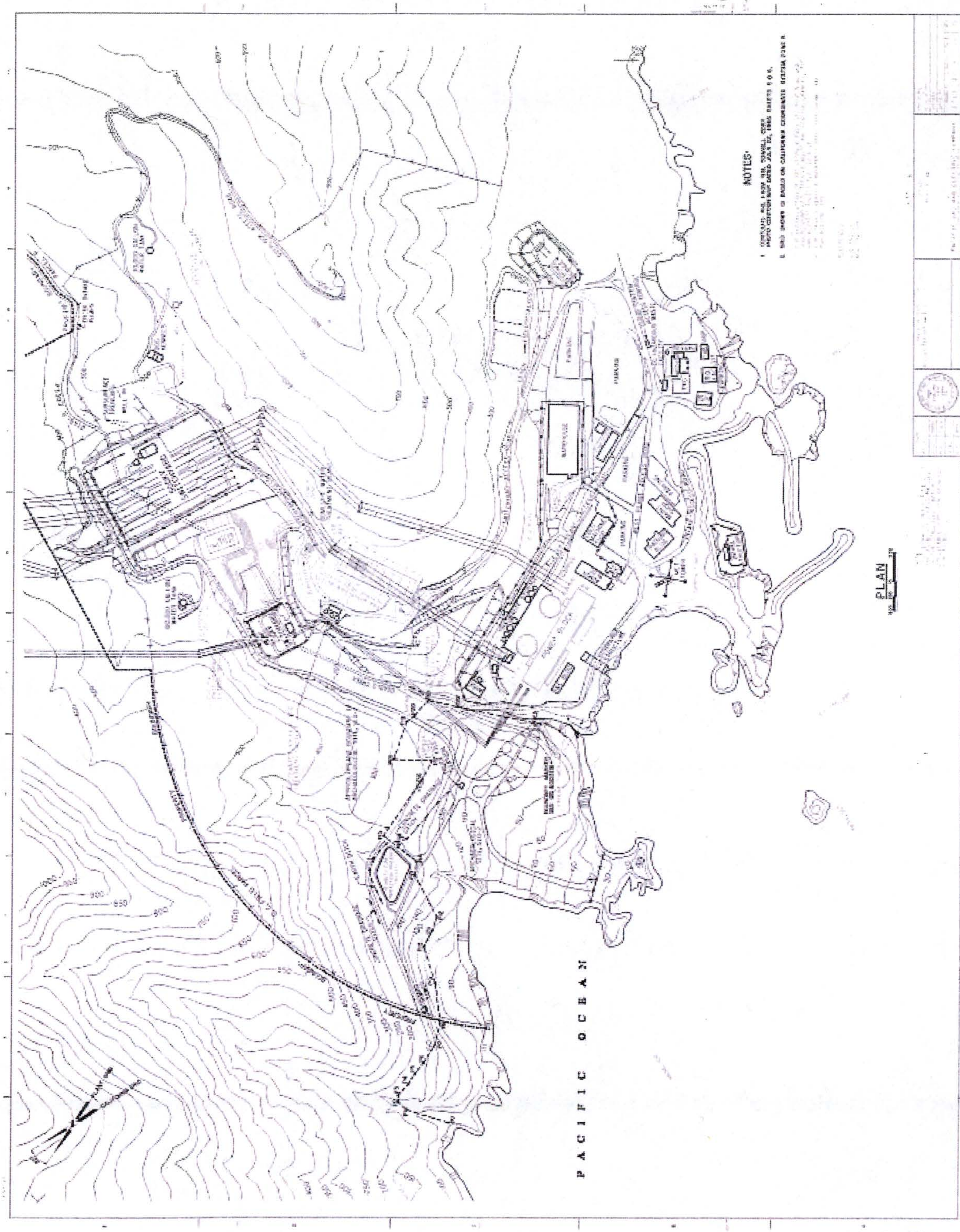
Source and Receptor		0-2 Hour	2-8 Hour	8-24 Hour	1-4 Day	4-30 Day
Unit 2 RWST Vent to TSC Normal Intake		3.52E-04	1.46E-04	6.12E-05	5.66E-05	4.63E-05
Unit 2 RWST Vent to TSC Center		3.61E-04	1.48E-04	6.30E-05	5.80E-05	4.69E-05
Unit 2 Containment Penetration (GE Area) to TSC Normal Intake		5.22E-04	2.21E-04	9.14E-05	8.61E-05	6.71E-05
Unit 2 Containment Penetration (GE Area) to TSC Center		5.49E-04	2.24E-04	9.60E-05	8.85E-05	7.05E-05
Unit 2 Containment Penetration (GW/FW Area) TSC Normal Intake		1.71E-03	7.07E-04	2.98E-04	2.76E-04	2.21E-04
Unit 2 Containment Penetration (GW/FW Area) to TSC Center		1.76E-03	7.16E-04	3.01E-04	2.84E-04	2.28E-04
Unit 2 MSL Break Location to TSC Normal Intake[1,2]		9.00E-04	4.17E-04	1.83E-04	1.52E-04	1.22E-04
Unit 2 MSL Break Location to TSC Center[1,2]		1.01E-03	4.62E-04	1.93E-04	1.71E-04	1.38E-04
Unit 2 FHB to TSC Normal Intake		4.88E-04	2.10E-04	8.65E-05	8.05E-05	6.24E-05
Unit 2 FHB to TSC Center		5.26E-04	2.19E-04	9.19E-05	8.55E-05	6.83E-05
Unit 2 EH to TSC Normal Intake		6.97E-04	2.92E-04	1.23E-04	1.14E-04	8.75E-05
Unit 2 EH to TSC Center		7.44E-04	3.03E-04	1.28E-04	1.19E-04	9.42E-05

Notes:

1. The MSL Break location release X/Q values are used to conservatively represent releases from either the MSSVs, the 10% ADVs or the MSL break location since these release points are essentially co-located, but the MSL Break location releases have the lowest elevation and is therefore the closest to the TSC.

2. When these χ/Q values are used for the MSSVs and 10% ADVs (which are uncapped / vertically oriented and 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 until t=10.73 hrs. Since χ/Q values are averaged over the identified period (i.e., 0-2 hrs, 2-8 hrs, 8-24 hrs, etc), and the vertical velocity has been estimated only up to 10.73 hrs, application of the factor of 5 reduction is not appropriate for χ/Q values applicable to averaging periods beyond the 2-8 hrs averaging period. For assessment of an environmental release between T= 8 to 10.73 hrs, continued use of the 2-8 hr χ/Q (with the factor of 5 reduction) is acceptable and conservative.

Figure 5.1-1
Diablo Canyon Power Plant – Site Drawing



6.0 DOSE CALCULATION METHODOLOGY

6.1 Inhalation and Submersion Doses from Airborne Radioactivity

Computer Code RADTRAD 3.03 is used to calculate the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE) from submersion due to airborne radioactivity at offsite locations and in the control room. The summation of CEDE and EDE is reported as the TEDE. As allowed in Section 4.1.4 of RG 1.183, since the submersion exposure is uniform to the whole body, the EDE is used in lieu of the deep dose equivalent (DDE) in determining the contribution of the submersion dose to the TEDE.

The CEDE is calculated using the inhalation dose conversion factors provided in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Reference 19). The factors in the column headed "effective" yield doses corresponding to the CEDE and are derived based on ICRP-30.

The submersion EDE is calculated using the air submersion dose coefficients provided in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Reference 20). The dose coefficients in the column headed "effective" yield doses corresponding to the EDE and are derived based on a semi-infinite cloud model. The submersion EDE is reported as the whole body dose in the RADTRAD 3.03 output.

RADTRAD 3.03 includes models for a variety of processes that can attenuate and/or transport radionuclides. It can model the effect of sprays and natural deposition that reduce the quantity of radionuclides suspended in the containment or other compartments. In addition, it can model the flow of radionuclides between compartments within a building, from buildings into the environment, and from the environment into a Control Room (CR). These flows can be through filters, piping, or simply due to air leakage. RADTRAD 3.03 can also model radioactive decay and in-growth of daughters. Ultimately the program calculates the whole body dose, the thyroid dose, and the TEDE dose (rem) to the public located offsite, and to onsite personnel located in the CR due to inhalation and submersion in airborne radioactivity based on user specified, fuel inventory, nuclear data, dispersion coefficients, and dose conversion factors. Note that the code uses a numerical solution approach to solve coupled ordinary differential equations. The basic equation for radionuclide transport and removal is the same for all compartments. The program breaks its processing into 2 parts a) radioactive transport and b) radioactive decay and daughter in-growth.

CB&I S&W Inc. computer program PERC2 is used to calculate the CEDE from inhalation and the EDE from submersion due to airborne radioactivity in the Technical Support Center (TSC). PERC2 is a multiple compartment activity transport code with the dose model consistent with Regulatory Guide 1.183 guidance. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included. The CEDE is calculated using the Federal Guidance Report No.11 dose conversion factors. The EDE in the TSC is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the TSC. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes the integrated activity in the TSC

air space, the photon energy release rates per energy group from activity airborne in the TSC, and the ANSI/ANS 6.1.1-1991 neutron and gamma-ray fluence-to-dose factors. (Reference 62)

Offsite Dose – In accordance with RG 1.183, for the first 8 hours, the breathing rate of the public located offsite is assumed to be 3.5×10^{-4} m^3/sec. From 8 to 24 hours following the accident, the breathing rate is assumed to be 1.8×10^{-4} m^3/sec. After that and until the end of the accident, the rate is assumed to be 2.3×10^{-4} m^3/sec. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is calculated and used in determining compliance with the dose criteria in 10CFR50.67. The LPZ TEDE is determined for the most limiting receptor at the outer boundary of the low population zone and is calculated for the entire accident duration.

Control Room Dose - The control room inhalation CEDE is calculated assuming a breathing rate of 3.5×10^{-4} m^3/sec for the duration of the event. The following occupancy factors are credited in determining the control room TEDE: 1.0 during the first 24 hours after the event, 0.6 between 1 and 4 days, and 0.4 from 4 days to 30 days. The submersion EDE is corrected for the difference in the finite cloud geometry in the control room and the semi-infinite cloud model used in calculating the dose coefficients. The following expression obtained from RG 1.183 is used in RADTRAD 3.03 to correct the semi-infinite cloud dose, EDE_∞, to a finite cloud dose, EDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room.

$$EDE_{finite} = \frac{EDE_\infty \, V^{0.338}}{1173}$$

Technical Support Center Dose - The TSC inhalation CEDE is calculated by CB&I S&W Inc. computer code PERC2 assuming the same breathing rate and occupancy factors as those used in determining the control room dose. The submersion EDE developed by PERC2 (which computes the photon fluence at the center of TSC and utilizes the ANSI/ANS 6.1.1-1991 fluence to effective dose conversion factors), is a close approximation of the dose determined using Table III.1 of FGR No. 12, column headed "effective" (see Section 4.1.4, RG 1.183, R0) and adjusted by the finite volume correction factor given in RG 1.183, R0, Section 4.2.7.

6.2 Direct Shine Dose from External and Contained Sources

CB&I S&W Inc. point kernel shielding computer program SW-QADCGGP is used to calculate the deep dose equivalent (DDE) in the control room, TSC and at the EAB due to external and contained sources. The calculated DDE is added to the inhalation (CEDE) and the submersion (EDE) dose due to airborne radioactivity to develop the final TEDE. Conservative build-up factors are used and the geometry models are prepared to ensure that un-accounted streaming/scattering paths were eliminated. The dose albedo method with conservative albedo values is used to estimate the scatter dose in situations where the scattering contributions are potentially significant. ANSI/ANS 6.1.1-1977 "neutron and gamma-ray flux-to-dose-rate factors" (Reference 31) is used to convert the gamma flux to the dose equivalent rate.

7.0 RADIOLOGICAL CONSEQUENCES USING AST

As discussed in Chapter 1, the methodology / scenarios used in the existing design basis accident analyses discussed in the DCPP UFSAR and listed below, are being updated to reflect AST in accordance with the guidance provided in Regulatory Guide 1.183.

1. Loss of Coolant Accident
2. Fuel Handling Accident in the Fuel Handling Building
3. Fuel Handling Accident in the Containment
4. Locked Rotor Accident
5. Control Rod Ejection Accident
6. Main Steam Line Break
7. Steam Generator Tube Rupture
8. Loss-of Load Event

In addition, the updated analyses reflect the results of a "licensing basis verification / design basis re-constitution" effort that was initiated by PG&E to support a total upgrade of the listed radiological post-accident dose consequence analyses. Appendix B provides a comparison of the critical input parameter values utilized in the current licensing basis dose consequence analysis versus that used to support this AST application.

Also included in this application is the use of updated atmospheric dispersion factors for the site boundary (EAB & LPZ), control room and technical support center. (Refer to Chapter 5.0 for detail)

The proposed changes to the current licensing basis that are incorporated in these analyses are summarized in Section 2.1. Section 2.2 provides a summary of the proposed plant modifications. Section 2.3 identifies the key plant operating procedures that will be updated. The full set of impacted procedures will be addressed in the AST ECP.

The assumptions and methodology utilized to estimate the dose consequences at the site boundary and in the control room for the listed design basis accidents are summarized in this Chapter. Parameter values are selected to ensure bounding dose consequences applicable to either unit.

In accordance with the guidance provided in RG 1.183, the assumptions regarding the occurrence and timing of a Loss of Offsite Power (LOOP) during an accident are selected with the intent of maximizing the dose consequences. A LOOP is assumed for events that have the potential to cause grid perturbation. The dose consequences of the LOCA, MSLB, SGTR, LRA, CREA and LOL event are evaluated with the assumption of a LOOP concurrent with reactor trip. The assumption of a LOOP related to a postulated design basis accident which leads to a reactor trip does not directly correlate to an FHA. Specifically, a FHA does not directly cause a reactor trip and a subsequent LOOP due to grid instability; nor can a LOOP be the initiator of an FHA. Thus the FHA dose consequence analyses are evaluated without the assumption of a LOOP.

The worst 2-hour period dose at the EAB, and the dose at the LPZ for the duration of the release, is calculated for each of the above events based on postulated airborne radioactivity

releases. This represents the post-accident dose to the public due to inhalation and submersion for each of these events. Due to distance/plant shielding, the dose contribution at the EAB / LPZ due to direct shine from contained sources is expected to be negligible for all the accidents. However, for purposes of completeness, the direct shine dose at the EAB following a LOCA was evaluated and the almost insignificant dose contribution (< 0.01 rem) included in the dose estimate.

The 30-day integrated dose to an operator in the Control Room due to airborne radioactivity releases is developed for all of the listed design basis accidents. This represents the post-accident dose to the operator due to inhalation and submersion.

The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The direct shine dose due to contained sources / the external radioactive cloud is included in the CR doses reported for the LOCA.

In accordance with current licensing basis, the 30-day integrated dose to an operator in the Technical Support Center (TSC) due to immersion, inhalation and direct shine is evaluated for the DBA that has the worst case radioactivity release, i.e., the LOCA. Also included is an assessment of the dose consequences in the TSC due to airborne radioactivity releases following Non-LOCA events.

7.1 Control Room Design / Operation / Transport Model

The DCPP main control room (CR) serves both units and is located at El 140' of the Auxiliary Building. The walls facing the Unit 1 and Unit 2 containments (i.e., the north and south walls) are made of 3'-0" concrete, whereas the CR east and west walls are made up of 2'-0" concrete. The floor and ceiling thickness / material reflect a minimum of 2'-0" and 3'-4" of concrete, respectively. The CR Mechanical Equipment and HVAC room is located adjacent to the CR (east side), at El 154'-6".

The CR has a normal intake per unit (each located on opposite sides the Auxiliary Building; i.e., north and south), and a pressurization flow intake per unit (each located on either side of the Turbine Building, i.e., north and south). The DCPP CR pressurization air intakes have dual ventilation outside air intake design as defined by Regulatory Position C.3.3.2 of RG 1.194. (See Section 5.2 for additional details)

During normal operation (CRVS Mode 1), both CR normal intakes are operational. Redundant PG&E Design Class I radiation monitors located at each CR normal intake (1-RE-25/26, 2-RE-25/26) have the capability of isolating the CR normal intakes on detection of high radiation and switching the CRVS to Mode 4 operation (i.e., CR filtered intake and pressurization). Other signals that initiate CRVS Mode 4 operation include the safety injection signal (SIS) and Containment Isolation Phase A. The SIS does not directly initiate CRVS Mode 4, however, it initiates Containment Isolation Phase A which initiates Mode 4.

CRVS Mode 4 operation utilizes redundant PG&E Design Class I radiation monitors located at each CR pressurization air intake and the provisions of acceptable control logic to automatically select the least contaminated inlet at the beginning of the accident, and manually select the least contaminated inlet during the course of the accident. Thus, during Mode 4 operation the

dose consequence analyses can utilize the χ/Q values for the more favorable pressurization air intake reduced by a factor of 4 to credit the "dual intake" design (See Section 5.2 for additional details).

During normal operations, 2100 (±10%) cfm of unfiltered air is drawn into the control room envelope (i.e., ~170,000 ft^3 of free volume) from the Unit 1 and Unit 2 normal intakes for a total of 4200 (±10%) cfm. In response to a CR radiation monitor or SI signal, the CR switches to CRVS Mode 4 operation, and control logic ensures that the CRVS pressurization fan of the non-accident unit is initiated and air is taken from the less contaminated of the Unit 1 or Unit 2 CR pressurization air intakes. The pressurization flow at either intake ranges between 650 - 900 cfm. The CR pressurization flowrate used in the dose consequence analyses is selected to maximize the estimated dose in the control room. With the exception of 100 cfm which is assumed to be unfiltered due to backdraft damper leakage, the pressurization flow is filtered.

The allowable methyl iodide penetration and filter bypass for the CRVS Mode 4 Charcoal Filter is controlled by DCPP TS 5.5.11 and is <2.5% and <1%, respectively. In accordance with the NRC SER for License Amendment Nos. 163 and 165, PG&E has committed to the test methods of ASTM D3803-1989, and thus in accordance with the guidance provided in GL 99-02 (Reference 41), use is made of a safety factor of 2 in determining the charcoal filter efficiency to be used in safety analyses. Thus the CR charcoal filter efficiency for elemental and organic iodine used in the DCPP safety analyses is 100% - [(2.5% + 1%) x 2] = 93%. The acceptance criteria for the in-place test of the high efficiency particulate air (HEPA) filters in DCPP TS 5.5.11 is a "penetration plus system bypass" < 1.0%. Thus using methodology similar to the charcoal filters, the HEPA filter efficiency for particulates used in the DCPP safety analyses is 100% - [(1%) x 2] = 98%.

During Mode 4 operation, the CR air is also recirculated and a portion of the recirculation flow filtered through the same filtration unit as the pressurization flow. The range of the flow through the filter banks is 1800 – 2200 cfm with the minimum filtered recirculation flow being 1250 cfm.

Unfiltered inleakage into the CR during Mode 1 and Mode 4 is assumed to be 70 cfm (includes 10 cfm for ingress/egress based on the guidance provided in SRP 6.4). Note that the December 2012 Control Room Tracer Gas Test recorded a maximum unfiltered inleakage of 37 cfm (i.e., 32±5 cfm). (Reference 50)

The CRVS Mode 4 parameter values assumed in the dose consequence analyses are summarized below. These values encompass the results of the recent CR tracer gas test.

Mode 4 CR Parameters	Min Flow (cfm)	Max Flow (cfm)
Pressurization Flow	650	900
Backdraft damper Lkg.	100	100
Filtered Intake	550	800
Charcoal Filter Flow	1800	2200
Filtered Recirc Flow	1250	1400
Unfiltered Inleakage	70	70
CR Exhaust Flow	720	970

For purposes of estimating the post-accident dose consequences, the DCPP control room (CR) is modeled as a single region. When in CRVS Mode 4, the Mode 1 intakes are isolated and outside air is a) drawn into the CR through the filtered emergency intakes; b) enters the CR as infiltration, c) enters the CR during operator egress/ingress, and d) enters the CR as unfiltered leakage via the emergency intake back draft dampers. The direction of flow uncertainty on the CRVS ventilation intake flowrates (normal as well as accident), are selected to maximize the dose consequence in the CR.

As discussed in Section 7.0, the dose consequence analyses for the events that can cause grid instability and credit CRVS Mode 4 operation (i.e., the LOCA, MSLB, SGTR and the CREA), assume a Loss of Offsite Power (LOOP) concurrent with reactor trip.

In accordance with current licensing basis the non-accident unit is assumed unaffected by the LOOP. Thus, to address the effect of a LOOP, and taking into consideration the fact that the time of receipt of the signal to switchover from CRVS Mode 1 to Mode 4 is dependent on the time of reactor trip and is therefore accident specific:

- Automatic isolation of the CR normal intake of the "non-accident" unit, is delayed by 12 seconds from receipt of the signal to switch to CRVS Mode 4. This delay takes into account a 2 second SI signal processing time and a 10 second damper closure time.
- Automatic isolation of the CR normal intake of the "accident" unit, and credit for CRVS Mode 4 operation is delayed by 38.2 seconds from receipt of the signal to switch to CRVS Mode 4. This delay takes into account a) 28.2 seconds for the diesel generator to become fully operational including sequencing delays, and b) 10 seconds for the CR ventilation dampers to re-align. The 2 second SI signal processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay. In addition, and as discussed earlier, the CRVS system design ensures that upon receipt of a signal to switch to Mode 4, the CR pressurization fans of the non-accident unit is initiated; thus fan ramp-up is assumed to occur well within the 38.2 seconds delay discussed above, unhampered by a LOOP.

As discussed in Section 7.0, the FHA dose consequence analyses do not address the potential effects of a LOOP.

The dose consequence analyses for the LRA and the LOL event assume that the CR remains in normal operation mode and do not credit CRVS Mode 4 operation.

Table 7.1-1 lists key assumptions / parameters associated with DCPP control room design.

7.2 Loss of Coolant Accident (LOCA)

The accidental rupture of a main coolant pipe is the event assumed to initiate a large break LOCA. Analyses of the response of the reactor system, including the emergency core cooling system (ECCS), to ruptures of various sizes are presented in DCPP UFSAR Sections 15.3.1 and 15.4.1. As demonstrated in these analyses, the ECCS, using emergency power, is designed to keep cladding temperatures well below melting and to limit zirconium-water

reactions to an insignificant level. However, as a result of the increase in cladding temperature and the rapid depressurization of the core, some cladding failure may occur in the hottest regions of the core. Following the cladding failure, some of the core fission products would be released to the primary coolant and subsequently to the inside of the containment building. There are several passive and active fission product removal mechanisms available inside containment. Active mechanisms include radioactive particulate and iodine removal by the containment sprays inclusive of the containment air mixing provided by the containment fan coolers. DCPP UFSAR Section 6.2 describes the design and operation of the containment spray system and the containment fan coolers.

RG 1.183, Appendix A, identifies the large break LOCA as the design basis case of the spectrum of break sizes for evaluating performance of release mitigation systems and the containment, and for facility siting relative to radiological consequences.

AST methodology as provided by RG 1.183 presents a more credible accident scenario than the instantaneous fuel damage scenario depicted in TID-14844 with respect to fission product releases from the core following a LOCA, and the timing and chemical form of such releases.

The core damage sequence of a AST LOCA scenario as defined by RG 1.183 addresses a delayed radioactivity release, i.e., a gap release starting at t=30 secs, followed by fuel melt starting at t = 30 mins and continuing on to t = 1.8 hrs. At DCPP, containment spray in the injection mode is exhausted within approximately an hour after accident initiation, or earlier if full safeguards are available. Thus in order for the containment spray to continue to be effective as a fission product removal mechanism, the sprays have to be made available beyond the injection mode and continue on in the recirculation mode.

7.2.1 Use of Containment Spray in the Recirculation Mode

DCPP is designed and licensed to operate using containment spray in the recirculation mode. In accordance with current licensing basis, and as documented in the NRC SER related to License Amendment No. 139 to Facility Operating License No's DPR-80 and DPR-82 (Reference 54), containment spray is not required per analyses to be actuated during recirculation, but may be actuated in accordance with the EOPs or at the discretion of the Technical Support Center. With this application, TS Bases 3.6.6 and the associated emergency operating procedures will be updated to require initiation of containment spray in the recirculation mode from the control room within 12 minutes of termination of injection spray.

<u>Minimum Core flow rate and Containment Spray flow rate when CS is operating in the recirculation mode</u>.

- The minimum ECCS core flow available when containment spray is operating in the recirculation mode is 713.4 gpm in addition to the spill flow via the break. The above value is greater than the required core flow rate acceptance criteria of 709.6 gpm and is based on a single active failure of Train B to minimize the available pumps for core cooling. Thus the minimum ECCS core flow is based on operation with 1 Residual Heat Removal (RHR) Pump / 1 Safety Injection Pump (SI) Pump / 1 Centrifugal Charging Pump (CCP). Note that the minimum *required* ECCS core flow of 709.6 gpm was determined by Westinghouse as 1.2 times the core boil-off with Replacement Steam

Generators. The boil-off rates are conservatively calculated based on the minimum RWST drain-down time of 24 minutes, and an assumed saturation temperature of 212°F for the ECCS fluid downstream of the RHR heat exchanger. The predicted minimum ECCS core flow rate of 713.4 gpm when operating in the recirculation spray mode is based on the calculated ECCS fluid temperature downstream of the RHR heat exchanger; thus the margin between the minimum ECCS core flow rate when operating in the recirculation spray mode, and the acceptance criteria will increase if it is adjusted to 212°F.

- The minimum containment recirculation spray flow rate is determined to be 1211 gpm. The above value is based on Valves 8809A and B being closed during the recirculation spray mode which is in accordance with current licensing basis. A single active failure of the Train-A RHR pump is assumed to minimize the CS flow and maximize the flow to the core. In summary, the minimum CS flowrate is based on operation with 1 RHR Pump / 2 SI Pumps / 2 CCPs.

7.2.2 Activity Release Pathways following a LOCA

DCPP has identified six activity release paths following a LOCA

1. Release via the Containment Pressure / Vacuum Relief pathway to the environment until the containment isolation valves are closed.

2. Containment leakage to the environment after containment isolation is achieved.

3. Sump water leakage from ESF systems that recirculate sump water outside containment.

4. Failure of the RHR pump seal at T=24 hrs resulting in a 50 gpm leak of sump water for 30 mins.

   Note: DCPP design includes an ESF atmosphere filtration system, so from a regulatory standpoint per SRP 15.6.5, Appendix B (Reference 51), as well as RG 1.183, inclusion of this leakage path in dose consequences is not required. However, the RHR pump seal failure resulting in a "filtered" release is DCPP's licensing basis with respect to passive single failure, and will be maintained for this application. Specifically,

   - *UFSAR Section 3.1.1.1 (Single Failure Criteria / Definitions), Item 2; discusses passive failures – "The structural failure of a static component that limits the component's effectiveness in carrying out its design function. When applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. Such leak rates are assumed for RHR pump seal failure."*

   - *UFSAR Appendix 6.3A.3.2 (discusses passive failures), indicates that – the design of the auxiliary building and related equipment is based on handling of leaks up to a maximum of 50 gpm. Means are provided to detect and isolate such leaks in the emergency core cooling pathway within 30 mins*

- *UFSAR Section 15.5.17.2.8 indicates that - failure of an RHR pump seal at 24 hrs is assumed as the single failure that can be tolerated without loss of the required functioning of the RHR system*

Therefore, the RHR Pump Seal Failure is retained as a release pathway for the AST dose consequence analysis.

5. Releases to the environment from the MEDT which collects component leakage hard-piped to the MEDT. The collected fluid includes both post-LOCA sump water and other non-radioactive fluid.

6. Releases to the environment via the refueling water storage tank (RWST) vent due to post-LOCA sump fluid back-leakage into the RWST via the mini-flow recirculation lines connecting the high head and low head safety injection pump discharge piping to the RWST.

The DCPP LOCA dose consequence analysis follows the guidance provided in the pertinent sections of RG 1.183 including Appendix A. Table 7.2-1 lists the key assumptions / parameters utilized to develop the radiological consequences following a LOCA at either unit.

7.2.3 Dose from Submersion and Inhalation

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOCA.

7.2.3.1 Containment Pressure /Vacuum Relief Line Release

In accordance with RG 1.183, Appendix A, Section 3.8, for containments such as DCPP that are routinely purged during normal operations, the dose consequence analysis must assume that 100% of the radionuclide inventory in the primary coolant is released to the containment at the initiation of the LOCA. The inventory of the release from containment should be based on primary coolant equilibrium activity as allowed by the Technical Specifications (see Table 4.2-1). Iodine spikes need not be considered.

Thus, in accordance with the above guidance, the 12 inch containment vacuum / over pressure relief valves are assumed to be open to the extent allowed by DCPP Technical Specifications (i.e., blocked to prevent opening beyond 50 degrees) at the initiation of the LOCA, and the release via this pathway terminated as part of containment isolation. The analysis assumes that 100% of the radionuclide inventory in the primary coolant, assumed to be at Technical Specification levels, is released to the containment at T= 0 hours. It is conservatively assumed that 40% of release flashes and is instantaneously and homogeneously mixed in the containment atmosphere, and that the activity associated with the volatiles, i.e., 100% of the noble gases and 40% of the iodine in the reactor coolant, is available for release to the environment via this pathway.

Containment pressurization (due to the RCS mass and energy release), combined with the relief line cross-sectional area, results in a 218 acfs release of containment air to the environment for a conservatively estimated period of 13 seconds. Credit is taken for pressure boundary integrity

of the containment pressure / vacuum relief system ductwork which is classified as PG&E Design Class II, and seismically qualified; thus, environmental releases are via the Plant Vent.

Since the release is isolated within 13 seconds after LOCA, i.e., before the onset of the gap phase release, releases associated with fuel damage are not postulated. The chemical form of the iodine released from the RCS to the environment is assumed to be 97% elemental and 3% organic.

7.2.3.2 Containment Leakage

The inventory of fission products in the reactor core available for release into the containment following a LOCA is provided in Table 4.1-1 which represents a conservative equilibrium reactor core inventory of the dose significant isotopes assuming maximum full power operation at 1.05 times the current licensed thermal power, and taking into consideration fuel enrichment and burnup. The notes provided at the bottom of Table 4.1-1 provide information on isotopes used to estimate the inhalation and submersion doses following a DCPP LOCA, vs isotopes that are considered to estimate the post-LOCA direct shine dose.

Per RG 1.183, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Containment spray is utilized as one of the primary means of fission product cleanup following a LOCA. Mixing of the "effectively" sprayed volume of containment, with the unsprayed volume of the containment is enhanced by operation of the PG&E Design Class I containment fan coolers. In order to quantify the effectiveness of the containment spray system, both the volume fraction of containment that is sprayed, and the mixing rate between the sprayed and unsprayed volumes are quantified.

The LOCA dose consequence analysis is based on an assumed worst case single failure of loss of one ESF train. A single train ESF consists of one train of ECCS, one train of CSS, and two Containment Fan Cooling Units (CFCUs). A single train scenario is selected to be consistent with the use of reduced iodine and particulate removal coefficients associated with single train operation.

7.2.3.2.1 Spray Duration

Containment Spray in the injection mode is initiated at 111 seconds after the LOCA and terminated at 3798 seconds. Manual operation is credited to initiate containment recirculation spray. Thus, based on single train operation, containment spray in the recirculation mode is initiated at 4518 seconds (i.e., twelve (12) minutes after injection spray is terminated), and terminated at 22,518 seconds. In summary, containment spray operation (injection plus recirculation) is credited until 6.25 hrs post-LOCA, with a twelve minute gap after injection spray is terminated.

7.2.3.2.2 Effectively Sprayed Volume Fraction of Containment

The current licensing basis 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% covered by sprays. It includes the sprayed volume below the grating in

the operating floor deck, and the refueling cavity volume. The percentage of the total containment free volume that is sprayed is 82.5%.

To justify the assumption that the unoccupied containment volume above the operating floor is 100% covered by sprays, the analysis supporting the current licensing basis 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% for Unit 1 and 44% 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.

In support of this application, an analysis was performed to evaluate 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. The minimum volumetric flow rate of water through the spray nozzles, the associated nozzle water pressure drop, and Figure 1 of NUREG/CR-5966 (Reference 33), 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% increase in the "unsprayed area percentage of the containment deck" for both Units. It concluded that the 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 use of a sprayed volume of 82.5% of the containment free volume, is acceptable, for both the containment spray injection as well as the containment spray recirculation mode.

7.2.3.2.3 Mixing between Sprayed and Unsprayed Regions of Containment

The PG&E Design Class I containment fan cooler units 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 Containment Fan Coolers Units (CFCUs) with a total volumetric flow rate of 68,000 cfm, between the unsprayed regions and sprayed regions. The design flow for each CFCU is 47,000 cfm; the value used to determine the mixing rate addresses surveillance margins and uncertainty.

Review of the layout and arrangement of the intake and exhaust registers of the CFCUs indicate that the air intakes are all located above the operating floor (sprayed region) and the air discharge registers are all located below the operating floor in the unsprayed region. Additional review of the containment configuration including the location of major openings, and the 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. CFCU operation is initiated at 86 seconds after the LOCA and

operates for the duration of the accident. In accordance with RG 1.183, Appendix A, Section 3.3, prior to CFCU initiation, the dose consequence model assumes a mixing rate attributable to natural convection between the sprayed and unsprayed regions of 2 turnovers of the unsprayed region per hour.

7.2.3.2.4 Fission Product Removal

RG 1.183, Appendix A, Section 3.3, invokes SRP 6.5.2 (Reference 42) and NUREG/ CR 5966 (Reference 33) as acceptable models for removal of iodines and particulates.

Regulatory guidance provided in SRP 6.5.2 outlines simplified methodology to develop steady state and conservative iodine and particulate removal coefficients in the containment for post-LOCA fission products. However, since with implementation of AST, the releases from the core are assumed to be predominantly particulate in nature, refinement of the SRP 6.5.2 methodology in determining particulate removal coefficients is deemed appropriate.

RG 1.183, Appendix A, Section 3.3, permits the use of time-dependent particulate aerosol removal coefficients by invoking NUREG/CR 5966, and indicates that no reduction in particulate aerosol removal coefficients is required when a DF of 50 is reached if the removal rates are based on the calculated time-dependent airborne aerosol mass.

Thus the fission product removal coefficients developed for the LOCA reflect the following guidance documents:

- Elemental iodine removal coefficients are calculated using guidance provided in SRP 6.5.2 which is invoked by Regulatory Guide 1.183, Appendix A, Sec 3.3
- Time dependent particulate aerosol removal coefficients are estimated using guidance provided in RG 1.183 Appendix A, Sec 3.3, for alternative source terms, and use of CB&I S&W Inc. 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 7.2-2. The methodology utilized to develop these values is summarized below.

1. 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), 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.

The methodology presented below envelopes both Unit 1 and Unit 2 and addresses the development *of time-dependent particulate aerosol removal coefficients.*

a. Removal of Particulates by Sprays

The particulate removal rate is calculated using CB&I S&W Inc. computer code SWNAUA. Computer code SWNAUA is a derivative of NAUA/MOD 4 (Reference 40). The results of SWNAUA have been accepted by the NRC for the AST applications supporting the design certification of CE System 80+, and for operating nuclear plants Beaver Valley Power Station [ML032530204] and Fort Calhoun Station [ML013030027].

The NAUA/MOD4 code does not include a model for aerosol removal by sprays. The aerosol removal model for sprays was developed and incorporated into the SWNAUA code by CB&I S&W Inc. as a conservative model suitable for design basis accident calculations. The model correlations implemented into SWNAUA conservatively underestimate the spray removal coefficient. The spray model incorporated in the SWNAUA code was originally described in Reference 34. 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 (Reference 35) 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 (sec^{-1})
F_m	=	Spray mass flow rate (gm/sec)
h	=	Spray fall height (cm)
ε	=	Collision efficiency
R_{sp}	=	Spray droplet radius (cm)
ρ_w	=	Density of the spray droplet (gm/cm^3)
V	=	Effectively sprayed volume of containment (cm^3)
v_{spray}	=	Velocity of the spray droplets (cm/sec)
v_{sed}	=	Aerosol sedimentation velocity (cm/sec)

The collision efficiency is divided into three contributing mechanisms as described in BMI-2104 (Reference 36):

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

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 NUREG-0772 (Reference 35):

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

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 BMI-2104 (Reference 36):

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

Where

r	=	Aerosol particle radius (cm)
ρ_p	=	Aerosol density (gm/cc)
C_c	=	Cunningham slip correction factor,
μ	=	Gas viscosity (gm/(cm-sec))

For droplet sizes typical of nuclear plant spray systems, the data of Walton and Woolcock (Reference 37) show that collision efficiency will be closer to that predicted for potential flow around the droplet. Calvert (Reference 38) fitted this data to the expression:

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

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:

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

which is a conservative approximation of the expression given by BMI-2104 (Reference 36). The efficiency due to Brownian motion is also taken from this report:

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

Where

Pe	=	Peclet number
	=	$2v_{spray}R_{sp}/D_B$,
D_B	=	Aerosol diffusion coefficient (cm^2/sec)
	=	$k_{Boltz}TB$ (Fuchs, Reference 39, p. 181),
k_{Boltz}	=	Boltzmann constant
	=	1.3804×10^{-16} (erg/°K).
T	=	Temperature (°K)

Fuchs (Reference 39, p. 27) 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.

The model has the capability of handling a distribution of up to 20 droplet radii with the spray removal efficiency being determined for each aerosol size bin.

Elia and Lischer (Reference 34) investigated the use of a single spray droplet size in the analysis instead of a drop size distribution. While Reference 34 does not specifically analyze the DCPP spray system, the parameter sensitivities for the spray model are applicable. The paper demonstrates that the droplet diameter distribution can be represented by a single diameter that is the mass mean diameter. The case 6 droplet distribution presented in the paper is for the SPRACO 1713A nozzle that is frequently used by the nuclear industry for fission product/heat removal spray systems. This diameter approximates that used in case 1, 1000μ. The spray flow rate used for both case 1 and case 6 is 10,000 gpm. Table 2 in the paper indicates that the spray removal rates for these two cases are very close. The mass mean spray droplet radii for DCPP are specified in the table below.

The paper also investigated the variation of particulate removal coefficient with droplet diameter. Cases 1 through 3 vary the mass mean droplet diameter from 500μ to 1500μ. Although Table 2 in the paper indicates that these cases assume a spray flow of 10,000 gpm, the spray removal coefficient reduction by about 67 percent is expected to be independent of spray flow rate.

The bounding plant parameters for the DCPP Units are listed below.

<u>Plant Parameters for Fission Product Cleanup Calculations</u>

Parameter	Value
Sprayed Containment Volume	5.960×10^{10} cm^3
Fall Height	3,536 cm
Spray Flow Rate	2,456 gpm (111 - 3,798 sec) 0 gpm (3,798 - 4,518 sec) 1,211 gpm (4,518 - 22,518 sec)
Spray droplet radius	500×10^{-4} cm : (111- 3,798 sec)* 500×10^{-4} cm : (4,518 - 22,518 sec)

* Spray droplet radius during the injection phase conservatively assigned the larger droplet radius applicable to the recirculation phase.

The DCPP spray coverage fraction of 82.5% is utilized for the duration of injection and recirculation spray.

The DCPP containment pressure, temperature, and relative humidity transient data following the limiting DBA are presented in Table 7.2-2A.

Description of Aerosol

The chemical composition of the aerosol is only important as it relates to the density of aerosol utilized in the development of spray removal rates. The chemical composition during the gap release phase is assumed to be pre-dominantly CsOH. The chemical composition during the in-vessel release phase is assumed to be 20 percent CsOH, 20 percent indium, and 60 percent silver. These assumed compositions are based on a review of the SASCHA experimental results. The aerosol input data for SWNAUA are provided below.

Description of Aerosol	
Minimum Aerosol Radius	1.0000E-07 cm
Maximum Aerosol Radius	1.0000E-02 cm
Maximum Number of Aerosol Size Bins	100
From 30 sec to 1830.0 sec	
Aerosol Injection Rate	10.29 (gm/sec)
Mean Geometric Radius	7.50000E-06 cm
Geometric Standard Deviation	1.56
Aerosol Density	3.7 gm/cc
From 1830 sec to 6510.0 sec	
Aerosol Injection Rate	88.74 (gm/sec)
Mean Geometric Radius	4.00000E-05 cm
Geometric Standard Deviation	1.46
Aerosol Density	4.6 gm/cc

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 diffusiophoresis model in the SWNAUA computer code is the same as that in the NAUA/MOD4 computer code.

The containment steam condensation rates used by SWNAUA are presented in Table 7.2.2B.

The coefficients for removal of particulates from the effectively sprayed and unsprayed regions of the containment are plotted versus time in Figures 7.2-1 and 7.2-2, respectively. For the effectively sprayed region, the aerosol removal is due to sprays and diffusiophoresis. The particulate removal coefficient in the unsprayed region is due to gravitational settling only.

2. *Elemental Iodine Removal*

The methodology presented in Section III, 4.C.i, of SRP 6.5.2 (Reference 42) 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 ($\lambda_{E, Spray}$)

a. Elemental Iodine Removal Coefficients Due to Wall Deposition ($\lambda_{E, Wall}$)

The elemental iodine removal coefficients due to wall deposition can be estimated using the equation provided in Reference 42.

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

Where:

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

Note: A value of 4.9 m/hr (16.08 ft./hr) for K_w conservatively envelopes all available experimental data (Reference 42).

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.

b. 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 Reference 42.

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

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 (Reference 42).

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 on pages 418 and 441 of Reference 43.

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

where:

D_g	= diffusivity of iodine in the gas film surrounding the drop (ft^2 /hr)
Re	= dimensionless Reynolds number = $U \cdot \rho \cdot d/\mu$
Sc	= dimensionless Schmidt number = $\mu/(\rho \cdot D_g)$

7.2.3.2.5 Sump Water pH

SRP Sections 6.1.1 (Reference 45) and 6.5.2 (Reference 42) 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 have 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. Per SRP 6.5.2, II.1.g, long term iodine retention may be assumed only when the equilibrium sump water pH after mixing and dilution with the primary coolant is above 7. Per RG 1.183, long-term production of acids (hydrochloric acid (HCl) and nitric acid (HNO_3)) by irradiation needs to be addressed in determining whether the plant chemical addition system is adequate for long-term pH control.

NUREG/CR-5732 (Reference 44), 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 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. It is therefore concluded that 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 ~ 1 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.

As part of the AST application, a conservative analysis is performed to confirm that the sump water pH at thirty (30) days following a LOCA remains greater than 7.0. The analysis assumes 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, i.e., the reactor coolant and the RWST. To establish the cable inventory inside containment, a simplified conservative upper bound approach is utilized by taking into consideration a) cable insulation data provided in NUREG/CR-5950 (Reference 46), specifically, the amounts of EPR/Hypalon cable from PWRs listed in Table 2.2 of Reference 46; and b) by examining the mass/type of cable installed in CB&I S&W constructed PWRs. A safety factor of 1.5 is applied to the largest mass of electrical cable identified (was determined to be for a 4-loop PWR with a power level slightly greater than DCPP), to estimate an upper bound value for the electrical cable installed inside the DCPP containments. Although the mass of electrical cable identified was applicable to both insulation and conductors, the DCPP analysis conservatively assumed that it was all insulation. The airborne LOCA radiation dose was conservatively assumed to be 2E+08 Rads which is commonly used for evaluating environmental qualification of electrical equipment in PWR containments, and was recommended post-TMI in IEB 79-01B (Reference 47) as a representative / upper bound value for the beta dose inside containment for PWRs, while the gamma dose estimate was a decade lower.

Based on the above approach, the DCPP minimum ultimate sump pH was conservatively determined to be ~ 7.8 without long-term acid production due to radiolysis, and >7.5 at T=30 days following a LOCA, inclusive of long-term acid production inside containment. Due to the availability of significant margin even with the use of conservative methodology, it is concluded that determination of the actual mass of cable insulation inside the DCPP containments is not required, and that the minimum sump pH at DCPP will be ≥ 7.5 inclusive of acid production. Iodine volatility is analyzed for a maximum sump pH of 7.0.

Based on the above assessment it is concluded that the 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.

7.2.3.2.6 Containment Leakage Transport Model

As indicated previously, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. In accordance with RG 1.183, two fuel release phases are considered for DBA analyses: (a) the *gap release*, which begins 30 seconds after

the LOCA and continues to t=30 mins and (b) the *early In-Vessel release* phase which begins 30 minutes into the accident and continues for 1.3 hours (i.e., t=1.8 hrs).

Per RG 1.183, the core inventory release fractions, by radionuclide groups, for the gap and early in-vessel damage are as follows. Per Note 10, of Section 3.2 of RG 1.183, the release fractions listed below are acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62,000 MWD/MTU.

Group	Gap Release Phase	Early In-Vessel Release Phase
Noble gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Group	-	0.05
Ba, Sr	-	0.02
Noble Metals	-	0.0025
Cerium Group	-	0.0005
Lanthanides	-	0.0002

The elements in each radionuclide group released to the containment following a LOCA are assumed to be as follows (note that the groupings were expanded from that in RG 1.183 to address isotopes in the core with similar characteristics; the added isotopes are in bold font):

Noble gases:	Xe, Kr
Halogens:	I, Br
Alkali Metals:	Cs Rb
Tellurium Grp:	Te, Sb, Se, **Sn, In, Ge, Ga, Cd, As, Ag**
Ba,Sr:	Ba, Sr
Noble Metals:	Ru, Rh, Pd, Mo, Tc, Co
Cerium Grp:	Ce, Pu, Np, **Th**
Lanthanides:	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, **Gd, Ho, Tb**

As discussed in Section 7.2.3.2.5, current DCPP design includes chemical addition into the containment spray system which ensures a long term sump pH equal to or greater than 7.0. Thus, the chemical form of the radioidine released from the fuel is assumed to be 95% particulate (Cesium iodide (CsI)), 4.85% elemental iodine, and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, all fission products released are assumed to be in particulate form.

The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LOCA.

Isotopic decay, containment leakage, selected natural removal mechanisms and spray removal are credited to deplete the inventory of fission products airborne in containment. Containment Spray in the injection mode is initiated at 111 seconds after the LOCA and terminated at 3798 seconds. As discussed in Section 7.2.3.2.2, the sprays are estimated to cover 82.5% of the

containment free volume of 2.55E+06 ft^3. Manual operation is credited to initiate containment recirculation sprays. Thus, based on single train operation, containment spray in the recirculation mode is initiated at 4518 seconds (i.e., twelve (12) minutes after injection spray is terminated), and terminated at 22,518 seconds. In summary, containment spray operation (injection plus recirculation) is credited until 6.25 hrs post-LOCA, with a twelve minute gap after injection spray is terminated.

In the effectively sprayed region the activity transport model takes credit for aerosol removal due to steam condensation and via containment spray based on spray flowrates associated with minimum ESF. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by concentration dependent aerosol removal lambdas, and isotopic in-growth due to decay.

During spray operation in the injection mode, the elemental iodine removal rate for the sprays exceeds 20 hr^{-1}, the maximum value permitted by SRP Section 6.5.2; thus the elemental iodine removal rate attributable to sprays is limited to 20 hr^{-1}. During recirculation spray operation, the elemental removal rate for the sprays is 19.34 hr^{-1}. As discussed earlier, the wall deposition removal coefficient for elemental iodine has been calculated with the model provided in SRP Section 6.5.2. In sprayed and unsprayed regions, prior to spray actuation, the wall deposition removal coefficient is estimated to be 2.74 hr^{-1}, while during spray operation, and in the sprayed region only, the wall deposition removal coefficient is estimated to be 0.57 hr^{-1}.

In the unsprayed region, the aerosol removal lambdas reflect gravitational settling. No credit is taken for elemental iodine removal in the unsprayed region.

Since the spray removal coefficients are based on calculated time dependent airborne aerosol mass, there is no restriction on the DF for particulate iodine. The maximum DF for elemental iodine is based on SRP 6.5.2 and is limited to a DF of 200. The maximum allowable DF for elemental iodine is developed using methodology outlined in RG 1.183 Section 3.3.

The methodology used to develop the elemental iodine and particulate removal coefficients in the sprayed and unsprayed region of the containment is discussed in Section 7.2.3.2.4. 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 7.2-2

As discussed in Section 7.2.3.2.5, the long term sump water pH is greater than 7.0. Consequently, iodine re-evolution is not addressed.

As discussed in Section 7.2.3.2.3, mixing between the sprayed and unsprayed regions of the containment is assumed for the duration of the accident. CFCU operation is initiated at 86 seconds after the LOCA and operates for the duration of the accident. The containment mixing rate between the sprayed and unsprayed regions following CFCU initiation is determined to be 9.13 turnovers of the unsprayed regions per hour with a total volumetric flow rate of 68,000 cfm between the unsprayed regions and sprayed regions. In accordance with RG 1.183, Appendix A, Section 3.3, prior to CFCU initiation, the dose consequence model assumes a mixing rate attributable to natural convection between the sprayed and unsprayed regions of 2 turnovers of the unsprayed region per hour.

Radioactivity is assumed to leak from both the sprayed and unsprayed region to the environment at the containment technical specification leak rate for the first day, and half that leak rate for the remaining duration of the accident (i.e., 29 days). To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflects the worst value between the containment wall release point, the plant Vent, the Containment Penetration Area GE (EL 140') and the Containment Penetration Areas GW/FW (EL 140').

7.2.3.3 *ESF System leakage outside Containment*

In accordance with RG 1.183, with the exception of noble gases, all the fission products released from the core during the gap and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment recirculation sump water at the time of release from the fuel. A minimum sump water volume of 480,015 gallons is utilized in this analysis.

In accordance with regulatory guidance, the DCPP ESF systems that recirculate sump fluids outside containment are analyzed to leak at twice the sum of the administratively controlled total allowable leakage applicable to all components in the ESF recirculation systems. With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. In addition, per RG 1.183, if the temperature of the leakage exceeds 212°F, the fraction of the total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. However, if the temperature of the leakage is less than 212°F, or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid unless a smaller amount can be justified based on the actual sump pH history data and area ventilation rates.

ESF leakage is assumed to occur at initiation of the recirculation mode for safety injection, which at DCPP occurs as early as t=829 seconds. The maximum temperature of the recirculation fluid is 259.9°F which has a flash fraction less than 10%, thus, per RG 1.183, ten percent (10%) of the halogens associated with this leakage are assumed to be airborne and are exhausted (without mixing and without holdup) to the environment. The iodine release from the core is 95% particulate (CsI), 4.85% elemental and 0.15% organic, however after interactions with sump water the environmental release is assumed to be 97% elemental and 3% organic.

At DCPP, the environmental release of ESF system leakage can occur via the 2 pathways listed below. As part of this application, DCPP is proposing to establish administrative acceptance criteria to ensure the total as-tested leakage from ESF systems that recirculate sump fluid outside containment is less than or equal to 126 cc/min, and with the following breakdown:

- Environmental release of ESF System leakage via the plant vent: The sum of the maximum allowable simultaneous leakage from all components in the ESF recirculation systems located in the auxiliary building is limited to 120 cc/min. Thus, and in accordance with the guidance provided in RG 1.183, the analysis addresses an ESF leakage of 240 cc/min in the Auxiliary Building (AB). The areas where these components are located are covered by the PG&E Design Class I Auxiliary Building ventilation system which discharges to the environment out of the Plant Vent. Only selected portions of the Auxiliary Building ventilation system are processed through the

PG&E Design Class I AB ventilation filters. For purposes of estimating the dose consequences, it is assumed that with the exception of the RHR pump rooms (refer to Section 7.2.3.4), this release pathway bypasses the PG&E Design Class I AB ventilation filters.

- Environmental release of ESF System leakage via Containment Penetration Area GE and Areas GW and FW: The sum of the maximum allowable simultaneous leakage from all components in the ESF recirculation systems located in the containment penetration areas is limited to 6 cc/min. Thus, and in accordance with the guidance provided in RG 1.183, the analysis addresses an ESF leakage of 12 cc/min in the containment penetration areas. The ventilation system covering this area is not PG&E Design Class I, thus the release path to the environment is unfiltered and could occur via the Plant Vent or via the closest structural opening in the Containment Penetration Areas GE and Areas GW and FW.

7.2.3.4 RHR Pump Seal Failure

As discussed in Section 7.2.2, the RHR pump seal failure resulting in a *filtered release* via the plant vent is DCPP's licensing basis with respect to the worst case passive single failure in the RHR System. Therefore, the RHR pump Seal Failure is retained as a release pathway for the AST LOCA dose consequence analysis.

The activity transport model is based on a 50 gpm leak of sump water activity for 30 minutes that occurs 24 hours after the LOCA. The temperature of the recirculation fluid is conservatively assumed to remain at the maximum temperature of 259.9°F. Thus as discussed above in Section 7.2.3.3 under ESF system leakage, the amount of iodine that becomes airborne is assumed to be 10% of the total iodine activity in the leaked fluid.

The ventilation exhaust from the RHR pump rooms is covered by the PG&E Design Class I Auxiliary Building ventilation system and processed through the PG&E Design Class I AB ventilation filters. Thus, credit for filtration of the release of a RHR pump seal failure by the Auxiliary Building Ventilation system is taken in determining the dose consequences to the public at the EAB and LPZ, to the operator in the control room, and to personnel in the technical support center.

The efficiency of the AB charcoal filters is determined using methodology similar to that documented in Section 7.1 for the CRVS Mode 4 ventilation filters. The allowable methyl iodide penetration / filter bypass for the Auxiliary Building Charcoal Filter is controlled by DCPP TS 5.5.11; currently the associated values are 15% and <1%, respectively. With this application, the allowable methyl iodide penetration for the AB filter will be reduced to 5%. Based on the above, an efficiency of 88% is assigned to the charcoal filters in the AB ventilation system prior to environmental release via the plant vent. Similar to the ESF system leakage, the environmental release of iodine is assumed to be 97% elemental and 3% organic.

7.2.3.5 *Refueling Water Storage Tank (RWST) Back-Leakage*

Dose consequences associated with the potential for post-LOCA radioactive leakage to tanks

vented to the atmosphere was raised in NRC Information Notice (IN) 91-56. (Reference 48)

At DCPP, the safety injection and containment spray systems function to provide reactor core cooling and mitigate the containment pressure and temperature rise, respectively, in the event of a LOCA. Both systems initially take suction from the RWST. Once the RWST water supply is depleted, both the containment spray and safety injection systems are supplied by the RHR System. The RHR pumps take suction from the containment recirculation sump water. Under LOCA conditions, the recirculation sump water is assumed to be radioactively contaminated by fission products, of which the main contributors to airborne dose are the various isotopes of iodine.

Per IN 91-56, during containment sump water recirculation, there is the potential for leakage from the mini-flow recirculation lines connecting the high head and low head safety injection pump discharge piping to the RWST. Since the RWST is vented to the atmosphere, this presents a pathway for iodine release to the atmosphere. As part of this application, DCPP is proposing to establish administrative acceptance criteria to ensure the total as-tested back leakage into the RWST from the containment recirculation sump is less than or equal to 1 gpm.

The methodology discussed below to determine the post-LOCA iodine and noble gas (iodine daughters) releases via the RWST vent has been previously used and accepted by NRC for the Beaver Valley AST application (SER to License Amendment Numbers 257 and 139 for License Nos DPR-66 and NPF-73, [ML032530204]) and for the Prairie Island LOCA Re-analysis [ML091490611]. A technical paper, "Modeling Radioactive Leakage from Atmospheric Tank Vents Following a LOCA", describing this methodology was presented by CB&I S&W Inc. in the ANS Summer Conference in 2007 and is published in Transactions of the American Nuclear Society Volume 96, Radiation Protection and Shielding Session I, pg 441. (Reference 49)

Dose consequences of RWST back-leakage assumes that leakage starts at the switchover to recirculation (829 second following the LOCA) and continues for 30 days. Per regulatory guidance, a safety factor of 2 is applied to the leak rate, i.e., a 2-gpm leakage rate is assumed for the full duration of the event, which is two times the allowable leakage of 1 gpm. Also, in accordance with RG 1.183, with the exception of noble gases, all fission products released from the fuel to the containment are instantaneously and homogeneously mixed in the sump water at the time of release. However, only iodine and their daughter products are released through RWST back-leakage since the particulates would remain in the sump water.

A significant portion of the iodine associated with sump water back-leakage into the RWST is retained within the RWST fluid due to the equilibrium iodine distribution balance between the RWST gas and liquid phases. The time dependent iodine partition coefficient takes into consideration the temperature and pH of the RWST liquid and sump fluid, the RWST liquid and gas volumes, and the temperature, pH and volume of the incoming leakage. The iodines that evolve into the RWST gas space as a result of the equilibrium iodine distribution balance, and the noble gas daughters of iodines, are released to the environment via the RWST vent, at a vent rate established by the temperature transient in the RWST (which includes the effect of decay heat), the increase in the liquid inventory of the RWST due to the incoming leakage, the gases evolving out of incoming leakage, and the environmental conditions outside the RWST.

The average time-dependent RWST iodine release fractions along with the fractional RWST gas venting rates (may be applied to the noble gas daughters of iodines) to the atmosphere from the

Units 1 and 2 RWSTs due to RWST back-leakage following switchover to the sump water recirculation mode of operation is summarized in Table 7.2-3. As discussed earlier, the release fractions / rates presented in Table 7.2-3 reflect a safety factor of 2 on the leak rates, i.e., are developed based on a RWST back-leakage of 2 gpm. The iodine released to the environment is assumed to be 97% elemental and 3% organic.

The equilibrium iodine concentration in the RWST gas space utilized to develop Table 7.2-3 is based on the iodine mass in the sump fluid entering the RWST vapor space as back-leakage or the total iodine mass contained in the RWST liquid, whichever results in higher RWST vapor phase concentrations. The RWST maximum venting rate averaged over an interval is primarily based on RWST back-leakage entering the RWST gas space and thermally equilibrating, and is used in conjunction with the higher RWST gas space iodine concentration to calculate an iodine mass release rate as a function of time. An interval based averaging approach is utilized in preparing Table 7.2-3 to reduce the number of input values to the dose analysis while preserving the boundaries for the time periods used for atmospheric dispersion; the actual iodine release calculated in an interval is normalized to the iodine mass leaking into the RWST during that time interval.

Examination of the average gas space venting rates indicate that after the first day, the noble gases formed by decay of iodine will primarily remain in the RWST during the 30 day period of evaluation and not be released. However, the dose consequence analysis conservatively releases the noble gases formed by decay of iodine, directly to the environment without taking any credit for tank holdup.

7.2.3.6 *Miscellaneous Equipment Drain Tank (MEDT) Leakage*

The DCPP Units 1 and 2 MEDT is a covered rectangular (12' x 5' x 10') stainless steel lined concrete tank located in the Auxiliary Building below El 60 ft. The MEDT tank vent is hard-piped to the Auxiliary Building ventilation ductwork; thus the airborne releases from the MEDT are ultimately discharged to the environment via the plant vent.

Following a LOCA, the MEDT will receive both post-LOCA sump fluids as well as non-radioactive fluids (i.e., ESF system leakage from the accident unit, as well as non-radioactive fluids from equipment drains and RWST leakage from the non-accident unit) which are hard-piped to the MEDT. As part of this application, DCPP is proposing to establish administrative acceptance criteria to ensure the total as-tested flow hard piped to the MEDT is less than 950 cc/min of ESF system leakage and 484 cc/min of non-radioactive fluid leakage.

Similar to the RWST back-leakage model, dose consequences due to releases from the MEDT assumes that leakage starts at the switchover to recirculation (829 second following the LOCA) and continues for 30 days. Per regulatory guidance, a safety factor of 2 is applied to the leak rate, i.e., 1900 cc/min of ESF system leakage and 968 cc/min of non-radioactive fluids into the MEDT is assumed for the full duration of the event, which is two times the allowable leakage. For purposes of bounding analyses, the boron concentration of the pre-existing fluid in the MEDT, as well as the incoming leakage is assumed to be at its upper bound levels. With the exception of noble gases, all fission products released from the fuel to the containment are instantaneously and homogeneously mixed in the sump water at the time of release. Only iodine and their daughter products are released through MEDT leakage since the particulates would remain in the sump water.

The methodology used to determine the post-LOCA iodine and noble gas releases via the MEDT vent and Plant Vent has been used previously in evaluating post-LOCA leakage into the RWST. Adaptation of the methodology to address overflows/room ventilation releases is straightforward with the room ventilation rate being treated as the tank exhaust rate. As discussed earlier in Section 7.2.3.5, this equilibrium based methodology for RWST vent release paths has been accepted by NRC for the Beaver Valley AST application (SER to License Amendment Numbers 257 and 139 for License Nos DPR-66 and NPF-73, [ML032530204]) and for the Prairie Island LOCA Re-analysis [ML091490611].

The transport model utilized to determine airborne releases from the MEDT takes into account the fact that the MEDT is a small tank with an auto-transfer capability which is PG&E Design Class II. Consequently, and for purposes of conservatism, it is assumed that a) the LOCA occurs when the MEDT water level is at the normal maximum setpoint to initiate auto transfer, b) the auto-transfer capability is not initiated because it is not a safety function, and c) the MEDT contents will spill over into the Equipment Drain Receiver Tank (EDRT) Room after the tank is full. Thus, for the post-LOCA scenario, the MEDT is conservatively assumed to overflow via its manway into the EDRT Room. The EDRT room drains into the Auxiliary Building Sump (ABS), which ultimately overflows into the U1 / U2 pipe tunnels. The ABS is also a covered rectangular (16' × 5' × 10') stainless steel lined concrete tank with a vent that is hard-piped to the Auxiliary building ductwork with a PG&E Design Class II auto transfer capability. The ABS is located adjacent to the MEDT.

The bounding transient release of iodine along with the gas venting rate to the atmosphere as a result of post-LOCA leakage of radioactive and non-radioactive fluid hard-piped into the MEDT is developed in 2 parts: a) prior to MEDT overflow and b) post MEDT overflow.

a) <u>Prior to MEDT overflow</u> - The iodines evolve into the MEDT gas space as a result of the equilibrium iodine distribution balance between the MEDT gas and liquid phases (either the MEDT liquid inventory or the incoming leakage), and are released to the environment via the plant vent, at a vent rate established by the temperature transient in the MEDT (including the effect of decay heat), the increase in the liquid inventory of the MEDT due to the incoming leakage, and the gases evolving out of the incoming leakage.

b) <u>After MEDT overflow</u> – The equilibrium iodine distribution balance is conservatively assumed to be between the iodine concentrations in the MEDT overflow liquid and the EDRT room (or U1/U2 pipe tunnels) ventilation flow (rather than the average concentration in the EDRT room (or U1/U2 pipe tunnels) free volume). This maximizes the iodine release rate. Thus, the iodines released are a sum total of the following:

   i) the iodines that evolve into the EDRT room air space as a result of the equilibrium iodine distribution balance between the spilled liquid from the MEDT (at the temperature of the MEDT), and the EDRT room ventilation flow, and is released to the environment via the plant vent, at the vent rate established by the EDRT room ventilation system, and

   ii) the iodines that evolve into the U1/U2 Pipe Tunnel air space as a result of the equilibrium iodine distribution balance between the spilled liquid from the MEDT (at the maximum temperature of the U1/U2 Pipe Tunnel), and the U1/U2 Pipe Tunnel

ventilation flow, and is released to the environment via the plant vent, at the vent rate established by the U1/U2 Pipe Tunnel ventilation system

The exhaust fans servicing the EDRT room and pipe tunnel are PG&E Design Class I. There is also a potential that the non-LOCA unit's ABVS will be operating with the flow being exhausted to the associated unit specific plant vent. Thus, it is conservatively assumed that the non-LOCA unit's ABVS is operating, and together with the accident units' exhaust fans, are providing the motive force to exhaust the airborne releases to the environment, unfiltered, via the respective plant vents.

The average time-dependent MEDT iodine release fractions, along with the fractional MEDT gas venting rates (which may be applied to the noble gas daughters of iodines prior to MEDT overflow) to the atmosphere following switchover to the sump water recirculation mode of operation, is summarized in Table 7.2-4. As discussed earlier, the release fractions / rates presented in Table 7.2-4 reflect a safety factor of 2 on the leak rates, i.e., are developed based on an input of 1900 cc/min of ESF system leakage and 968 cc/min of non-radioactive fluids into the MEDT. Through the use of extremely conservative assumptions, the calculated iodine release fractions / gas venting rates presented in Table 7.2-4 when used in combination with the analyzed ESF system leak rate, bound the iodine releases of all combinations of radioactive and non-radioactive leakages less than or equal to the leak rates analyzed. The iodine released from the ventilation system is assumed to be 97% elemental and 3% organic, and is released to the environment via the plant vent. In addition, the dose consequence analysis conservatively releases the noble gases formed by decay of iodine, directly to the environment without taking any credit for tank holdup.

7.2.4 Offsite Dose Assessment

Due to the delayed post-LOCA fuel release sequence of an AST model, and the rate at which aerosols and elemental iodine are removed from the containment, the maximum 2-hour EAB dose for a PWR LOCA typically occurs between 0.5 hrs to 2.5 hrs.

To establish the "worst case 2-hour release window" for the DCPP EAB dose, the integrated dose versus time for each of the six pathways discussed above was evaluated. The 0-2 hr EAB Atmospheric Dispersion Factor was utilized for all cases.

The analysis demonstrated that for DCPP the *maximum 2 hour EAB dose will occur, as a result of the RHR pump seal failure, between T=24 hrs to T=26 hrs, and is unrelated to the post-LOCA fuel release sequence associated with AST.*

For purposes of completeness, the "worst case 2-hour release window" for the DCPP EAB dose is estimated for 2 cases:

a) Without the RHR pump seal failure release, and

b) With the RHR pump seal failure release.

The direct shine dose at the EAB due to a) the airborne activity inside containment, and b) the sump water collected in the RWST due to RWST back-leakage, was also evaluated. Based

on the results of the EAB evaluation which determined that the dose contribution due to direct shine was minimal (<0.01 rem), the dose at the LPZ due to direct shine is deemed negligible.

The bounding EAB and LPZ dose following a LOCA at either unit is presented in Section 8.

7.2.5 Control Room Occupancy Dose

7.2.5.1 Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical LOCA-specific assumptions associated with control room response and activity transport.

<u>Timing for Initiation of CRVS Mode 4 (if applicable)</u>:

- An SI signal will be generated at t = 6 sec following a LOCA.
- The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The CR dampers are fully closed 10 secs later, or at t=44.2 secs (i.e., 6 + 28.2 + 10). The 2 second SI signal processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.
- In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=18 secs (i.e., 6 + 2 secs signal processing time + 10 sec damper closure time).

<u>Control Room Atmospheric Dispersion Factors</u>:

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a LOCA at either unit are provided in Table 7.2-5. The χ/Q values presented in Table 7.2-5 take into consideration the various release points-receptors applicable to the LOCA to identify the bounding χ/Q values applicable to a LOCA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

7.2.5.2 Direct Shine Dose to the Control Room from External and Contained Sources

The direct shine dose to an operator in the control room due to contained or external sources resulting from a postulated LOCA is calculated using CB&I S&W Inc. point kernel shielding computer program SW-QADCGGP. The post-LOCA gamma energy release rates (MeV/sec) and integrated gamma energy release (MeV-hr/sec) in the various external sources are developed with CB&I S&W Inc. computer program PERC2.

The LOCA sources that could potentially impact the CR operator dose due to direct shine are identified below.

1. Direct shine from containment – shine from the airborne source in the containment structure via the bulk shielding (3'-8" thick concrete walls below the bendline, 2'-6" thick concrete dome), including shine through one of the main steam line penetrations and the Personnel Hatch facing the CR
2. Direct shine from the contaminated cloud outside the control room pressure boundary resulting from containment leakage, ESF system leakage, RHR Pump seal leakage, RWST back-leakage, MEDT leakage - shine occurs through the CR walls, via wall penetrations such as CR doors to the outside, and from the airborne activity in cable spreading room below via CR floor penetrations.
3. Dose due to scattered gamma radiation through wall penetrations from the CRVS filters located in the adjacent mechanical equipment room.
4. Direct shine from the sump fluid that is postulated to collect in the RWST

Cloud shine through CR doorways was found to be the most significant of all the identified contained or external post-LOCA radiation sources listed above, followed by the dose contribution through the CR floor penetrations. Note that other radiation sources were identified and deemed insignificant due to the presence of significant shielding between the operator and the radiation sources. Examples of these dose contributors include most of the large and small electrical and pipe penetrations in the Containment outer wall that faces the CR, and the ESF system piping and components located in the Auxiliary Building.

The direct shine dose estimate in the CR takes into consideration the function of Room 506 (which serves as a control room foyer adjacent to the Shift Supervisor's office), where occupancy is deemed to be minimal; i.e., conservatively estimated at less than 5% of the total time spent daily in the control room. The above "occupancy adjustment" is utilized to determine the maximum 30-day integrated dose in Control Room (i.e., the total direct shine dose in the CR includes the 30-day dose in Room 506 adjusted by the referenced occupancy factor).

The bounding control room operator dose following a LOCA at either unit is presented in Section 8.

7.2.6 Control Room Operator Dose during Access

Diablo Canyon assumes that the dose received by the operator during routine access to the control room for the 30 day period following the LOCA is minimal. Thus, as long as some reasonable margin exists between the regulatory limit and the estimated dose to the operator during control room occupancy, the additional dose due to ingress / egress can be accommodated.

This approach is consistent with the approach used by other licensees, and is reasonable since a) transit to and from the control room is only expected after the first 24 hours following the accident by which time the airborne levels inside containment has reduced significantly due to the use of active fission product removal mechanisms such as containment sprays, and by radioactive decay, and b) the operator is protected from radioactive ESF fluids by the shielding provided by the buildings that house such equipment. In addition, it is expected that during a

postulated event, access to the control room will be controlled by Health Physics and the Emergency Plan based on real time data, with the purpose of minimizing personnel dose.

It is also noted that the dose received by the operator during transit outside the control room is not a measure of the "habitability" of the control room which is defined by the radiation protection provided to the operator by the control room shielding and ventilation system design. Thus, the estimated dose to the operator during routine post-LOCA access to the control room is addressed separately from the control room occupancy dose which is used for the demonstration of control room habitability.

DCPP's current licensing basis provides an estimated dose contribution to the operator during egress and ingress to the control room following a LOCA. The access dose estimate currently reported in the UFSAR has been part of DCPP's licensing basis since the original Safety Analysis Report. While several of the input values and the calculations for the original ingress/egress dose values are no longer available, selected inputs, the ingress/egress dose values and methodology are still in the UFSAR.

To address the existing licensing basis, a TEDE dose is estimated for operator access to the control room. Because RG 1.183 does not provide guidance on determining the egress and ingress to the control room following an accident, the same inputs used to estimate the current licensing basis values for access to the control room, along with the associated dose estimate presented in the UFSAR, are used to determine the TEDE dose estimate for ingress/egress.

With this application, and consistent with the assumption made by other licensees, PG&E is proposing to demonstrate that the dose contribution due to routine ingress/egress during the accident is minimal.

In accordance with DCPP original licensing basis, radiation exposures to personnel during egress and ingress (i.e., during routine access to the control room for the duration of the accident) could result from the following sources:

(1) Airborne activity in the containment leakage plume

(2) Direct gamma radiation from fission products in the containment structure

Post-accident egress-ingress exposures were based on 27 outbound excursions, from the control room to the site boundary, and 26 inbound excursions, from the site boundary to the control room. It was estimated that each excursion would take 5 minutes, and no credit was taken for breathing apparatus or special whole body shielding.

Egress-ingress thyroid and whole body exposures from airborne activity are functions of containment activity, containment leakage, atmospheric dispersion, and excursion time. The airborne activity concentrations were calculated and the then conventional exposure equations from Regulatory Guide 1.4, Revision 1, were used to calculate gamma, beta, and thyroid exposures (Reference 8). The exposure from betas was calculated on the basis of an infinite uniform cloud, and exposure from gammas was calculated on the basis of a semi-infinite cloud.

Because of the containment shielding and short excursion time, egress-ingress containment shine exposures were estimated to be small. The shine model assumed a cylindrical radiation

source having the same radius and height as the containment structure with a 3.5-foot-thick concrete shield surrounding it. The receptor point was assumed to be a distance of 10 meters from the outer surface of the containment wall.

The estimated egress-ingress exposures developed in support of DCPP original licensing basis are listed below.

- The dose to control room personnel during egress / ingress from airborne fission products in the containment leakage plume: 0.0066 rem gamma, 0.0243 rem beta, and 4.72 rem thyroid

- The dose to control room personnel during egress / ingress as a result of direct radiation shine from the fission products in the containment structure: 0.022 rem.

As part of this AST licensing application, DCPP has identified additional post-LOCA fission product release pathways, as discussed in Section 7.2.2. The postulated effect of these additional radioactivity release paths, as well as the implementation of AST, on the estimated dose to control room personnel during routine egress / ingress takes into consideration the following:

- The transport models used to develop the dose to the control room operator during occupancy address a control room occupancy factor of 1.0 till t=24 hours after the accident. This implies that during the first 24 hours the control room operator stays in the control room. This is also reflected in the DCPP original licensing basis which addresses one more outbound trip than the inbound trips.

- Routine ingress / egress to the control room during the 30 day period following a LOCA falls into the mission dose category as discussed in NUREG-0737, II. B. 2 (Reference 15).

- NUREG-0737, Item II. B. 2 states that leakage of systems outside containment need not be considered as potential sources.

Based on the above considerations, the dose consequences of the additional activity release paths addressed in Section 7.2.2 (and listed below), is addressed as follows:

- <u>Containment Pressure /Vacuum relief release</u> - this release occurs at accident initiation (before t=24hr), so there is no dose contribution to the control room operator during routine ingress /egress during the 30 day period following the accident.

- <u>Containment leakage</u>:

    - The airborne activity in the containment after t=24 hours with an AST source term is primarily *100% of the core noble gases and 0.06% of the core iodines* that were released to containment.

      <u>Note</u>: The iodine source term at t=24 hrs is essentially the organic iodines released to the containment which are not affected by sprays, and which per Regulatory Guide 1.183,

represent 0.06% of the core iodines (i.e., 0.15% of the 40% core iodines released to containment atmosphere at accident initiation). Also, the essentially particulate nature of the radioactivity release associated with an AST source term, and the effectiveness of particulate removal by sprays / settling makes the dose contribution from the particulate source minimal after t=24 hours.

- The corresponding airborne activity in the containment after t=24 hours for a TID-14844 source term is *100% of the core noble gases and 1% of the core iodines.*

  Note: Per Regulatory Guide 1.4, Revision 1, the organic iodines released to the containment is 4% of the 25% iodines released to containment atmosphere at accident initiation.

- Based on the above it is concluded that after t=24hrs:

  a. The dose consequences due to containment leakage based on a TID-14844 based scenario will bound the dose consequences based on an AST scenario.

  b. Since the thyroid dose is primarily due to iodines, the associated dose to the operator will vary proportionately to the amount of iodine airborne in containment. Thus the thyroid dose to the operator during ingress /egress for an AST scenario may be estimated by adjusting the TID-14844 based dose by the ratio of the iodine estimated to be airborne in containment for each of the scenarios. As noted earlier, the current licensing basis thyroid dose to the operator during ingress / egress is 4.72 rem. The corresponding thyroid dose based on an AST scenario is estimated to be 4.72 x 0.06 = 0.28 rem thyroid.

- The RHR Pump Seal Failure, ESF System Leakage, RWST back leakage and MEDT leakage – All of these releases are based on leakage of systems outside containment. In accordance with NUREG-0737 II. B. 2, the dose contribution due to these sources need not be considered for access calculations.

To address the TEDE dose acceptance criteria applicable to the use of AST, the original licensing basis egress-ingress exposures have been updated as noted below using the guidance provided in 10CFR20.1003 (Reference 61). The referenced federal regulation defines TEDE as the sum of the deep dose equivalent for external exposures (i.e., external whole body exposure) and the committed effective dose equivalent for internal exposures (i.e., sum of the product of the weighting factor applicable to each organ irradiated and the dose to that organ). Per 10CFR20.1003, the weighting factor for the whole body is 1.0 and for the thyroid is 0.03. While the weighting factor for beta radiation is undefined, the contribution of the beta dose to the total effective dose equivalent is expected to be insignificant. Therefore,

- Radiation from airborne fission products in the containment leakage plume to the control room personnel during egress ingress is approximately 0.0066 rem + 0.28 x 0.03 rem, i.e., 0.015 rem TEDE
- Direct radiation from the fission products in the containment structure to control room personnel during egress ingress is 0.022 rem TEDE.

Thus the total dose to the control room operator during access is estimated to be 0.037 rem TEDE; i.e., 0.015 + 0.022. This value is 1% of the estimated operator dose due to control room occupancy following a LOCA (Refer to Table 8.1-1) and is therefore considered to be minimal.

7.2.7 Technical Support Center Dose

In accordance with current licensing basis, the Technical Support Center (TSC) design has been evaluated for the LOCA. An assessment of the dose consequences in the TSC due to due to airborne radioactivity releases following Non-LOCA events is summarized in Section 7.9.

CB&I S&W Inc. computer code PERC2 is used to calculate the dose to TSC personnel due to airborne radioactivity releases following a LOCA. The direct shine dose to an operator in the TSC due to contained or external sources resulting from a postulated LOCA is calculated using CB&I S&W Inc. point kernel shielding computer program SW-QADCGGP. The post-LOCA gamma energy release rates (MeV/sec) and integrated gamma energy release (MeV-hr/sec) in the various external sources are developed with computer program PERC2.

The TSC serves both units and is located at El 104' on the south-west side of the Unit 2 turbine building and is shared between Unit 1 and Unit 2. The north and south walls are made up of 2'-2" of concrete, whereas the east and west walls are made of 1'-4" and 1'-6" of concrete, respectively. The floor and ceiling thickness / material reflect a minimum of 1'-0" and 1'-8" of concrete, respectively.

As part of this application, DCPP proposes to process the TSC normal ventilation intake flow through a HEPA filter. In addition, the nominal air intake flowrate during normal operations will be 500 cfm. The above air intake is filtered through the referenced HEPA filter and drawn into the TSC envelope which has a free volume of 51,250 ft^3. The TSC normal intake is isolated and the TSC ventilation placed into filtered / pressurized Mode 4 operation by manual operator action within 2 hours of the LOCA.

The post-accident pressurization flow to the TSC is provided via the CRVS Mode 4 pressurization intakes (i.e., 1 per unit, each located on either side of the Turbine Building). As noted in Section 7.1, the DCPP CR pressurization air intakes have dual ventilation outside air intake design. The nominal air intake flowrate during the TSC pressurization mode is 500 cfm.

Mode 4 operation of the TSC utilizes the CRVS. As discussed in Section 7.1, CRVS Mode 4 operation (which also serves the TSC) utilizes redundant PG&E Design Class I radiation monitors located at each pressurization air intake and has the provisions of acceptable control logic to automatically select the least contaminated inlet at the beginning of the accident, and manually select the least contaminated inlet during the course of the accident. Thus, during Mode 4 operation the TSC dose consequence analysis can utilize the χ/Q values for the more favorable pressurization air intake reduced by a factor of 4 to credit the "dual intake" design (See Section 5.2 for additional details).

The allowable methyl iodide penetration and filter bypass for the TSC Mode 4 Charcoal Filter is <2.5% and <1%, respectively. Thus in accordance with GL 99-02 (Reference 41), the CR charcoal filter efficiency for elemental and organic iodine used in the TSC dose analysis is 100%

- [(2.5% + 1%) x 2] = 93%. The acceptance criteria for the TSC normal operation and Mode 4 HEPA filters is "penetration plus system bypass" < 1.0%. Thus, using methodology similar to the charcoal filters, the HEPA filter efficiency for particulates used in the TSC dose analysis is 100% - [(1%) x 2] = 98%.

During TSC Mode 4 operation, the TSC air is also recirculated through the same filtration unit as the pressurization flow. The air flow allowable through the pressurization charcoal / HEPA filter during TSC Mode 4 is 1000 cfm. This flow is comprised of a minimum filtered recirculation flow of 500 cfm and the previously discussed pressurization flow of 500 cfm, for a total flow through the filtration unit of 1000 cfm (pressurization plus recirculation).

Unfiltered inleakage into the TSC during Mode 1 and Mode 4 is assumed to be 60 cfm (includes 10 cfm for ingress/egress based on the guidance provided in SRP 6.4).

For purposes of estimating the post-LOCA dose consequences, the DCPP TSC is modeled as a single region. When in TSC Mode 4, the Mode 1 intakes are isolated and outside air is a) drawn into the TSC through the filtered emergency intakes; b) enters the TSC as infiltration, and c) enters the TSC during operator egress/ingress.

The dose assessment model utilizes nominal values for the ventilation intake flowrates since the intake pathways (normal as well as accident) are filtered, thus the controlling dose contributor is the unfiltered inleakage. The effect of intake flow uncertainty on the TSC dose is expected to be insignificant.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / TSC receptors applicable to a LOCA at either unit are provided in Table 7.2-6. The χ/Q values presented take into consideration the various release points-receptors applicable to the LOCA to identify the bounding χ/Q values applicable to an accident at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5.

The direct shine dose into the TSC due to the external cloud and contained sources is calculated in a manner similar to that described for the control room in Section 7.2.5.2. The LOCA sources that could potentially impact the TSC operator dose due to direct shine are identified below.

1. Direct shine from containment – shine from the airborne source in the containment structure via the bulk shielding (3'-8" thick concrete walls below the bendline, 2'-6" thick concrete dome), including shine through the Personnel Hatch facing the TSC
2. Direct shine from the contaminated cloud outside the TSC pressure boundary resulting from containment leakage, ESF system leakage, RHR Pump seal leakage, RWST back-leakage, MEDT leakage - shine occurs through the TSC walls and via wall penetrations such as TSC doors to the outside.
3. Dose due to scattered gamma radiation through wall penetrations from the TSC filters located in the adjacent mechanical equipment room, and scatter past labyrinths provided for selected doors.

Note that other radiation sources were identified and deemed insignificant due to the presence of significant shielding between the operator in the TSC and the radiation sources.

Table 7.1-2 lists key assumptions / parameters associated with DCPP TSC design.

The bounding TSC operator dose following a LOCA at either unit is presented in Section 8.

7.3 Fuel Handling Accident (FHA)

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a FHA. In accordance with current licensing basis, an FHA is assumed to occur in the Spent Fuel Pool located in the Fuel Handling building (FHB), or in the Containment.

This event postulates that a spent fuel assembly is dropped during refueling in the Spent Fuel Pool (SFP) located in the FHB, or in the reactor cavity located in the Containment. In accordance with current licensing basis, all of the fuel rods (264 rods) in the dropped fuel assembly are assumed to be damaged; thus all of the activity in the fuel gap of the dropped assembly is assumed to be instantaneously released into the SFP or into the reactor cavity. As documented in the NRC SER for License Amendments 8 and 6 to DCPP Facility Operating License Nos. DPR-80 and DPR-82, respectively (Reference 58), the assumption that all fuel rods in one assembly rupture is conservative because the kinetic energy available for causing damage to a fuel assembly dropped through water is fixed by the drop distance. The kinetic energy associated with the maximum drop height for a fuel handling accident is not considered sufficient to rupture the equivalent number of fuel rods of one assembly in both the dropped assembly and the impacted assembly.

This assessment follows the guidance provided for the FHA in pertinent sections of RG 1.183 including Appendix B. As discussed in Section 4.3, and as part of the change in licensing basis requested with this application, the core gap activity is assumed to be comprised of 8% of the core I-131 inventory, 23% of the core I-132 inventory, 35% of the core Kr-85 inventory, 4% of the remaining core noble gas inventory, 5% of the remaining core halogen inventory, and 46% of the core alkali metal (Cesium and Rubidium) inventory. Table 7.3-1 lists the key assumptions / parameters utilized to develop the radiological consequences following an FHA at either location and at either unit.

Current DCPP procedures prohibit movement of recently irradiated fuel which is defined as fuel that has occupied part of a critical reactor core within the previous 100 hours. As a part of this application, it is proposed that the definition of recently irradiated fuel at DCPP be updated to reflect fuel that has occupied part of a critical reactor core within the previous 72 hours. Table 7.3-2 provides the gap activity inventory of the noble gases, iodines and alkali metals in a single fuel assembly at t=72 hrs post reactor shutdown.

DCPP TS 3.7.15 requires the SFP water level to be ≥23 feet over the top of irradiated fuel assemblies seated in the storage racks. TS 3.9.7 requires the refueling cavity water level to be maintained ≥23 feet above the top of the reactor vessel flange. Additional margin is provided through operating procedures.

The fission product inventory in the fuel rod gap of all the rods in the damaged assembly are assumed to be instantaneously released into the spent fuel pool or reactor cavity, both of which have a minimum of 23 ft of water above the damaged fuel assembly. A radial peaking factor of 1.65 is applied to the activity release.

Per RG 1.183, the radioiodine released from the fuel gap is assumed to be 95% particulate (CsI), 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the fuel pool (pH less than 7), the CsI is assumed to immediately disassociate and re-evolve as elemental iodine, thus changing the chemical form of iodine to 99.85% elemental and 0.15% organic. In addition, and per RG 1.183, an iodine decontamination factor of 200 is assumed for the SFP / reactor cavity. Noble gases and unscrubbed iodines rise to the water surface where they are mixed in the available air space. All of the alkali metals released from the gap are retained in the pool. In accordance with RG 1.183, the chemical form of the iodines above the pool is 57% elemental and 43% organic.

Per RG 1.183, the activity released due to an FHA is assumed to be discharged to the environment in a period of 2 hrs (or less if the ventilation system promotes a faster release rate).

FHA in the FHB

The radioactivity release pathways following an FHA in the FHB are established taking into consideration the following Administration Controls:

During fuel movement in the FHB:

- The movable wall is put in place and secured
- No exit door is propped open
- One FHBVS exhaust fan is operating (The supply fan flow (if operating) has been confirmed by design to have less flow than the exhaust fan)

Operation of the Fuel Handling Building Ventilation system (FHBVS) with a minimum of 1 exhaust fan operating and all significant openings administratively closed will ensure negative pressure in the FHB which will result in post-accident environmental release of radioactivity occurring via the Plant Vent. The activity release due to the FHA in the FHB is assumed to be discharged to the environment as follows:

- A maximum release rate of 46,000 cfm via the Plant Vent due to operation of the FHBVS with a closed FHB configuration.

- A maximum conservatively assumed outleakage of 500 cfm occurring from the closest edge of the FHB to the control room normal intake (i.e., 30 cfm outleakage is assumed for ingress/egress; 470 cfm is assumed for outleakage from miscellaneous gaps/openings in the FHB structure).

It has been determined that for the FHA in the FHB, the actual release rate lambda based on the FHBVS exhaust (i.e., 8.7 hr^{-1}) is larger than the release rate applicable to "a 2-hr release" per regulatory guidance (i.e., 3.45 hr^{-1}). Thus the larger exhaust rate lambda associated with

FHBVS operation plus the exhaust rate lambda for the 500 cfm outleakage is utilized in the analysis.

FHA in the Containment

The potential radioactivity release pathways following a FHA in the containment are established taking into consideration

- Operation of the containment purge system which would result in radioactivity release via the plant vent

- Plant Technical Specification Section 3.9.4 that allows for an "open containment" during fuel movement in containment during offload or reload. The most significant containment opening closest to the Control room normal operation intake is the equipment hatch. The equipment hatch is an approximately 20-ft wide circular opening in containment. In the event the containment purge system ceased to operate (a viable scenario since it is single train and has non-vital power), the density driven convective flow out of the equipment hatch (due to the thermal gradient between inside and outside containment conditions), could be significant.

It has been determined that for the FHA in the Containment, the release rate assuming a regulatory based 2 hr release is larger than that dictated by the containment purge ventilation system, or convective flow out of the equipment hatch. Thus the regulatory based release rate (i.e., 3.45 hr^{-1}), is utilized for this analysis. Review of the atmospheric dispersion factors associated with the plant vent vs the equipment hatch indicates that dose consequences due to releases via the equipment hatch will be bounding.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. Since the FHA is based on a 2-hour release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

The bounding EAB and LPZ dose following a FHA at either location and at either unit is presented in Section 8.

Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical FHA-specific assumptions associated with control room response and activity transport.

Design Basis FHA (occurs at t=72 hours after reactor shutdown)

The current licensing basis analyses supporting the DBA FHA in the FHB and the FHA in the Containment assume that the CR remains in normal operation mode for the duration of the accident.

As part of this application, credit is taken for PG&E Design Class I area radiation monitors located at the Control Room (CR) normal intakes (1-RE-25/26, 2-RE-25/26) to initiate CRVS Mode 4 (filtered / pressurized accident ventilation) upon detection of high radiation levels at the CR normal intakes as a result of an FHA.

An analytical safety limit of 1 mR/hr for the gamma radiation environment at the CR normal operation air intakes has been used in the FHA analyses to initiate CRVS Mode 4. As discussed in Section 2.2, the actual monitor trip setpoint will be lower to include the instrument loop uncertainty.

The radiation monitor response time is primarily dependent on the type of monitor, the setpoint, the background radiation levels and the magnitude of increase in the radiation environment at the detector location.

For a monitor with an instrument time constant of "τ" (2 seconds) and a background of 0.05 mR/hr, the response time "t" to a high alarm Setpoint (HASP < 1 mr/hr), for a step increase of radiation level DR (mR/hr) is determined by solving the following equation that represents the monitor reading approaching the final reading exponentially.

$$HASP = 0.05 + DR(1 - e^{-\frac{t}{\tau}})$$

It is determined that a DBA FHA (i.e., occurs at 72 hrs post shutdown) will result in a radiation environment at the CR normal operation intakes that greatly exceed the analytical limit of 1 mR/hr for initiating CRVS Mode 4. This will result in an almost instantaneous generation of a radiation monitor signal to initiate CRVS Mode 4 (radiation monitor response time is estimated to be < 1 sec). For purposes of conservatism, and since the delay in isolation of the normal intake has a significant impact on the estimated dose consequences, the analysis conservatively assumes a monitor response time to the HASP of 20 secs.

As discussed in Section 7.1, when crediting CRVS Mode 4, the FHA dose consequence analyses does not address the potential effects of a LOOP. Thus delays associated with diesel generator sequencing are not required.

Therefore, the time delay between the arrival of radioactivity released due to a DBA FHA at both the CR normal Intakes (assumed to be instantaneous) and CRVS Mode 4 operation is estimated to be the sum total of the monitor response time (20 secs), the signal processing time (2 secs) and the damper closure time (10 secs) for a total delay of 32 seconds.

<u>Delayed FHA:</u>

It is recognized that the response time for radiation monitors are dependent on the magnitude of the radiation level / energy spectrum of the airborne cloud at the location of the detectors, which in turn are dependent on the fuel assembly decay time. Thus an additional case is considered for each of the two FHA scenarios described above (i.e., a FHA in the FHB and a FHA in Containment) when determining the dose to the CR operator; i.e., a case that reflects a delayed FHA at Fuel Offload or a FHA during Reload, occurring at a time when the fuel has decayed to such an extent that the radiation environment at the CR normal intake radiation

monitors is just below the setpoint; thus the CR remains in normal operation mode and CRVS Mode 4 is not initiated.

The analyses determined that the dose consequences of a DBA FHA bound that associated with the delayed FHA for both the FHA in the FHB and the FHA in the containment.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an FHA at either location, and at either unit, are provided in Table 7.3-3. The χ/Q values presented in Table 7.3-3 take into consideration the various release points-receptors applicable to the FHA to identify the bounding χ/Q values applicable to a FHA at either unit and at either location, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

The bounding Control Room dose following a FHA at either location and at either unit is presented in Section 8.

7.4 Locked Rotor Accident (LRA)

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LRA.

This event is caused by an instantaneous seizure of a primary reactor coolant pump (RCP) rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage is predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed Steam Generator (SG) tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system via the 10% ADVs and MSSVs. Following reactor trip, and based on an assumption of a Loss of Offsite Power (LOOP) coincident with reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling. DCPP has established that the LOL event generates the maximum primary to secondary heat transfer and the LRA assumes these same conservatively bounding secondary steam releases.

Regulatory guidance provided for the LRA in pertinent sections of RG 1.183 including Appendix G is used to develop the dose consequence model. The fuel gap fractions used for non-LOCA events are discussed in Section 4.3. Table 7.4-1 lists the key assumptions / parameters utilized to develop the radiological consequences following a LRA.

Consistent with current licensing basis, the LRA is postulated to result in 10% fuel failure resulting in the release of the associated gap activity. As discussed in Section 4.3, and as part of the licensing basis change requested with this application, the core gap activity is assumed to be comprised of 8% of the core I-131 inventory, 23% of the core I-132 inventory, 35% of the core Kr-85 activity, 4% of the remaining core noble gas inventory, 5% of the remaining core halogen inventory, and 46% of the core alkali metal (Cesium and Rubidium) inventory.

In accordance with RG 1.183, the activity released from the fuel is assumed to be released instantaneously and mixed homogenously through the primary coolant mass and transmitted to the secondary side via primary to secondary SG tube leakage. A radial peaking factor of 1.65 is applied to the activity release from the fuel gap. The activity associated with the release of the primary to secondary leakage of normal operation RCS, (at Technical Specification levels) via the MSSVs/10% ADVs are insignificant compared to the failed fuel release and are therefore not included in this assessment.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the LRA dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

The chemical form of the iodines in the gap are assumed to be 95% particulate (CsI), 4.85% elemental and 0.15% organic. The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a Westinghouse Owners Group (WOG) Program and demonstrated to be insignificant; therefore, the gap iodines are assumed to have a partition coefficient of 100 in the SG. The iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The gap noble gases are released freely to the environment without retention in the SG whereas the particulates are assumed to be carried over in accordance with the design basis SG moisture carryover fraction.

The condenser is assumed unavailable due to the loss of offsite power. Consequently, the radioactivity release resulting from a LRA is discharged to the environment from all steam generators via the MSSVs and the 10% ADVs. The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the RHR system, and environmental releases are terminated.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For the LRA, the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a LRA at either unit is presented in Section 8.

Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical LRA-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4 (if applicable):

The LRA does not initiate any signal which could automatically start the control room emergency ventilation. Thus the dose consequence analysis for the LRA assumes that the CR remains in normal operation mode.

Control Room Atmospheric Dispersion Factors

As noted in Section 5.0, because of the proximity of the MSSV/10% ADVs to the CR normal intake of the affected unit (~ 15 ft above the CR intake, horizontal distance is ~ 1.5 meters), and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation CR intake of the faulted unit (closest to the release point) will be insignificant. Therefore, only the unaffected unit's CR normal intake is assumed to be contaminated by a release from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an LRA at either unit are provided in Table 7.4-2. The χ/Q values presented in Table 7.4-2 take into consideration the various release points-receptors applicable to the LRA to identify the bounding χ/Q values applicable to a LRA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

The bounding Control Room dose following a LRA at either unit is presented in Section 8.

7.5 Control Rod Ejection Accident (CREA)

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a CREA.

This event consists of an uncontrolled withdrawal of a control rod from the reactor core. The CREA results in reactivity insertion that leads to a core power level increase, and under adverse combinations of circumstances fuel failure, and a subsequent reactor trip. In this case, some of the activity in the fuel rod gaps would be released to the coolant and in turn to the inside of the containment building. As a result of pressurization of the containment, some of this activity could leak to the environment.

Following reactor trip, and based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling. DCPP has established that the LOL event generates the maximum primary to secondary heat transfer and the CREA assumes these same conservatively bounding secondary steam releases.

Regulatory guidance provided for the CREA in pertinent sections of RG 1.183 including Appendix H is used to develop the dose consequence model. Table 7.5-1 lists the key assumptions / parameters utilized to develop the radiological consequences following a CREA.

Consistent with current licensing basis, the CREA is postulated to result in 10% fuel failure resulting in the release of the associated gap activity. Per RG 1.183, the core gap activity is assumed to be comprised of 10% of the core noble gases and halogens. A radial peaking factor of 1.65 is applied to the activity release from the fuel gap.

In accordance with guidance provided in RG 1.183, two independent release paths to the environment are analyzed: first, via *containment leakage* of the fission products released due to the event from the primary system to containment, *assuming that the containment pathway is the only one available*; and second, via releases from the *secondary system*, outside containment, following primary-to-secondary leakage in the steam generators, assuming that *the latter pathway is the only one available*.

The actual doses resulting from a postulated CREA would be a composite of doses resulting from portions of the release going out via the containment building and, portions via the secondary system. If regulatory compliance to dose limits can be demonstrated for each of the scenarios, the dose consequence of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios.

The DCPP CREA dose consequence analysis evaluates the following two scenarios.

<u>Scenario 1</u>: The failed fuel resulting from a postulated CREA is released into the RCS, which is released in its entirety into the containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment, and then released to the environment at the containment technical specification leak rate for the first 24 hrs and at half that value for the remaining 29 days.

<u>Scenario 2</u>: The failed fuel resulting from a postulated CREA is released into the RCS which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Environmental releases occur from the steam generators via the MSSVs and 10% ADVs.

The chemical composition of the iodine in the gap is assumed to be 95% particulate (CsI), 4.85% elemental and 0.15% organic. However, because the sump pH is not controlled following a CREA, it is conservatively assumed that the iodine released via the containment leakage pathway has the same composition as the iodine released via the secondary system release pathway; i.e.; it is assumed that for both scenarios, 97% of all halogens available for release to the environment are elemental, while the remaining 3% is organic.

Scenario 1: Transport From Containment

The failed fuel activity released due to a CREA into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the technical specification leak rate of 0.001 per day for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen or noble gas concentrations airborne in the containment. Per RG 1.183, the chemical composition of the iodine in the gap fuel is 95% particulate (CsI), 4.85% elemental and 0.15% organic. However, since no credit is taken for the actuation of sprays or pH control, the iodine released via containment leakage pathway is

assumed to have the same composition as iodine activity released to the environment from the secondary coolant; i.e.; 97% elemental and 3% organic. Environmental releases due to containment leakage can occur unfiltered as a diffuse source from the containment wall, and as a point source via the containment penetration areas or the Plant Vent. The dose consequences are estimated based on the worst case atmospheric dispersion factors, i.e., an assumed environmental release via the containment penetration areas.

Scenario 2: Transport from Secondary System

The failed fuel activity released due to a CREA into the RCS is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary SG tube leakage. The activity associated with the release of the initial inventory in secondary steam/liquid, and primary to secondary leakage of normal operation RCS, (both at Technical Specification levels) via the MSSVs/10% ADVs are insignificant compared to the failed fuel release, and are therefore not included in this assessment.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the CREA dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant; therefore, the gap iodines have a partition coefficient of 100 in the SG. The gap noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to the loss of offsite power. Consequently, the radioactivity release resulting from a CREA is discharged to the environment from steam generators via the MSSVs and the 10% ADVs. Per RG 1.183, 97% of all halogens available for release to the environment via the Secondary System are elemental, while the remaining 3% are organic. The SG releases continue until shutdown cooling is initiated via operation of the RHR system (10.73 hours after the accident) and environmental releases are terminated.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For Scenario 1 (release via Containment leakage), the worst case 2-hour period occurs during the first 2 hours). For Scenario 2 (release via secondary side), the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a CREA at either unit for both scenarios are presented in Section 8.

Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical CREA-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

The time to generate a signal to switch CRVS operation from Mode 1 to Mode 4 is based on the containment pressure response following a 2 inch small-break LOCA (SBLOCA), and the fact that at DCPP, a Containment High Pressure signal will initiate a SI signal which will automatically initiate CRVS Mode 4 pressurization. The containment pressure response analysis for a 2 inch SBLOCA shows that the 5 psig setpoint for Containment High Pressure is reached in ~ 150 seconds after the SBLOCA. As indicated earlier, releases to the containment following a CREA are through a ruptured control rod drive mechanism housing. The control rod shaft diameter is 1.840 inches and the RCCA housing penetration opening is 4 inches in diameter. Based on the above and for the purposes of conservatism, the time to generate the Containment High Pressure SI signal following a CREA is assumed to be double the value applicable to the 2 inch SBLOCA, or 300 seconds.

Based on the above, following a CREA,

- An SI signal will be generated at t = 300 sec following a CREA.
- The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The CR dampers are fully closed 10 secs later, or at t=338.2 secs (i.e., 300 + 28.2 + 10). The 2 second SI signal processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.
- In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=312 secs (i.e., 300 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors:

As noted in Section 5.0, because of the proximity of the MSSV/10% ADVs to the CR normal intake of the affected unit (~ 15 ft above the CR intake, horizontal distance is ~ 1.5 meters), and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation CR intake of the faulted unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's CR normal intake is assumed to be contaminated by a release from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a CREA at either unit are provided in Table 7.5-2. The χ/Q values presented in Table 7.5-2 take into consideration the various release points-receptors applicable to the CREA to identify the bounding χ/Q values applicable to a CREA at either unit,

and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

The bounding Control Room dose following a CREA at either unit is presented in Section 8.

7.6 Main Steam Line Break (MSLB)

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a MSLB.

This event consists of a double-ended break of one main steam line. The analysis focusses on a MSLB *outside* the containment since a MSLB inside containment will clearly result in a lesser dose to a control room operator or to the offsite public due to hold-up of activity in the containment.

Following a MSLB, the affected SG rapidly depressurizes and releases the initial contents to the environment via the break. Based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs / 10% ADVs of the intact steam generators are used to cool down the reactor until initiation of shutdown cooling. The activity in the RCS leaks into the faulted and intact steam generators via SG tube leakage and is released to the environment from the break point, and from the MSSVs / 10% ADVs, respectively.

Regulatory guidance provided for the MSLB in pertinent sections of RG 1.183 including Appendix E is used to develop the dose consequence model. Table 7.6-1 lists the key assumptions / parameters utilized to develop the radiological consequences following a MSLB.

No melt or clad breach is postulated for the DCPP MSLB event. Thus, and in accordance with RG 1.183, Appendix E, item 2, the activity released is based on the maximum coolant activity allowed by the plant technical specifications. The plant technical specifications focus on the noble gases and iodines. In addition, and per RG 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike and b) an accident-initiated iodine spike.

a. Pre-accident Iodine Spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b. Accident-Initiated Iodine Spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

The initial secondary coolant iodine activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm DE I-131.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the MSLB dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

Following a MSLB, the primary and secondary reactor coolant activity is released to the environment via two pathways.

<u>Faulted Steam Generator</u>

The release from the faulted SG occurs via the postulated break point of the main-steam line. The faulted SG is estimated to dry-out almost instantaneously following the MSLB (in ~10 seconds), releasing all of the iodine in the secondary coolant (at Technical Specification concentrations) that was initially contained in the steam generator. The EAB and LPZ dose to the public is calculated using an instantaneous release of the iodine inventory (Ci) in the SG liquid in the faulted SG. The secondary steam activity initially contained in the faulted steam generator is also released; however, the associated dose contribution is not included in this analysis since it is considered insignificant.

To maximize the control room and offsite doses following a MSLB, the maximum allowable primary to secondary SG tube leakage for all SGs (0.75 gpm or 1080 gpd at Standard Temperature and Pressure (STP) conditions), is conservatively assumed to occur in the faulted SG. All iodine and noble gas activities in the referenced tube leakage are released directly to the environment without hold-up or decontamination. The primary to secondary SG tube leakage is assumed to go on until the RCS reaches 212°F, which based on minimum heat transfer rates, is conservatively estimated to occur 30 hours after the event.

<u>Intact Steam Generators</u>

The initial iodine activities in the secondary coolant at Technical Specification levels are released to the environment in proportion to the steaming rate and the inverse of the partition coefficient (limited to 100) defined in RG 1.183. The noble gases are released freely to the environment without retention in the steam generators. However, there is no primary to secondary leakage into the intact SG as all primary to secondary leakage (1080 gpd or 0.75 gpm) is assumed to be occurring in the faulted SG.

The iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The condenser is assumed unavailable due to the loss of offsite power. The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the

EAB dose.

- The Source/Release for the Pre-incident Spike Case is at its maximum levels between 0 and 2 hours.
- The Source/Release for the Accident-Initiated Spike Case is at its maximum levels towards the end of the spiking period.

Regardless of the starting point of the "Worst 2-hr Window," the 0-2 hrs χ/Q is utilized.

The bounding EAB and LPZ dose following a MSLB at either unit for both scenarios are presented in Section 8.

Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical MSLB-specific assumptions associated with control room response and activity transport.

<u>Timing for Initiation of CRVS Mode 4</u>:

- An SI signal will be generated at t = 0.6 sec following a MSLB.
- The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The CR dampers are fully closed 10 secs later, or at t=38.8 secs (i.e., 0.6 + 28.2 + 10). The 2 second SI signal processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.
- In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=12.6 secs (i.e., 0.6 + 2 secs signal processing time + 10 sec damper closure time).

<u>Transport of radioactivity from the Break Location</u>

Since the normal operation (CRVS Mode 1) control room intake of the <u>faulted unit</u> is in such close proximity to the break point, an atmospheric dispersion factor (χ/Q) cannot be accurately determined. Thus, atmospheric dispersion is not credited when determining the control room operator dose from the secondary coolant discharge or the primary to secondary SG tube leakage released from the faulted SG via the break point.

Secondary Coolant Discharge: The radioactivity release due to the almost immediate dry-out of the faulted SG following a MSLB is based on a) the radioactivity concentration of the iodine in a finite cloud created by the secondary coolant liquid flash at the break point; b) conservation of total iodine activity in the SG liquid. The activity concentration at the release point is conservatively based on saturated steam at a density of 5.98E-04 gm/cm^3, (i.e., at 1 atmosphere and 212°F). The activity concentration entering the CR is assumed to be the same as the concentration at the

break point until the CR normal ventilation is isolated and the CRVS re-aligned to Mode 4 Pressurization.

Primary to Secondary SG Tube Leakage: Due to the close proximity of the normal operation CR intake of the faulted unit and MSL break release point and consequent unavailability of viable atmospheric dispersion factors, the primary to secondary SG tube leakage into the faulted SG is conservatively assumed to be piped directly into the control room. This model is reasonable since the relatively small plume of steam created by the ~0.485 gallon *{i.e., (0.75 gallon/min)(38.8 s) / 60 s/min}* of reactor coolant released due to SG tube leakage via the MSL break point could easily be swept into the CR due to the close proximity of the CR normal intake to the break point.

<u>Control Room Atmospheric Dispersion Factors</u>

As noted in Section 5.0, because of the proximity of the MSSVs/10% ADVs to the CR normal intake of the affected unit (~ 15 ft above the CR intake, horizontal distance is ~ 1.5 meters), and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation CR intake of the affected unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's CR normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a MSLB at either unit are provided in Table 7.6-2. The χ/Q values presented in Table 7.6-2 take into consideration the various release points-receptors applicable to the MSLB to identify the bounding χ/Q values applicable to a MSLB at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

The bounding Control Room dose following a MSLB at either unit is presented in Section 8.

7.7 Steam Generator Tube Rupture (SGTR)

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a SGTR.

This event is caused by the instantaneous rupture of a SG tube with a resultant release of primary coolant into the lower pressure secondary system. No melt or clad breach is postulated for the SGTR event. The calculation assumes a stuck-open 10% ADV of the ruptured steam generator for 30 minutes. Based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs / 10% ADVs of the intact steam generators are used to cool down the reactor until initiation of shutdown cooling. A portion of the primary coolant break flow in the ruptured SG flashes and is released a) to the condenser before reactor trip and b) directly to the environment after reactor trip, via the MSSVs and 10% ADVs. The remaining break flow mixes with the secondary side liquid, and is released to the environment via steam releases through MSSVs and 10% ADVs. The activity in the RCS also leaks into the intact

steam generators via SG tube leakage and is released to the environment from the MSSVs / 10% ADVs.

Regulatory guidance provided for the SGTR in pertinent sections of RG 1.183 including Appendix F is used to develop the dose consequence model. Table 7.7-1 lists the key assumptions / parameters utilized to develop the radiological consequences following a SGTR. Table 7.7-2 provides the time dependent steam flow from the ruptured and intact SGs and the flashed and unflashed break flow in the ruptured SG.

No melt or clad breach is postulated for the DCPP SGTR event. Thus, and in accordance with RG 1.183, Appendix F, item 2, the activity released is based on the maximum coolant activity allowed by the plant technical specifications. The plant technical specifications focus on the noble gases and iodines. In addition, and per RG 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike and b) an accident-initiated iodine spike.

a. Pre-accident Iodine Spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b. Accident-Initiated Iodine Spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 335 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

The initial secondary coolant iodine activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm DE I-131.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the SGTR dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd). To maximize the dose consequences, the analysis conservatively assumes that all of the 0.75 gpm SG tube leakage occurs in the intact SGs.

Following a SGTR, the primary and secondary reactor coolant activity is released to the environment via two pathways.

Ruptured Steam Generator

A SGTR will result in a large amount of primary coolant being released to the rutured steam generator via the break location with a significant portion of it flashed to the steam space.

In accordance with the guidance provided in RG 1.183, the noble gases in the entire break flow and the iodine in the flashed portion of the break flow are assumed to be immediately

available for release from the steam generator. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and the inverse of the allowable partition coefficient of 100. The iodine releases from the SGs are assumed to be 97% elemental and 3% organic.

Before the reactor trip at t=179 seconds, the radioactivity in the steam is released to the environment from the air ejector which discharges into the plant vent. All noble gases and organic iodines in the steam are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returns to both the intact steam generators and the ruptured steam generator and will be available for future steaming releases.

After the reactor trip, the radioactivity in the steam is released to the environment from the MSSVs/10% ADVs, due to the assumption of loss of offsite power (LOOP). To isolate the ruptured steam loop, the auxiliary feed water to the ruptured SG is secured. The calculation assumes the 10% ADV of the ruptured SG fails open for 30 minutes. The fail-open 10% ADV is isolated at t = 2653 seconds at which time the ruptured steam loop is isolated. The break flow continues until the primary system is in equilibrium with the secondary side of the ruptured SG at t = 5872 seconds. The iodines in the flashed break flow and the noble gases in the entire break flow is bottled up in the steam space of the ruptured SG and released to the environment during the manual depressurization of the ruptured SG after t = 2 hours.

<u>Intact Steam Generators</u>

The radioactivity released from the intact steam generators includes two components: (a) a portion of the break flow activity that is transferred to the intact steam generators via the condenser before reactor trip, and (b) due to SG tube leakage.

Approximately 75% (3 intact SGs vs 1 ruptured SG) of the flashed break flow activity that is transported and retained in the condenser before reactor trip will be transferred to the intact steam generators and released to the environment during the cool-down phase.

The total primary-to-secondary tube leak rate in the 3 intact SGs is conservatively assumed to be 0.75 gpm. The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant. Thus all leaked primary coolant iodine activities are assumed to mix uniformly with the steam generator liquid and are released in proportion to the steaming rate and the inverse of the partition coefficient. Before the reactor trip, the activity in the main steam is released from the plant vent via the air ejector/ condenser. After the reactor trip, the steam is released from the MSSVs/10% ADVs. The reactor coolant noble gases that enter the intact steam generator are released directly to the environment without holdup. The iodine releases from the SGs are assumed to be 97% elemental and 3% organic. The intact SG steam release continues until shutdown cooling (SDC) is initiated at t = 10.73 hours

Initial Secondary Coolant Activity Release

The initial iodine activities in the secondary coolant are released to the environment in proportion to the steaming rate and the inverse of the partition coefficient from the ruptured and intact SGs. Twenty five percent of the initial secondary coolant iodine inventory is in the ruptured SG and 75% of the initial secondary coolant iodine inventory is in the 3 intact SGs.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose.

For the SGTR, the EAB dose is controlled by the release of the flashed break flow in the ruptured SG which stops at 3402 seconds. The break flow stops at 5872 seconds and the ruptured SG is manually depressurized 2 hours after the accident. Therefore the maximum EAB dose occurs during the 0-2hr period for both the pre-accident and accident initiated iodine spike cases.

Regardless of the starting point of the "Worst 2-hr Window," the 0-2 hrs χ/Q is utilized.

The bounding EAB and LPZ dose following a SGTR at either unit for both scenarios are presented in Section 8.

Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical SGTR-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

- An SI signal will be generated at t = 219 sec following a SGTR.
- The CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to delays associated with diesel generator loading onto the 4kv buses. The CR dampers are fully closed 10 secs later, or at t=257.2 secs (i.e., 219 + 28.2 + 10). The 2 second SI signal processing time occurs in parallel with diesel generator sequencing and is therefore not included as part of the delay.
- In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=231 secs (i.e., 219 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors

As noted in Section 5.0, because of the proximity of the MSSVs/10% ADVs to the CR normal intake of the affected unit (~ 15 ft above the CR intake, horizontal distance is ~ 1.5 meters),

and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation CR intake of the affected unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's CR normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a SGTR at either unit are provided in Table 7.7-3. The χ/Q values presented in Table 7.7-3 take into consideration the various release points-receptors applicable to the SGTR to identify the bounding χ/Q values applicable to a SGTR at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

The bounding Control Room dose following a SGTR at either unit is presented in Section 8.

7.8 Loss of Load Event (LOL)

NRC sponsored computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOL event.

In accordance with DCPP current licensing basis documented in UFSAR Chapter 15.5.10, Condition II events that are expected to result in atmospheric steam releases are:

- Loss of electrical load and/or turbine trip (UFSAR 15.2.7)
- Loss of normal Feedwater (UFSAR 15.2.8)
- Loss of offsite power to the station auxiliaries (UFSAR 15.2.9)
- Accidental depressurization of the main steam system (UFSAR 15.2.14)

At DCPP, the mass of environmental steam releases for the Loss of Load Event bound all Condition II events and encompass the LRA and CREA.

SRP 15.2.1 to 15.2.5 (Reference 52) indicates that a Loss of Load event is different from the Loss of Alternating Current (AC) power condition (discussed in SRP 15.2.6, Reference 53), in that offsite AC power remains available to support station auxiliaries (e.g., reactor coolant pumps). The Loss of AC power condition results in the condenser being unavailable and reactor cooldown being achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling.

In keeping with the concept of developing steam releases that bound all Condition II events and encompass the LRA and CREA, the analysis performed to determine the mass of steam released following a Loss of Load event incorporates the assumption of Loss of offsite power to the station auxiliaries.

Neither RG 1.183 nor NUREG-0800 provides specific guidance with respect to scenarios to be assumed to determine radiological dose consequences from Condition II events. Thus the scenario outlined below for the bounding Condition II event that results in environmental releases is based on the conservative assumptions outlined in RG 1.183 for the MSLB.

Table 7.8-1 lists the key assumptions / parameters utilized to develop the radiological consequences following a LOL event. The conservative assumptions utilized to assess the dose consequences ensure that it represents the Limiting Condition II event.

As noted in DCPP UFSAR Section 15.2, no melt or clad breach is postulated for the DCPP LOL- Limiting Condition II event. Thus, and in accordance with RG 1.183, Appendix E, item 2, the activity released is based on the maximum coolant activity allowed by the plant technical specifications. The plant technical specifications focus on the noble gases and iodines. In addition, and per RG 1.183, two scenarios are addressed, i.e., a) a pre-accident iodine spike and b) an accident-initiated iodine spike.

a. Pre-accident Iodine Spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

b. Accident-Initiated Iodine Spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Technical Specification levels.

The initial secondary coolant iodine activity is the Technical Specification limit of 0.1 μCi/gm DE I-131.

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To accommodate any potential accident induced leakage, the LOL dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd).

The entire primary-to-secondary tube leakage of 0.75 gpm (maximum leak rate at STP conditions; total for all 4 SGs) is leaked into an effective SG. In accordance with RG 1.183, the pre-existing iodine activity in the secondary coolant and iodine activity due to reactor coolant leakage into the 4 SGs is assumed to be homogeneously mixed in the bulk secondary coolant. The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events) has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant. Therefore, per RG 1.183, the iodines are released to the environment via the MSSVs / 10% ADVs in proportion to the steaming rate and the inverse of a partition coefficient of 100. The iodine releases to the environment from the SG are assumed to be 97% elemental and 3% organic. The noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to a loss of offsite power coincident with reactor trip. Consequently, the radioactivity release resulting from a LOL-Limiting Condition II event is discharged to the environment from the steam generators via the MSSVs and the 10% ADVs.

The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated.

EAB 2 hr Worst Case Window

AST methodology requires that the worst case dose to an individual located at any point on the boundary at the EAB, for any 2-hr period following the onset of the accident be reported as the EAB dose. For the LOL-Limiting Condition II event, the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a LOL-Limiting Condition II event at either unit is presented in Section 8.

Accident Specific Control Room Model Assumptions

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 7.1. Provided below are the critical LOL event-specific assumptions associated with control room response and activity transport.

<u>Timing for Initiation of CRVS Mode 4 (if applicable)</u>:

The LOL-Limiting Condition II event does not initiate any signal which could automatically start the control room pressurization air ventilation. Thus the dose consequence analysis for the LOL-Limiting Condition II event assumes that the CR remains in normal operation mode.

<u>Control Room Atmospheric Dispersion Factors</u>

As noted in Section 5.0, because of the proximity of the MSSVs/10% ADVs to the CR normal intake of the affected unit (~ 15 ft above the CR intake, horizontal distance is ~ 1.5 meters), and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation CR intake of the affected unit (closest to the release point) will be insignificant. Therefore, only the unaffected unit's CR normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an LOL-Limiting Condition II event at either unit are provided in Table 7.8-2. The χ/Q values presented in Table 7.8-2 take into consideration the various release points-receptors applicable to the LOL to identify the bounding χ/Q values applicable to a LOL-Limiting Condition II event at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5 and summarized in the notes of Tables 5.2-2 and 5.2-3.

The bounding Control Room dose following a LOL-Limiting Condition II event at either unit is presented in Section 8.

7.9 Dose Consequences in the TSC due to Non-LOCA Events

Summarized herein is the assessment performed to estimate the inhalation and submersion doses in the TSC due to airborne releases following the non-LOCA events listed below:

1. FHA in the Fuel Handling Building or Containment (FHA)
2. Locked Rotor Accident (LRA)
3. Control Rod Ejection Accident (CREA)
4. Main Steam Line Break (MSLB)
5. Steam Generator Tube Rupture (SGTR)
6. Loss-of Load (LOL) Event

<u>TSC Non-LOCA Atmospheric Dispersion Factors</u>

Atmospheric dispersion factors (χ/Qs) from all of the Units 1 or 2 environmental release points associated with Non-LOCA events to TSC receptors (with the exception of the TSC pressurization intakes) are provided in Table 5.2-4.

Since the DCPP control room ventilation system (CRVS) pressurization intake also serves the TSC during CRVS Mode 4 operation, the CR Mode 4 χ/Qs applicable to the non-LOCA environmental release points listed above, are also applicable to the TSC. Similar to the CR dose model, credit is taken for a factor of 4 reduction in the CR/TSC pressurization intake χ/Qs during Mode 4 operation due to the availability of redundant safety related radiation monitors at each CR/TSC pressurization intake location, and the associated capability of initial selection of the less contaminated intake.

In the interest of model simplification, the Unit 1 & 2 MSL break locations are also used to represent the Unit 1 and Unit 2 MSSVs and 10% ADVs. This approach is acceptable since these release points are essentially co-located, but the MSL Break location releases have the lowest elevation and is therefore the closest to the TSC.

When the χ/Q values associated with the MSL break location to TSC receptors are applied to the MSSVs / 10% ADVs, they are reduced by a factor of 5 to address the high vertical velocity discharge for the first 10.73 hours of the accident. This is consistent with the approach used in the DCPP Control Room (CR) dose consequence analyses.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / TSC receptors applicable to the Non-LOCA events at either unit are provided in Table 7.9-1. The χ/Q values presented in Table 7.9-1 take into consideration the various release points-receptors applicable to the six non-LOCA accidents evaluated in this section to identify the bounding χ/Q values applicable to each event at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 5, and summarized above.

<u>Dose Consequences in the TSC due to Non-LOCA events</u>

A simplified approach is utilized to conservatively estimate the 30 day integrated inhalation and submersion doses in the TSC for each of the Non-LOCA events.

The analysis utilizes the LPZ dose model documented for each of the Non-LOCA events to predict the associated dose consequences at the "TSC location" by adjusting the X/Qs and the breathing rates. Specifically, the χ/Q values in the RADTRAD files that develop the LPZ doses were replaced with χ/Q values applicable to the center of the TSC roof (note: the EAB files were used for the FHA since environmental releases are terminated in 2 hours). In addition, the breathing rates used in the LPZ models were changed to 3.5E-04 m^3/s for the duration of the accident. This approach is excessively conservative since it is reflective of an operator located on the roof of the TSC, without taking credit for either the TSC ventilation systems or the TSC structure.

The above approach is successfully used to demonstrate that the TSC dose is less than that reported for the LOCA for all of the accidents listed above, with the exception of the MSLB, and the CREA (containment release scenario).

The 30-day integrated dose following the MSLB and the CREA (containment release scenario) is developed using the full activity transport model, inclusive of the TSC envelope and ventilation / filtration system.

It is concluded that the dose consequences in the TSC due to the FHA, the SGTR, the LRA, the LOL and the CREA (secondary release scenario) are bounded by the dose reported in the TSC for the LOCA; i.e. 4.1 rem TEDE. This conclusion is based on conservative simplistic evaluations that did not credit the TSC structure / ventilation system.

The dose consequences in the TSC following a MSLB and CREA (containment release scenario) are developed taking credit for the TSC structure / ventilation system, and are estimated to be 0.7 rem TEDE and 4 rem TEDE, respectively.

It is concluded that the dose consequences in the TSC due to airborne radioactivity releases following Non-LOCA events fall within the 4.1 rem TEDE dose reported for the LOCA.

Table 7.1-1
Control Room
Analysis Assumptions & Key Parameter Values

Parameter	Value
Free Volume	170,000 ft^3
Unfiltered Normal Operation Intake	Unit 1: 2100 cfm ± 10% Unit 2: 2100 cfm ± 10%
Emergency Pressurization Flow Rate	650 – 900 cfm
Maximum Unfiltered Backdraft Damper Leakage during CR Pressurization Operation	100 cfm
Carbon / HEPA Filter Flow during CR Pressurization Operation	1800 – 2200 cfm
Emergency Filtered Recirculation Rate	1250 cfm (minimum)
Pressurization Intake and Recirculation Carbon/HEPA Filter Efficiency (includes filter bypass)	93% (iodine) 98% (particulates)
Unfiltered Inleakage (Normal and Pressurization Mode)	70 cfm (maximum) Includes 10 cfm ingress / egress
Occupancy Factors	0-24 hr (1.0)
	1 - 4 d (0.6)
	4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.50E-04 m^3/sec)

Table 7.1-2
Technical Support Center
Analysis Assumptions & Key Parameter Values

Parameter	Value
Free Volume	51,250 ft^3
Filtered (HEPA only) Normal Operation Intake Flow Rate	500 cfm
Normal Intake HEPA Filter Efficiency (includes filter bypass)	98% (particulates)
Filtered (Carbon / HEPA) Pressurization Flow Rate	500 cfm
Flow through Carbon / HEPA Filter during Pressurization mode	1000 cfm
Filtered Recirculation flow rate during Pressurization mode	500 cfm (minimum)
Pressurization Intake and Recirculation Carbon/HEPA Filter Efficiency (includes filter bypass)	93% (iodine) 98% (particulates)
Unfiltered Inleakage	60 cfm (maximum) Includes 10 cfm ingress/egress
Occupancy Factors	0-24 hr (1.0)
	1 - 4 d (0.6)
	4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.50E-04 m^3/sec)

Table 7.2-1
Loss of Coolant Accident
Assumptions & Key Parameter Values

Parameter	Value
Core Power Level (105% of the rated power of 3411 MWth)	3580 MWt
Fuel Activity Release Fractions	Per Reg. Guide 1.183 (See Section 7.2.3.2.6)
Fuel Release Timing (gap)	Onset: 30 sec
	Duration: 0.5 hr
Fuel Release Timing (Early-In-Vessel)	Onset: 0.5 hr
	Duration: 1.3 hr
Core Activity	Table 4.1-1
Chemical Form of Iodine released from fuel to containment atmosphere	4.85% elemental 95% particulate 0.15% organic
Chemical Form of Iodine Released from RCS and sump water	97% elemental 3% organic
Containment Vacuum/Pressure Relief Parameters	
Minimum Containment Free Volume:	2.550E+06 ft^3
Primary Coolant Tech Spec Activity	Table 4.2-1
Chemical Form of Iodine Released	97% elemental; 3% organic
Maximum RCS flash fraction after LOCA Noble Gases Halogens	 100% 40%
Maximum containment pressure relief line air flow rate	218 actual cubic feet per second (acfs)
Maximum duration of release via containment pressure relief line	13 sec
Release Point	Plant Vent
Containment Leakage Parameters	
Containment Spray Coverage – Injection Spray and Recirculation Spray Modes: Sprayed Volume Unsprayed Volume	82.5% (sprayed fraction) 2.103E+06 ft^3 4.470E+05 ft^3
Minimum mixing flow rate from unsprayed to sprayed region: Before actuation of CFCUs After actuation of CFCUs	 2 unsprayed regions/hr 9.13 unsprayed regions/hr
Minimum duration of mixing via CFCUs	Start = 86 sec End = 30 days
Containment spray in injection mode Initiation time Termination time	 111 sec 3798 sec

Table 7.2-1
Loss of Coolant Accident
Assumptions & Key Parameter Values

Parameter	Value
Maximum delay between end of injection spray and initiation of recirculation spray	12 min (based on manual operator action)
Containment spray in recirculation mode Initiation time Termination time	 4518 sec 22,518 sec
Long-term Sump Water pH	≥ 7.5
Maximum allowable DF for fission product removal	Elemental Iodine: 200 Others: not applicable
Elemental iodine and particulate spray removal coefficients in sprayed region during both injection spray and recirculation spray modes	See Table 7.2-2
Elemental iodine removal coefficients due to wall deposition	See Table 7.2-2
Particulate removal coefficients in unsprayed region due to gravitational settling	See Table 7.2-2
Containment Leak rate (0-24 hr)	0.1% weight fraction per day
Containment Leak rate (1-30 day)	0.05% weight fraction per day
Containment Leakage Release Point (Unfiltered)	From the worst case release point of the following: Diffuse source via the containment wall Via Plant Vent Via Containment Pen Area GE Via Containment Pen Areas GW & FW
ESF System Environmental Leakage Parameters	
Minimum post-LOCA containment water volume sources	480,015 gal.
Minimum time after LOCA when recirculation is initiated	829 sec
Duration of leakage	30 days
Maximum ECCS fluid temperature after initiation of recirculation	259.9 °F
Maximum ECCS leak rate (including safety factor of 2)	Unfiltered via plant vent = 240 cc/min Unfiltered via Containment Penetration Areas GE or GW & FW = 12 cc/min
RHR pump seal failure	Filtered(1) via plant vent 50 gpm starting at t = 24 hrs for 30 min
Iodine Airborne Release Fraction	10%
Auxiliary Building ESF Ventilation System filter efficiency	Elemental iodine: 88% Organic iodine: 88%

Table 7.2-1 Loss of Coolant Accident Assumptions & Key Parameter Values	
Parameter	Value
Refueling Water Storage Tank (RWST) Back-Leakage Parameters	
Earliest initiation time of RWST back-leakage	829 sec
Maximum ECCS / sump water back-leakage rate to RWST (includes safety factor of 2)	2 gpm
RWST back-leakage iodine release fractions	See Table 7.2-3
RWST back-leakage noble gas, as iodine daughters, release rate from the RWST vent	See Table 7.2-3
Miscellaneous Equipment Drain Tank (MEDT) Leakage Parameters	
MEDT inflow rate (includes safety factor of 2)	1900 cc/min
MEDT leakage Iodine release fractions	See Table 7.2-4
MEDT leakage noble gas, as iodine daughters release rate from plant vent	See Table 7.2-4
CR Emergency Ventilation: Initiation Signal/Timing	
Initiation time (signal)	SI signal generated: 6 sec Non-Affected Unit NOP Intake Isolated: 18 sec Affected Unit NOP Intake Isolated and CRVS Mode 4 in full Operation: 44.2 sec
Bounding Control Room Atmospheric Dispersion Factors for LOCA	Table 7.2-5

Note:

(1) Releases from the RHR Pump Seal failure are filtered for CR dose evaluation and filtered for Site Boundary Dose Evaluation

<table>
<tr><th colspan="6">Table 7.2-2
Loss of Coolant Accident
Total Elemental Iodine & Particulate Removal Coefficients</th></tr>
<tr><th rowspan="2">From Time (sec)</th><th rowspan="2">To Time (sec)</th><th colspan="2">Elemental Iodine Removal Coefficient (hr^{-1}) – Note 1</th><th colspan="2">Particulate Removal Coefficient (hr^{-1})</th></tr>
<tr><th>Sprayed Region</th><th>Unsprayed Region</th><th>Sprayed Region</th><th>Unsprayed Region</th></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>0</td><td>30</td><td rowspan="2">2.74</td><td rowspan="2">2.74</td><td>N/A</td><td>N/A</td></tr>
<tr><td>30</td><td>111</td><td>5.89</td><td>0.0062</td></tr>
<tr><td>111</td><td>1,800</td><td rowspan="2">20.57 (Note 2)</td><td rowspan="10">0.00</td><td>2.24</td><td>0.0071</td></tr>
<tr><td>1,800</td><td>3,798</td><td>9.35</td><td>0.1144</td></tr>
<tr><td>3,798</td><td>4,518</td><td>0.00 (Note 3)</td><td>1.02</td><td>0.1229</td></tr>
<tr><td>4,518</td><td>5,030</td><td rowspan="6">19.91 (Note 2)</td><td>7.50</td><td>0.1239</td></tr>
<tr><td>5,030</td><td>6,480</td><td>6.40</td><td>0.1237</td></tr>
<tr><td>6,480</td><td>7,200</td><td>4.74</td><td>0.1236</td></tr>
<tr><td>7,200</td><td>8,004</td><td>3.39</td><td>0.1222</td></tr>
<tr><td>8,004</td><td>22,152</td><td rowspan="2">1.53</td><td>0.1040</td></tr>
<tr><td>22,152</td><td>22,518</td><td rowspan="2">0.00</td></tr>
<tr><td>22,518</td><td>720 hrs</td><td>0.00</td><td>0.00 (Note 4)</td></tr>
</table>

<u>Notes:</u>

1. Per RG 1.183 and SRP 6.5.2, removal credit for elemental iodine by sprays is eliminated after a DF=200 is reached in the containment atmosphere.
2. Wall deposition removal coefficient (0.57 hr^{-1}) is included.
3. Time period without spray.
4. For purposes of conservatism, no credit is taken for particulate removal in the sprayed region after termination of recirculation spray

Figure 7.2-1
Aerosol Removal Rates Within Sprayed Region

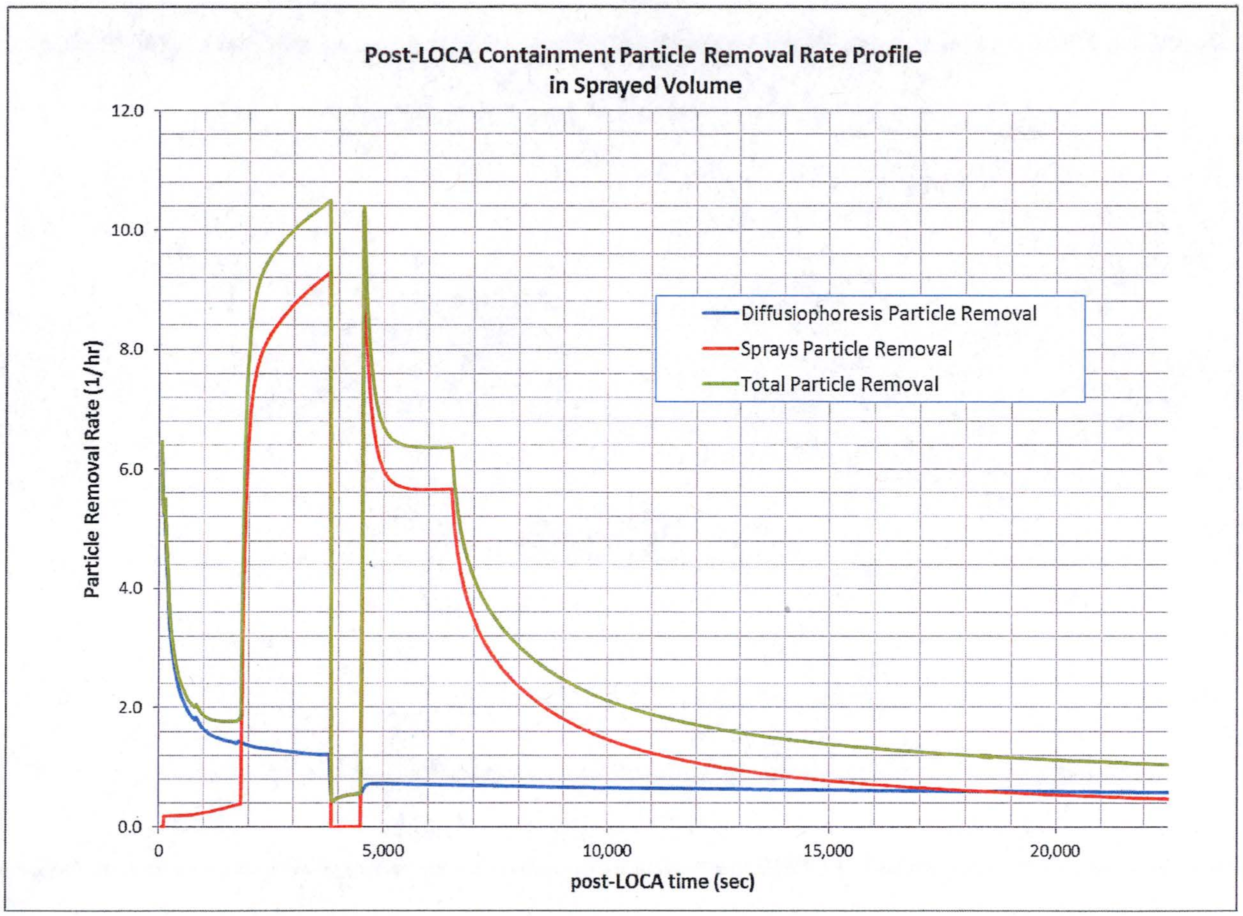


Figure 7.2-2
Aerosol Removal Rates Within Unsprayed Region

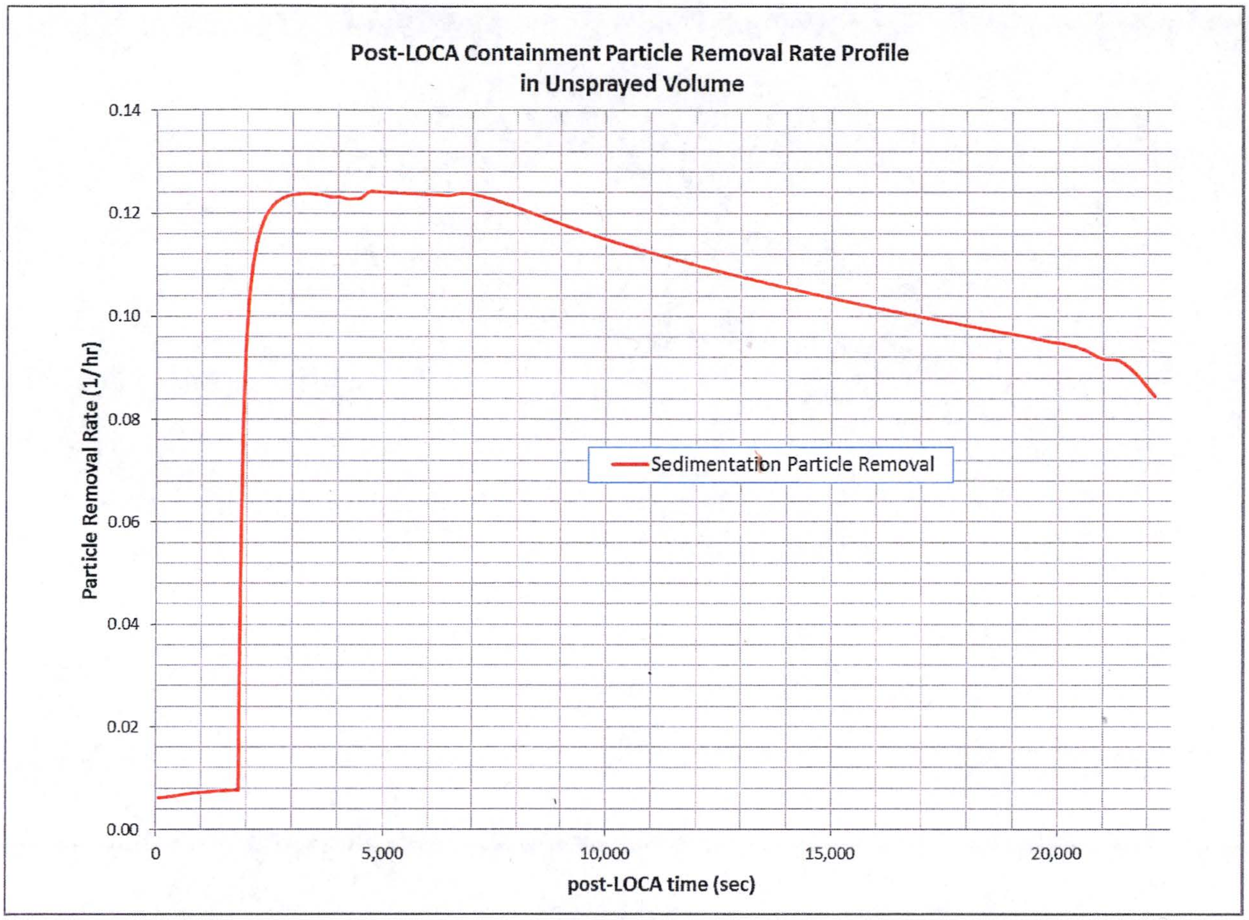


Table 7.2-2A
Containment Pressure/Temperature/ Relative Humidity Data – LOCA

Post-LOCA Time	Containment Pressure	Containment Temperature	Containment RH
Seconds	psia	°F	%
0.00	16.00	120	18
0.52	18.81	145.9	72.3
1.04	21.45	163.8	88.2
1.54	23.76	176.78	93.5
2.04	25.88	186.8	96
2.54	27.80	194.83	97.4
3.04	29.44	201.03	98.1
3.54	30.86	205.97	98.5
4.04	32.10	210.01	98.9
4.54	33.21	213.43	99.1
5.04	34.24	216.47	99.2
7.04	38.04	226.75	99.5
7.54	38.96	229.03	99.5
8.54	40.68	233.06	99.6
10.04	43.06	238.34	99.7
10.54	43.81	239.9	99.7
11.54	45.21	242.76	99.8
13.04	47.12	246.47	99.8
14.54	48.82	249.65	99.8
16.04	50.36	252.4	99.9
17.54	51.70	254.7	99.9
19.04	52.87	256.67	99.9
20.54	53.88	258.34	99.9
21.54	54.24	258.9	99.9
22.04	54.36	259.08	100
23.54	54.48	259.25	100
25.04	54.40	259.12	100
29.54	53.87	258.24	100
32.54	53.62	257.85	100
48.54	53.14	257.02	100
54.54	53.37	257.03	100
68.04	53.70	256.79	100
86.54	53.18	255.93	100
144.18	50.88	251.95	100
158.18	50.44	251.15	100

Table 7.2-2A
Containment Pressure/Temperature/ Relative Humidity Data – LOCA

Post-LOCA Time	Containment Pressure	Containment Temperature	Containment RH
Seconds	psia	°F	%
188.18	49.70	249.82	100
200.18	49.48	249.41	100
212.18	49.33	249.15	100
266.18	49.21	248.92	100
333.18	49.37	249.2	100
400.18	49.70	249.82	99.9
534.18	50.56	251.76	99.1
668.18	51.49	254.34	97.4
803.18	52.60	256.32	97.3
816.18	52.43	255.22	98.7
857.18	52.08	254.17	99.5
912.18	51.73	253.6	99.4
1021.19	51.21	252.76	99.3
1131.19	50.83	252.11	99.2
1240.19	50.55	251.61	99.1
1458.19	50.16	250.91	99
1677.19	49.94	250.49	98.9
1730.19	50.43	251.61	98.5
1746.19	50.26	250.56	99.9
1859.19	49.41	248.91	100
1988.19	48.57	247.32	100
2247.19	47.07	244.4	100
2505.19	45.75	241.75	100
2764.19	44.56	239.29	100
3022.19	43.47	236.96	99.9
3281.19	42.45	234.71	99.8
3604.24	41.26	231.94	99.9
3798.24	40.10	229.1	100
3888.29	40.52	230.99	98.2
3978.29	40.92	233.84	94.6
4068.29	41.24	235.73	92.5
4158.29	41.48	237.00	91.3
4338.29	41.86	238.43	90.3
4518.29	42.13	239.07	90.3
4536.29	42.05	237.81	92.2

Table 7.2-2A
Containment Pressure/Temperature/ Relative Humidity Data – LOCA

Post-LOCA Time	Containment Pressure	Containment Temperature	Containment RH
Seconds	psia	°F	%
4555.29	41.94	235.91	95.2
4573.29	41.87	234.69	97.2
4592.29	41.82	233.97	98.4
4666.29	41.74	233.23	99.5
5110.73	41.37	232.3	99.6
5700.73	40.79	230.94	99.6
6890.73	39.55	227.94	99.6
8080.73	38.36	224.96	99.5
10000.80	36.64	220.36	99.4
11001.50	35.82	218.08	99.4
12001.50	35.08	215.93	99.4
13001.50	34.39	213.85	99.4
14001.50	33.74	211.85	99.4
15001.50	33.13	209.91	99.4
16001.50	32.57	208.07	99.3
18001.50	31.59	204.69	99.3
20001.50	30.71	201.53	99.3
21001.50	30.31	200.06	99.3
22518.00	29.78	198.01	99.4

Table 7.2-2B Containment Steam Condensation Data – Loss of Coolant Accident						
Post-LOCA Time	Steam Condensation Rate					
	Thermal Conductor	Containment Fan Coolers	Injection Spray	Recirculation Spray	Total Steam Condensation Rate	
Seconds	lbm/sec	lbm/sec	lbm/sec	lbm/sec	lbm/sec	g/sec
0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.52	8.37	0.00	0.00	0.00	8.37	3796.57
1.04	49.71	0.00	0.00	0.00	49.71	22548.08
2.54	204.71	0.00	0.00	0.00	204.71	92854.90
3.04	250.47	0.00	0.00	0.00	250.47	113611.29
3.54	290.16	0.00	0.00	0.00	290.16	131614.37
4.04	325.28	0.00	0.00	0.00	325.28	147544.54
5.04	384.15	0.00	0.00	0.00	384.15	174247.52
7.54	509.97	0.00	0.00	0.00	509.97	231318.52
10.04	611.01	0.00	0.00	0.00	611.01	277149.49
12.54	684.86	0.00	0.00	0.00	684.86	310647.29
15.04	734.83	0.00	0.00	0.00	734.83	333313.30
17.54	766.16	0.00	0.00	0.00	766.16	347524.35
20.54	783.18	0.00	0.00	0.00	783.18	355244.50
24.54	742.58	0.00	0.00	0.00	742.58	336828.64
28.54	684.40	0.00	0.00	0.00	684.40	310438.64
32.54	644.28	0.00	0.00	0.00	644.28	292240.51
37.04	623.09	0.00	0.00	0.00	623.09	282628.89
53.54	538.96	0.00	0.00	0.00	538.96	244468.16
70.04	469.30	0.00	0.00	0.00	469.30	212870.91
87.04	412.48	0.00	0.00	0.00	412.48	187097.79
87.57	410.75	44.28	0.00	0.00	455.03	206398.15
88.07	409.15	45.29	0.00	0.00	454.44	206130.53
107.14	354.70	44.87	13.61	0.00	413.18	187415.31
124.18	312.19	44.28	72.76	0.00	429.23	194695.47
146.18	267.97	43.51	70.37	0.00	381.85	173204.26
169.18	231.50	42.78	68.46	0.00	342.74	155464.26
197.18	197.26	41.97	66.66	0.00	305.89	138749.38
234.18	166.95	41.32	49.25	0.00	257.52	116809.11
262.18	149.95	41.00	48.48	0.00	239.43	108603.63
327.18	121.98	40.50	47.28	0.00	209.76	95145.54
403.18	100.43	40.23	46.58	0.00	187.24	84930.64
449.18	91.02	40.17	46.66	0.00	177.85	80671.41
502.18	82.32	40.11	47.29	0.00	169.72	76983.70

Table 7.2-2B
Containment Steam Condensation Data – Loss of Coolant Accident

Post-LOCA Time	Steam Condensation Rate					
	Thermal Conductor	Containment Fan Coolers	Injection Spray	Recirculation Spray	Total Steam Condensation Rate	
Seconds	lbm/sec	lbm/sec	lbm/sec	lbm/sec	lbm/sec	g/sec
558.18	74.79	40.09	48.64	0.00	163.52	74171.43
560.18	74.55	40.08	48.66	0.00	163.29	74067.10
629.18	67.45	40.12	47.87	0.00	155.44	70506.40
683.18	63.43	40.22	47.23	0.00	150.88	68438.02
754.18	59.41	40.45	47.01	0.00	146.87	66619.12
802.18	57.28	40.64	46.99	0.00	144.91	65730.07
832.18	49.92	40.37	57.62	0.00	147.91	67090.85
876.18	43.40	40.00	58.41	0.00	141.81	64323.94
937.18	37.29	39.57	57.66	0.00	134.52	61017.25
1013.19	32.26	39.16	56.94	0.00	128.36	58223.12
1094.19	28.58	38.82	56.39	0.00	123.79	56150.20
1148.19	26.76	38.64	56.09	0.00	121.49	55106.94
1243.19	24.16	38.39	55.68	0.00	118.23	53628.23
1341.19	22.03	38.19	55.34	0.00	115.56	52417.14
1423.19	20.55	38.07	55.11	0.00	113.73	51587.06
1492.19	19.48	37.98	54.94	0.00	112.40	50983.79
1564.19	18.50	37.90	54.79	0.00	111.19	50434.94
1607.19	17.97	37.86	54.72	0.00	110.55	50144.64
1644.19	17.54	37.83	54.66	0.00	110.03	49908.77
1672.19	17.23	37.81	54.61	0.00	109.65	49736.41
1678.19	17.19	37.71	54.73	0.00	109.63	49727.33
1730.19	19.93	36.37	52.72	0.00	109.02	49450.64
1794.19	16.03	35.05	60.34	0.00	111.42	50539.27
1859.19	14.01	33.89	60.13	0.00	108.03	49001.59
1985.19	11.88	32.19	59.74	0.00	103.81	47087.43
2052.19	11.05	31.50	59.53	0.00	102.08	46302.71
2116.19	10.33	30.96	59.34	0.00	100.63	45645.00
2244.19	9.09	30.09	58.96	0.00	98.14	44515.56
2311.19	8.50	29.72	58.76	0.00	96.98	43989.39
2439.19	7.47	29.12	58.40	0.00	94.99	43086.74
2567.19	6.53	28.62	57.93	0.00	93.08	42220.38
2695.19	5.66	28.19	57.37	0.00	91.22	41376.70
2763.19	5.23	27.94	57.08	0.00	90.25	40936.71
2890.19	4.47	27.51	56.53	0.00	88.51	40147.46

Table 7.2-2B
Containment Steam Condensation Data – Loss of Coolant Accident

Post-LOCA Time	Steam Condensation Rate					
	Thermal Conductor	Containment Fan Coolers	Injection Spray	Recirculation Spray	Total Steam Condensation Rate	
Seconds	lbm/sec	lbm/sec	lbm/sec	lbm/sec	lbm/sec	g/sec
3018.19	3.76	27.11	56.00	0.00	86.87	39403.57
3082.19	3.43	26.92	55.74	0.00	86.09	39049.77
3210.19	2.80	26.56	55.23	0.00	84.59	38369.38
3338.19	2.21	26.22	54.74	0.00	83.17	37725.28
3466.19	1.64	25.89	54.25	0.00	81.78	37094.79
3594.19	1.12	25.58	53.78	0.00	80.48	36505.12
3722.24	0.24	25.02	54.13	0.00	79.39	36010.70
3796.24	0.02	24.72	53.61	0.00	78.35	35538.96
3843.29	2.24	24.91	0.00	0.00	27.15	12315.03
3901.29	4.39	25.13	0.00	0.00	29.52	13390.05
3995.29	6.72	25.42	0.00	0.00	32.14	14578.46
4105.29	8.33	25.70	0.00	0.00	34.03	15435.75
4189.29	9.21	25.87	0.00	0.00	35.08	15912.02
4291.29	10.03	26.06	0.00	0.00	36.09	16370.15
4383.29	10.63	26.21	0.00	0.00	36.84	16710.34
4463.29	11.06	26.33	0.00	0.00	37.39	16959.82
4515.29	11.25	26.41	0.00	0.00	37.66	17082.29
4518.29	11.26	26.41	0.00	0.92	38.59	17504.13
4584.29	10.17	26.48	0.00	8.26	44.91	20370.83
4592.29	10.15	26.49	0.00	9.02	45.66	20711.03
4654.29	10.18	26.55	0.00	11.12	47.85	21704.40
4698.29	10.19	26.59	0.00	11.33	48.11	21822.33
4734.29	10.21	26.60	0.00	11.39	48.20	21863.15
4785.29	10.19	26.62	0.00	11.43	48.24	21881.30
4807.29	10.17	26.63	0.00	11.44	48.24	21881.30
4843.29	10.14	26.63	0.00	11.46	48.23	21876.76
4851.29	10.13	26.64	0.00	11.46	48.23	21876.76
4895.29	10.09	26.64	0.00	11.48	48.21	21867.69
4926.29	10.05	26.64	0.00	11.49	48.18	21854.08
4932.29	10.06	26.63	0.00	11.49	48.18	21854.08
4988.29	9.99	26.63	0.00	11.51	48.13	21831.40
6120.73	8.70	25.98	0.00	11.43	46.11	20915.15
7400.73	7.52	25.02	0.00	11.13	43.67	19808.38
8680.73	6.51	24.04	0.00	10.86	41.41	18783.26

Table 7.2-2B
Containment Steam Condensation Data – Loss of Coolant Accident

Post-LOCA Time	Steam Condensation Rate					
	Thermal Conductor	Containment Fan Coolers	Injection Spray	Recirculation Spray	Total Steam Condensation Rate	
Seconds	lbm/sec	lbm/sec	lbm/sec	lbm/sec	lbm/sec	g/sec
9510.73	6.06	23.47	0.00	10.70	40.23	18248.02
14001.50	4.42	20.91	0.00	9.98	35.31	16016.35
18001.50	3.69	18.95	0.00	9.59	32.23	14619.28
22518.00	3.23	17.41	0.00	9.27	29.91	13566.95

Table 7.2-3
Loss of Coolant Accident
RWST Iodine Release Fraction and Gas Venting Rate to Atmosphere

From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Sec	Sec	Fraction $I_{released}$ / $I_{entering}$	Fraction V_{rwst} / day
829	7200	9.451E-05	2.610E+00
7200	28,800	6.357E-05	7.291E-01
28,800	86,400	8.796E-06	7.375E-02
86,400	345,600	4.560E-07	9.955E-03
345,600	471,600	6.347E-07	1.311E-02
471,600	1,011,600	8.231E-07	1.489E-02
1,011,600	2,048,400	1.114E-06	1.547E-02
2,048,400	2,592,000	1.483E-06	1.702E-02

Where:

$I_{released}$ = Total Iodine mass released to atmosphere during specified time interval, gm
$I_{entering}$ = Total Iodine mass entering to the RWST during specified time interval, gm
Frac. V_{rwst} = Rate of Fractional RWST gas volume vented during specified time interval

Table 7.2-4
Loss of Coolant Accident
MEDT Iodine Release Fraction and Gaseous Venting Rate to Atmosphere

From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Sec	Sec	Fraction $I_{released}$ / $I_{entering}$	Fraction V_{MEDT} / day
829	7,200	4.521E-07	5.024E+00
7,200	28,800	1.386E-08	3.024E-02
28,800	86,400	2.362E-07	3.324E-01
86,400	183,289	3.950E-07	6.497E+00
183,289	345,600	1.236E-02 (Note 2)	(Note 1)
345,600	752,400	2.028E-02 (Note 2)	(Note 1)
752,400	1,530,000	2.390E-02 (Note 2)	(Note 1)
1,530,000	2,592,000	2.166E-02 (Note 2)	(Note 1)

Where:

$I_{released}$ = Total Iodine mass released to atmosphere during specified time interval, gm
$I_{entering}$ = Total Iodine mass entering to the MEDT during specified time interval, gm
Frac V_{MEDT} = Rate of Fractional MEDT gas volume vented during specified time interval

Note 1: After the MEDT overflows at t = 183,289 sec, the gas venting rates are 2640 cfm from the EDRT room, and 1760 cfm from the U1/U2 Pipe Tunnels (i.e., the exhaust ventilation rate from the respective rooms + 10%). To be consistent with the methodology used to determine the iodine release fractions after spillover, the noble gases generated by decay of iodines in the tank and spilled liquid after overflow occurs, should also be released instantaneously to the environment without hold-up.

Note 2: The room ventilation flows addressed in Note 1 (utilized as clean in-coming air) are incorporated into the determination of the iodine equilibrium concentration in the EDRT room and U1/U2 Pipe Tunnels air space, respectively. The bounding iodine release fractions presented above after spillover assume instantaneous release of iodines to the environment without hold-up in the room.

Table 7.2-5
Loss of Coolant Accident
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
Control Room Normal Intakes					
Plant Vent Release					
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	9.08E-04	-----	-----	-----	-----
Containment Penetration Areas					
- Affected Unit Intake	6.60E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	2.08E-03	-----	-----	-----	-----
Control Room Infiltration					
Plant Vent	1.25E-03	9.08E-04	3.61E-04	3.65E-04	3.17E-04
Containment Penetration Areas	3.09E-03	1.83E-03	7.22E-04	7.13E-04	6.50E-04
RWST Vent	1.05E-03	5.55E-04	2.12E-04	2.12E-04	1.72E-04
Control Room Pressurization Intake					
Plant Vent	5.55E-05	3.68E-05	1.36E-05	1.38E-05	1.11E-05
Containment Penetration Areas	6.00E-05	3.98E-05	1.63E-05	1.37E-05	1.10E-05
RWST Vent	4.75E-05	3.23E-05	1.25E-05	1.14E-05	8.73E-06

Note 1: Release from the Containment penetration areas (i.e., areas GE or GW & FW): applicable to containment leakage and ESF system leakage that occurs in the Containment Penetration Area

Note 2: Release from Plant Vent: applicable to ESF system leakage that occurs in the Auxiliary building, MEDT releases, RHR Pump Seal Failure Release and Containment Vacuum/Pressure Relief Line Release

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit:

- Releases from the Plant Vent to the CR Normal intakes (affected and non-affected unit) are based on Unit 1 releases.
- Releases from the containment penetration areas to the CR Normal intakes (affected and non-affected unit) are based on Unit 2 GE area releases
- Releases from the Plant Vent to the CR Center (i.e., for CR Inleakage) are based on Unit 1 releases.
- Releases from the containment penetration areas to the CR Center are based on Unit 2 GE area releases for the 0-24 hour period and on the Unit 1 GW/FW area for the 1-30 day time period
- Releases from the RWST vent to the CR Center are based on Unit 2 releases.
- Releases from the Plant Vent to the CR pressurization intakes are based on Unit 1 releases to the U2 CR intake
- Releases from the containment penetration areas to the CR pressurization intakes are based on Unit 1 GW/FW area releases to the Unit 2 CR intake for the 0-2 hrs and 4-30 day time periods, from the Unit 2 GW/FW area releases to the Unit 1 CR intake for the 2-24 hrs time period and from the Unit 2 GE area releases to the Unit 1 CR intake for the 1-4 day time period
- Releases from the RWST vent to the CR pressurization intakes are based on Unit 2 releases to the Unit 1 CR intake.

Table 7.2-6
Loss of Coolant Accident
TSC Limiting Atmospheric Dispersion Factors (sec/m³)

Release Location / Receptor	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
TSC Normal Intakes					
Plant Vent Release	5.47E-04	-----	-----	-----	-----
Containment Penetration Areas	1.71E-03	-----	-----	-----	-----
RWST Vent	3.52E-04	-----	-----	-----	-----
TSC Infiltration					
Plant Vent	5.41E-04	2.09E-04	9.67E-05	7.95E-05	6.43E-05
Containment Penetration Areas	1.76E-03	7.16E-04	3.01E-04	2.84E-04	2.28E-04
RWST Vent	3.61E-04	1.48E-04	6.30E-05	5.80E-05	4.69E-05
CR/TSC Pressurization Intake					
Plant Vent	-----	3.68E-05	1.36E-05	1.38E-05	1.11E-05
Containment Penetration Areas	-----	3.98E-05	1.63E-05	1.37E-05	1.10E-05
RWST Vent	-----	2.93E-05	1.13E-05	1.08E-05	8.50E-06

Note 1: Release from the Containment penetration areas (i.e., areas GE or GW & FW): applicable to containment leakage and ESF system leakage that occurs in the Containment Penetration Area

Note 2: Release from Plant Vent: applicable to ESF system leakage that occurs in the Auxiliary building, MEDT releases, RHR Pump Seal Failure Release and Containment Vacuum/Pressure Relief Line Release

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Releases from the Plant Vent to the TSC Normal intake are based on Unit 2 releases.
- Releases from the containment penetration areas to the TSC Normal intake are based on Unit 2 GW/FW area releases
- Releases from the RWST vent to the TSC Normal intake are based on Unit 2 releases
- Releases from the Plant Vent to the TSC Center (i.e., for TSC Inleakage) are based on Unit 2 releases.
- Releases from the containment penetration areas to the TSC Center are based on Unit 2 GW/FW area releases
- Releases from the RWST vent to the TSC Center are based on Unit 2 releases.
- Releases from the Plant Vent to the CR/TSC pressurization intakes are based on Unit 1 releases to the U2 CR intake
- Releases from the containment penetration areas to the CR/TSC pressurization intakes are based on Unit 1 GW/FW area releases to the Unit 2 CR intake for the 0-2 hrs and 4-30 day time periods, from the Unit 2 GW/FW area releases to the Unit 1 CR intake for the 2-24 hrs time period and from the Unit 2 GE area releases to the Unit 1 CR intake for the 1-4 day time period
- Releases from the RWST vent to the CR/TSC pressurization intakes are based on Unit 2 releases to the Unit 1 CR intake.

Table 7.3-1
Fuel Handling Accident in Fuel Handling Building or Containment
Analysis Assumptions & Key Parameter Values

Parameter	Value
Power Level	3580 MWt
Number of Damaged Fuel Assemblies	1
Total Number of Fuel Assemblies	264
Decay Time Prior to Fuel Movement	72 hours
Radial Peaking Factor	1.65
Fraction of Core Inventory in gap	I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%)
Isotopic Inventory in Fuel Gap (Decayed 72 hours)	Table 7.3-2
Iodine form of gap release before scrubbing	99.85% elemental
	0.15% Organic
Iodine form of gap release after scrubbing	57% elemental
	43% Organic
Scrubbing Decontamination Factors	Iodine (200, effective)
	Noble Gas (1)
	Particulates (∞)
Rate of Release from Fuel	Puff
Environmental Release Rate	All airborne activity released within a 2 hour period (or less if the ventilation system promotes a faster release rate)
Environmental Release Points and Rates	
Accident in SFP in the FHB – Release flow rates	-Plant Vent – 46,000 cfm FHB Outleakage -Ingress/Egress locations – 30 cfm -Miscellaneous gaps/openings – 470 cfm
Minimum free volume in FHB above SFP	317,000 ft^3
Accident in Containment - Release flow rates	-Open Equipment Hatch – All airborne activity released in 2 hrs
Minimum Free Volume in Containment above Operating Floor	2,013,000 ft^3
CR Emergency Ventilation: Initiation Signal/Timing	
Signal(s) available to switch the CRVS from normal operation (NOP) Ventilation (Mode 1) to Pressurized Filtered Ventilation (Mode 4) following a FHA	Radiation signals from gamma sensitive intake monitors that initiate closure of the CR normal intake dampers and switch the CRVS from normal operation Ventilation Mode 1 to Pressurized Filtered Ventilation

Table 7.3-1
Fuel Handling Accident in Fuel Handling Building or Containment
Analysis Assumptions & Key Parameter Values

Parameter	Value
	Mode 4.
Radiation Monitor Analytical Safety Limit	1 mR/hr
Delay time for CRVS Mode 4 operation, including monitor response, signal processing, and damper closure time	32 seconds (see below)
Radiation Monitor Response Time	20 seconds (conservative assumption) - (Refer to Section 7.3)
Radiation monitor signal processing time	2 seconds
NOP Ventilation Damper Closure Time	10 seconds
Bounding Control Room Atmospheric Dispersion Factors for FHA	Table 7.3-3

Table 7.3-2
Isotopic Gap Activity – Fuel Handling Accident
Single Fuel Assembly (Decayed 72 hours)

Nuclide	Activity Per Assembly (Ci)	Gap Fraction	Gap Activity per Assembly (w/o Peaking Factor)
I-129	2.07E-02	0.05	1.04E-03
I-130	3.29E+02	0.05	1.65E+01
I-131	4.09E+05	0.08	3.27E+04
I-132	3.99E+05	0.23	9.18E+04
I-133	9.73E+04	0.05	4.87E+03
I-135	5.01E+02	0.05	2.51E+01
KR-83M	2.51E-04	0.04	1.00E-05
KR-85	5.75E+03	0.35	2.01E+03
KR-85M	1.77E+00	0.04	7.08E-02
KR-88	7.73E-03	0.04	3.09E-04
XE-127	9.64E-02	0.04	3.86E-03
XE-129M	5.28E+01	0.04	2.11E+00
XE-131M	6.96E+03	0.04	2.78E+02
XE-133	8.31E+05	0.04	3.32E+04
XE-133M	1.88E+04	0.04	7.52E+02
XE-135	1.07E+04	0.04	4.28E+02
XE-135M	8.18E+01	0.04	3.27E+00
CS-132	2.16E+01	0.46	9.94E+00
CS-134	1.25E+05	0.46	5.75E+04
CS-134M	1.04E-03	0.46	4.78E-04
CS-135	3.01E-01	0.46	1.38E-01
CS-136	3.10E+04	0.46	1.43E+04
CS-137	7.10E+04	0.46	3.27E+04
RB-86	1.16E+03	0.46	5.34E+02
RB-87	1.37E-05	0.46	6.30E-06
RB-88	8.63E-03	0.46	3.97E-03

Table 7.3-3
Fuel Handling Accident
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	0-22 sec	22 sec - 2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
Control Room Normal Intakes						
Containment Hatch Release						
- Affected Unit Intake	2.48E-02	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	2.67E-03	-----	-----	-----	-----	-----
Plant Vent Release						
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	9.08E-04	-----	-----	-----	-----	-----
FHB Out-leakage points						
- Affected Unit Intake	6.68E-03	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	2.69E-03	-----	-----	-----	-----	-----
Control Room Infiltration						
Containment Hatch Release	5.09E-03	5.09E-03	-----	-----	-----	-----
Plant Vent	1.25E-03	1.25E-03	-----	-----	-----	-----
FHB Out-leakage points	3.61E-03	3.61E-03	-----	-----	-----	-----
Control Room Pressurization Intake						
Containment Hatch Release	-----	6.15E-05	-----	-----	-----	-----
Plant Vent	-----	5.55E-05	-----	-----	-----	-----
FHB Out-leakage points	-----	6.13E-05	-----	-----	-----	-----

Note 1: Release from the Containment Hatch: applicable to FHA in Containment

Note 2: Release from Plant Vent / FHB Out-leakage: applicable to FHA in FHB

Note3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Releases from the Containment Hatch to the CR Normal intake of the affected and non-affected unit are based on Unit 2 and Unit 1 releases, respectively.
- Releases from the Plant Vent to the CR Normal intake of the affected and non-affected unit are based on Unit 1 releases
- Releases from the FHB to the CR Normal intake of the affected and non-affected unit are based on Unit 1 releases
- Releases from the Containment Hatch to the CR Center (i.e., for CR Inleakage) are based on Unit 2 releases.
- Releases from the Plant Vent to the CR Center are based on Unit 1 releases
- Releases from the FHB to the CR Center are based on Unit 2 releases.
- Releases from the Containment Hatch to the CR pressurization intakes are based on Unit 2 releases to the U1 CR intake
- Releases from the Plant Vent to the CR pressurization intakes are based on Unit 1 releases to the Unit 2 CR intake
- Releases from the FHB to the CR pressurization intakes are based on Unit 2 releases to the Unit 1 CR intake.

Table 7.4-1 Locked Rotor Accident Analysis Assumptions & Key Parameter Values	
Parameter	**Value**
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Primary to Secondary SG tube leakage	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3)
Melted Fuel Percentage	0%
Failed Fuel Percentage	10%
Equilibrium Core Activity	Table 4.1-1
Radial Peaking Factor	1.65
Fraction of Core Inventory in Fuel Gap	I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%)
Isotopic Inventory in Fuel Gap	Table 4.3-1
Iodine Chemical Form in Gap	4.85% elemental
	95% Particulate
	0.15% organic
Secondary Side Parameters	
Initial and Minimum SG Liquid Mass	92,301 lbm/SG
Iodine Species Released to Environment	97% elemental; 3% organic
Time period when tubes not totally submerged	insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs
Iodine Partition Coefficient in SGs	100
Particulate Carry-Over Fraction in SGs	0.0005 by weight
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of releases from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation : Initiation Signal/Timing	
	Control Room is assumed to remain on normal ventilation (CRVS Mode 1) for duration of the accident.
Control Room Atmospheric Dispersion Factors	Table 7.4-2

Table 7.4-2
Locked Rotor Accident
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release point and receptor	0-2hr	2-8 hr	8-10.73 hr
MSSVs/10% ADVs to CR NOP Intake (Note 1)	8.12E-04	5.32E-04	5.32E-04
MSSVs/10% ADVs to CR In-leakage (CR Centerline)	2.46E-03	1.59E-03	1.59E-03

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs to the normal operation CR 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 normal operation CR intake of the affected unit. Thus the χ/Q s presented reflect those applicable to the CR intake of the unaffected unit.

Note 2: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10%ADVs to the Unit 2 CR intake.
- Releases from the MSSVs/10% ADVs to the CR Center (i.e., for CR Inleakage) are based on Unit 1 10%ADV releases for the 0-2hrs time period, and Unit 2 10%ADV releases for the 2-10.73 hrs time period.

Table 7.5-1 Control Rod Ejection Accident Analysis Assumptions & Key Parameter Values	
Parameters	Value
Containment Leakage Pathway	
Power Level	3580 MWt
Free Volume	2.550E+06 ft^3
Containment leak rate (0 -24 hr)	0.1% vol. fraction per day
Containment leak rate(1-30 day)	0.05% vol. fraction per day
Failed Fuel Percentage	10%
Percentage of Core Inventory in Fuel Gap	10% (noble gases & halogens)
Melted Fuel Percentage	0%
Chemical Form of Iodine in Failed fuel	4.85% elemental 95% particulate 0.15% organic
Radial Peaking Factor	1.65
Core Activity Release Timing	Puff
Form of Failed Iodine in the Containment Atmosphere	97% elemental 3% organic
Equilibrium Core Activity	Table 4.1-1
Termination of Containment Release	30 days
Environmental Release Point	Same as LOCA Containment Leakage pathway
Secondary Side Pathway	
Reactor Coolant Mass	446,486 lbm
Primary-to-Secondary Leak rate	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3
Failed Fuel Percentage	Same as containment leakage pathway
Percentage of Core Inventory in Fuel Gap	Same as containment leakage pathway
Minimum Post-Accident SG Liquid Mass	92,301 lbm / SG
Iodine Species released to Environment	97% elemental 3% organic
Time period when tubes not totally submerged	Insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs.
Iodine Partition Coefficient in SGs	100

Table 7.5-1
Control Rod Ejection Accident
Analysis Assumptions & Key Parameter Values

Parameters	**Value**
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of Release from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation: Initiation Signal/Timing	
Initiation time (signal)	300 sec (SIS Generated) 312 sec (Non-Affected Unit NOP Intake fully Closed) 338.2 sec (Affected Unit NOP Intake fully Closed with full Mode 4 Emergency Ventilation Operation).
Control Room Atmospheric Dispersion Factors	Table 7.5-2

Table 7.5-2 Control Rod Ejection Accident Control Room Limiting Atmospheric Dispersion Factors (sec/m³)						
Release Location / Receptor	**0-2hr**	**2-8hr**	**8-10.73hr**	**10.73-24hr**	**24-96hr**	**96-720hr**
Control Room Normal Intakes						
Containment leakage						
- Affected Unit Intake	6.60E-03	-----		-----	-----	-----
- Non-Affected Unit Intake	2.08E-03	-----		-----	-----	-----
*MSSVs/*10% ADVs						
- Affected Unit Intake	Note 3	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	8.12E-04	-----	-----	-----	-----	-----
Control Room Infiltration						
Containment leakage	3.09E-03	1.83E-03	7.22E-04	7.22E-04	7.13E-04	6.50E-04
MSSVs/10% ADVs	2.46E-03	1.59E-03	1.59E-03	-----	-----	-----
Control Room Pressurization Intake						
Containment leakage	6.00E-05	3.98E-05	1.63E-05	1.63E-05	1.37E-05	1.10E-05
MSSVs/10% ADVs	1.40E-05	9.40E-06	9.40E-06	-----	-----	-----

Note 1: Containment leakage: Used for Containment release scenario; based on Containment penetration area release point.

Note 2: MSSV /10% ADVs: Used for Secondary System Release Scenario;

Note 3: Due to the proximity of the release from the MSSVs/10% ADVs to the normal operation CR 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 normal operation CR intake of the affected unit.

Note 4: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Releases from the containment penetration areas to the CR Normal intake of the affected and non-affected units are based on Unit 2 GE area releases
- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10%ADVs to the Unit 2 CR intake.
- Releases from the containment penetration areas to the CR Center are based on Unit 2 GE area releases for the 0-24 hour period and on the Unit 1 GW/FW area for the 1-30 day time period
- Releases from the MSSVs/10% ADVs to the CR Center (i.e., for CR Inleakage) are based on Unit 1 10%ADV releases for the 0-2hrs time period, and Unit 2 10%ADV releases for the 2-10.73 hrs time period.
- Releases from the containment penetration areas to the CR pressurization intakes are based on Unit 1 GW/FW area releases to the Unit 2 CR intake for the 0-2 hrs and 4-30 day time periods, from the Unit 2 GW/FW area releases to the Unit 1 CR intake for the 2-24 hrs time period and from the Unit 2 GE area releases to the Unit 1 CR intake for the 1-4 day time period
- Releases from the MSSVs/10% ADVs to the CR pressurization intakes are based on Unit 2 MSSV releases to the Unit 1 CR intake for the 0-2 hour time period and Unit 2 10%ADV releases to the Unit 1 CR intake for the 2-10.73 hour time period.

TABLE 7.6-1 Main Steam Line Break Analysis Assumptions & Key Parameter Values	
Parameter	**Value**
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Leak rate to Faulted Steam Generator	0.75 gpm (conservative assumption) ; leakage density 62.4 lbm/ft^3
Leak rate to Intact Steam Generators	0 gpm (all leakage assumed into faulted SG)
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine Conc.	Table 4.2-1 (1 µCi/gm DE I-131)
RCS Tech Spec Noble Gas Conc.	Table 4.2-1 (270 µCi/gm DE Xe-133)
RCS Equilibrium. Iodine Appearance Rates	Table 4.2-2 (1 µCi/gm DE I-131)
Pre-Accident Iodine Spike Concentrations	Table 4.2-2 (60 µCi/gm DE I-131)
Accident-Initiated Iodine Spike Appearance Rate	500 times equilibrium appearance rate
Duration of Accident- Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	Table 4.2-1 (0.1 µCi/gm DE I-131)
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Fraction of Iodine Released form Faulted SG	1.0 (Released to Environ without holdup)
Fraction of Noble Gas Released from Faulted SG	1.0 (Released to Environ without holdup)
Liquid mass in each SG	Faulted: 182,544 lbm (max.) Intact: 92,301 lbm (min. and initial)
Release Rate of SG liquid activity from Faulted SG	Dryout within10 seconds
Time period when tubes not totally submerged (intact SG)	Insignificant
Steam Releases from intact SGs	0-2 hrs: 384,000 lbm 2-8 hrs: 893,000 lbm 8-10.73 hrs: Same release rate as that for 2-8 hrs
Iodine Partition Coefficient in Intact SG	100 (SGs fully covered)
Termination of release (0.75 gpm leak): Faulted SG	30 hrs when RCS reaches 212 °F
Termination of release from Intact SG	10.73 hours
Release Point: Faulted SG	Outside containment, at the steam line break location
Release Point: Intact SG	MSSVs/10% ADVs

TABLE 7.6-1 Main Steam Line Break Analysis Assumptions & Key Parameter Values	
Parameter	Value
CR Emergency Ventilation : Initiation Signal/Timing	
Initiation (signal)	SIS
Unaffected Unit CRVS inlet damper fully closed	Within 12.6 seconds
Affected Unit CRVS inlet dampers fully closed	Within 38.8 seconds
Control Room Atmospheric Dispersion Factors	Table 7.6-2

Table 7.6-2
Main Steam Line Break Accident
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Receptor - Release Point	0-2hr	2-8 hr	8-10.73hr	10.73-30hr
CR NOP Intake - Faulted SG (Break Location)	Note 1			
CR NOP Intake - Intact SG (MSSVs/10% ADVs) - Note 2	8.12E-04			
CR Inleakage - Faulted SG (Break Location)	1.14E-02	7.22E-03	3.00E-03	3.00E-03
CR Inleakage - Intact SG (MSSVs/10% ADVs)	2.46E-03	1.59E-03	1.59E-03	-----
CR Emergency Intake & Bypass - Faulted SG (Break Location)	6.85E-05	4.70E-05	1.85E-05	1.85E-05
CR Emergency Intake & Bypass - Intact SG (MSSVs/10% ADVs)	1.40E-05	9.40E-06	9.40E-06	-----

Note 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).

Note 2: Due to the proximity of the release from the MSSVs/10% ADVs to the normal operation CR 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 normal operation CR intake of the affected unit. Thus the χ/Q s presented reflect those applicable to the CR intake of the unaffected unit.

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Releases from the MSSVs/10% ADVs of the Intact SG to the CR Normal intake of the non-affected unit are based on Unit 1 10%ADVs releases to the Unit 2 CR intake.
- Releases from the MSL break point of the faulted SG to the CR Center (i.e., for CR Inleakage) are based on Unit 1 for the 0-2 hour time period and Unit 2 releases for the 2- 30 hour time period .
- Releases from the MSSVs/10% ADVs of the intact SG to the CR Center are based on Unit 1 10%ADV releases for the 0-2hrs time period, and Unit 2 10%ADV releases for the 2-10.73 hrs time period.
- Releases from the MSL break point of the faulted SG to the CR pressurization intakes are based on Unit 1 releases to the Unit 2 CR intake for the 0-2 hour time period and Unit 2 releases to the Unit 1 CR intake for the 2-30 hour time period.
- Releases from the MSSVs/10% ADVs of the Intact SG to the CR pressurization intakes are based on Unit 2 MSSV releases to the Unit 1 CR intake for the 0-2 hour time period and Unit 2 10%ADV releases to the Unit 1 CR intake for the 2-10.73 hour time period.

Table 7.7-1
Steam Generator Tube Rupture
Analysis Assumptions & Key Parameter Values

Parameter	Value
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Time of Reactor Trip	179.0 sec
Time of isolation of stuck-open 10% ADV on the Ruptured SG	2653 sec
Termination of Break Flow from Ruptured SG that flashes	3402 sec
Termination of Break Flow from Ruptured SG	5872 sec
Time of manual depressurization of the Ruptured SG	2 hours
Break Flow to Ruptured Steam Generator that flashes	Table 7.7-2, Column "A"
Break Flow to Ruptured Steam Generator that does not flash	Table 7.7-2, Column "B"
Tube Leakage rate to Intact Steam Generators	0.75 gpm (total for all 4 SGs; conservatively assumed for 3 intact SGs); leakage density 62.4 lbm/ft^3
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine Concentration	1 µCi/gm DE I-131 (Table 4.2-1)
RCS Tech Spec Noble Gas Concentration	270 µCi/gm DE Xe-133 (Table 4.2-1)
RCS Equilibrium Iodine Appearance Rates	Table 4.2-2 (1 µCi/gm DE I-131)
Pre-Accident Iodine Spike Concentration	60 µCi/gm DE I-131 (Table 4.2-2)
Accident-Initiated Iodine Spike Appearance Rate	335 times TS equilibrium appearance rate
Duration of Accident-Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131 (Table 4.2-1)
Secondary System Release Parameters	
Initial SG liquid mass	89,707 lbm / SG
Iodine Species released to Environment	97% elemental; 3% organic
Steam flow rate to condenser from Ruptured SG before trip	63,000 lbm/min
Steam flow rate to condenser from intact SGs before trip	189,000 lbm/min
Partition Factor in Main Condenser	0.01 (elemental iodine)
	1 (organic iodine and noble gases)
Steam Releases from Ruptured SG	Table 7.7-2, Column "C"
Steam Releases from intact SG	Table 7.7-2, Column "D"
Post-accident minimum SG liquid mass for Ruptured SG	89,707 lbm
Post-accident minimum SG liquid mass for intact SGs	89,707 lbm per SG

Table 7.7-1 Steam Generator Tube Rupture Analysis Assumptions & Key Parameter Values	
Parameter	Value
Time period when tubes not totally submerged (intact SG)	insignificant
Fraction of Iodine Released (flashed portion)	1.0 (Released without holdup)
Fraction of Noble Gas Released from all SGs	1.0 (Released without holdup)
Iodine Partition Coefficient	100
Termination of Release from intact SG	10.73 hrs
Environmental Release Points	Plant Vent : 0 – 179 sec MSSVs/10% ADVs:179 sec – 10.73 hr
CR emergency Ventilation : Initiation Signal/Timing	
Initiation time (signal)	SIS: 219 sec Unaffected Unit inlet damper closed: 231 sec Affected Unit inlet damper closed: 257.2 sec
Control Room Atmospheric Dispersion Factors	Table 7.7-3

Table 7.7-2
Steam Generator Tube Rupture
Break Flows and Steam Releases

	Break Flow and Steam Release within each Time Interval			
	A	B	C	D
Time from Break (sec)	**Flashed Break Flow (lbm)**	**Un-flashed Break Flow (lbm)**	**Ruptured SG Steam Releases (lbm)**	**Intact SGs Steam Releases (lbm)**
0	1678	8422	187822	563100
179	2217	30003	10527	42565
853	12121	90754	113657	118
2653	1355	15906	0	146
2953	779	23177	0	85467
3402	0	45026	0	97164
4324	0	16870	0	9237
4739	0	23892	0	29103
5872	0	0	0	103300
7200	0	0	270000	1,342,400
38628	0	0	0	0

Note: Data in row for T=0 is applicable to time interval between T=0 sec to T=179 sec (typ)

TABLE 7.7-3 Steam Generator Tube Rupture Accident Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)					
Release Location / Receptor	**0-179 s**	**179-257.2 s**	**257.2 s- 2 h**	**2-8 hr**	**8-10.73 hr**
Control Room Normal Intakes					
- Plant Vent	1.29E-03	-----	-----	-----	-----
- MSSVs/10% ADVs (Note 1)	-----	8.12E-04	-----	-----	-----
Control Room Infiltration					
- Plant Vent	1.25E-03	-----	-----	-----	-----
- MSSVs/10% ADVs	-----	2.46E-03	2.46E-03	1.47E-03[2]	1.47E-03[2]
Control Room Pressurization Intake					
- MSSVs/10% ADVs	-----	-----	1.40E-05	9.40E-06[2]	9.40E-06[2]

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs to the normal operation CR 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 normal operation CR intake of the affected unit. Thus the χ/Q s presented reflect those applicable to the CR intake of the unaffected unit.

Note 2: Since the 0-2hour activity intake following a SGTR controls the 30-day integrated dose, the SGTR dose model utilizes a simplified model with respect to selection of the X/Q values for the 2-10.73hr time period. Specifically, the bounding X/Q value is selected for the release point / receptor for the 0-2 hr time period, but unlike the dose models used for the other accidents, the X/Q values for time periods beyond t=2hr, are not switched to the other unit if they display higher values.

Note 3: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit.

- Releases from the Plant Vent to the CR Normal intakes (occurs prior to reactor trip) are based on Unit 1 releases to both Unit 1 and Unit 2 CR normal intakes (i.e., an average X/Q; applied to the combined Unit 1 and Unit 2 CR normal intake flow)
- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10%ADVs releases to the Unit 2 CR intake.
- Releases from the Plant Vent to the CR Center (i.e., CR inleakage) are based on Unit 1 releases
- Releases from the MSSVs/10% ADVs to the CR Center are based on Unit 1 10%ADV releases. (Note that the X/Q value for the Unit 2 10% ADV to CR Center during the 2-10.73hr period is greater than the listed value. However, the dose consequences associated with the SGTR is dominated by the 0-2hour release, and the 0-2hr X/Q for Unit 1 is bounding.)
- Releases from the MSSVs/10% to the CR pressurization intakes are based on Unit 2 MSSV releases to the Unit 1 CR intake for the 0-2 hour time period and Unit 2 10%ADV releases to the Unit 1 CR intake for the 2-10.73 hour time period.

Table 7.8-1 Loss of Load Analysis Assumptions & Key Parameter Values	
Parameter	**Value**
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Primary to Secondary SG tube leakage	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3
Failed/Melted Fuel Percentage	0%
RCS Technical Specification Iodine Levels	Table 4.2-1 (1 µCi/gm DE I-131)
RCS Technical Specification Noble Gas Levels	Table 4.2-1 (270 µCi/gm DE Xe-133)
RCS Equilibrium Iodine Appearance Rates	Table 4.2-2 (1 µCi/gm DE I-131)
Pre-Accident Iodine Spike Concentration	Table 4.2-2 (60 µCi/gm DE I-131)
Accident-Initiated Iodine Spike Appearance Rate	500 times TS equilibrium appearance rate
Duration of Accident-Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131 (Table 4.2-1)
Initial and Minimum SG Liquid Mass	92,301 lbm/SG
Time period of tubes uncovered	insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs
Iodine Partition Coefficient in SGs	100
Iodine Species Released to Environment	97% elemental; 3% organic
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of releases from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation : Initiation Signal/Timing	
	Control Room is assumed to remain on normal ventilation for duration of the accident.
Control Room Atmospheric Dispersion Factors	Table 7.8-2

Table 7.8-2
Loss of Load Accident
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release point and receptor	0-2hr	2-8 hr	8-10.73 hr
MSSVs/10% ADVs to CR NOP Intake (Note 1)	8.12E-04	5.32E-04	5.32E-04
MSSVs/10% ADVs to CR Inleakage (CR Centerline)	2.46E-03	1.59E-03	1.59E-03

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs to the normal operation CR 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 normal operation CR intake of the affected unit. Thus the χ/Q s presented reflect those applicable to the CR intake of the unaffected unit.

Note 2: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Releases from the MSSVs/10% ADVs to the CR Normal intake of the non-affected unit are based on Unit 1 10%ADVs to the Unit 2 CR intake.
- Releases from the MSSVs/10% ADVs to the CR Center (i.e., for CR Inleakage) are based on Unit 1 10%ADV releases for the 0-2hrs time period, and Unit 2 10%ADV releases for the 2-10.73 hrs time period.

Table 7.9-1
Non-LOCA Events
Technical Support Center Limiting Atmospheric Dispersion Factors (sec/m^3)

Receptor - Release Point	0-2hr	2-8 hr	8-10.73hr	10.73-30hr	
MSLB					
TSC NOP Intake - Faulted SG (Break Location)	9.00E-04	-----	-----	-----	
TSC NOP Intake - Intact SG (MSSVs/10% ADVs)	1.80E-04	-----	-----	-----	
TSC Inleakage - Faulted SG (Break Location)	1.01E-03	4.62E-04	1.93E-04	1.93E-04	
TSC Inleakage - Intact SG (MSSVs/10% ADVs)	2.02E-04	9.24E-05	9.24E-05	-----	
CR/TSC Pressurization Intake - Faulted SG (Break Location)	-----	4.70E-05	1.85E-05	1.85E-05	
CR/TSC Pressurization Intake - Intact SG (MSSVs/10% ADVs)	-----	9.40E-06	9.40E-06	-----	
SGTR / LRA / LOL / CREA (Secondary Side Release Scenario)					
TSC Center of Roof – MSSVs/10% ADVs	2.02E-04	9.24E-05	9.24E-05	-----	
FHA					
TSC Center of Roof - Equipment Hatch	7.44E-04	-----	-----	-----	
Receptor - Release Point	**0-2hr**	**2-8 hr**	**8-24hr**	**1-4 days**	**4-30 days**
CREA (Containment Release Scenario)					
TSC NOP Intake – Containment Leakage	1.71E-03	-----	-----	-----	-----
TSC Inleakage - Containment Leakage	1.76E-03	7.16E-04	3.01E-04	2.84E-04	2.28E-04
CR/TSC Pressurization Intake – Containment Leakage	-----	3.98E-05	1.63E-05	1.37E-05	1.10E-05

Note1: The selection of the χ/Q values for the release points/ receptors listed above are intended to provide bounding dose estimates for an event at either unit

- Except as noted below for the CREA Containment leakage release point to the CR/TSC Pressurization Intakes, the χ/Q values for U2 release points are bounding for all TSC receptors (i.e., the TSC NOP Intake, the TSC Center of Roof (also used for TSC Inleakage) and the CR/TSC Pressurization Intakes). Releases from the containment penetration areas are based on the Unit 2 GW/FW area release point.
- Releases from the containment penetration areas to the CR/TSC pressurization intakes are based on the Unit 2 GW/FW area releases to the Unit 1 CR/TSC intake for the 2-24 hrs time period, from the Unit 2 GE area releases to the Unit 1 CR/TSC intake for the 1-4 day time period and from the Unit 1 GW/FW area releases to the Unit 2 CR/TSC intake for the 4-30 day time period.

Note 2: The χ/Q values presented above for MSSVs / 10% ADVs reflect a factor of 5 reduction to address the high vertical velocity discharge for the first 10.73 hours of the accident

Note 3: The χ/Q values presented above for the CR/TSC pressurization intake reflect a factor of 4 reduction to address the availability of redundant safety related radiation monitors at each CR/TSC pressurization intake location, and the associated capability of initial selection of the less contaminated intake

8.0 SUMMARY OF RESULTS: CONTROL ROOM / SITE BOUNDARY DOSES

The accidents listed below have been analyzed for dose consequences at the site boundary and control room.

1. Loss of Coolant Accident
2. Fuel Handling Accident in the Fuel Handling Building
3. Fuel Handling Accident in the Containment
4. Locked Rotor Accident
5. Control Rod Ejection Accident
6. Main Steam Line Break
7. Steam Generator Tube Rupture
8. Loss-of Load Event

In accordance with RG 1.183, the "worst 2-hour period" dose at the EAB, and the dose at the LPZ "for the duration of the release" is presented in Table 8.1-1. These dose values represent the post-accident dose to the public due to inhalation and submersion for each of these events. Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible for all the accidents. The associated regulatory limit as discussed in Section 2.4 is also presented.

Per regulatory guidance, the CR dose is integrated over 30 days. The calculated doses address the fact that for events with a duration less than 30 days, the CR dose needs to include the remnant radioactivity within the CR envelope after the event has terminated. The 30-day integrated dose to the control room operator, due to inhalation and submersion, is presented in Table 8.1-1 for all of the referenced design basis accidents. No credit is taken for use of personal protective equipment or prophylactic drugs.

The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The dose contribution due to direct shine from post LOCA contained sources/external cloud is identified and included in the CR doses reported for the LOCA in Table 8.1-1.

The dose received by the operator during transit outside the control room is not a measure of the "habitability" of the control room which is defined by the radiation protection provided to the operator by the control room shielding and ventilation system design. Thus, the estimated dose to the operator during routine post-LOCA access to the control room is addressed separately from the control room occupancy dose and is not included with the control room occupancy dose for the demonstration of control room habitability.

As demonstrated in Section 7.2.6, the dose contribution to the operator during routine access to control room for the duration of the LOCA is minimal (~ 1% of the occupancy dose).

In accordance with current licensing basis, the TSC design has been evaluated for the LOCA. The 30-day integrated dose to the TSC operator due to inhalation, submersion, and direct shine from the post LOCA contained sources/external cloud is estimated to be 4.1 rem TEDE (note: the dose contribution of direct shine to this total is ~1.3 rem TEDE). The dose consequences in the TSC due to airborne radioactivity releases following Non-LOCA events fall within the 4.1 rem TEDE dose reported for the LOCA (See Section 7.9 for detail).

Table 8.1-1
AST Site Boundary and Control Room Dose (TEDE, rem)

Accident	EAB (1) (4)	LPZ (2)	Regulatory Limit	Control Room	Regulatory Limit
LOCA	5.6	1	25	3.7 (0.7) (3)	5
Fuel Handling Accident in Fuel Handling Building	1.0	0.1	6.3	1.0	5
Fuel Handling Accident in Containment	1.0	0.1	6.3	4.3	5
Locked Rotor Accident	0.5	0.1	2.5	1.7	5
Control Rod Ejection Accident Containment Release Secondary Release	 0.7 0.7	 0.3 0.2	6.3	 3.4 0.5	5
Main Steam Line Break Pre-incident iodine Spike Accident-Initiated Iodine Spike	 0.1 0.7	 <0.1 0.2	 25 2.5	 2.0 4.1	5
Steam Generator Tube Rupture Pre-incident iodine Spike Accident-Initiated Iodine Spike	 1.3 0.7	 0.1 <0.1	 25 2.5	 0.6 0.3	5
Loss of Load Pre-incident iodine Spike Accident-Initiated Iodine Spike	 <0.1 <0.1	 <0.1 <0.1	 2.5 2.5	 <0.1 <0.1	5

Notes

(1) EAB doses are based on *worst 2-hour period* following onset of accident. Except as noted, the maximum 2-hr dose period for the EAB dose for each of the accidents is the 0 to 2 hrs time period.

- LOCA : 24-26 hrs (based on RHR Pump Seal Failure; see note 4 below for additional information)
- LRA: 8.73 to 10.73 hrs
- MSLB (accident initiated spike model): 7.6 to 9.6 hrs
- LOL (accident initiated spike model): 8.73 to 10.73 hrs.

(2) LPZ Doses are based on the duration of the release.

(3) The dose presented represents the operator dose due to occupancy. Value shown in parenthesis represents that portion of the total dose reported that is the contribution of direct shine from contained sources/external cloud. The dose to the CR operator during routine access for the 30 day duration of the accident is discussed in Section 7.2.6 and summarized in the text of Section 8.0.

(4) The maximum 2 hr EAB dose is based on the assumed RHR pump seal failure resulting in a 50 gpm leak of sump water occurring at t=24 hr for 30 mins. This release pathway is considered a part of DCPP licensing basis with respect to passive system failure. If this assumed release pathway were not included, the maximum 2 hr dose at the EAB would occur between t=0.5 hrs to t=2.5 hrs (i.e., during the post-LOCA ex-vessel release phase and would be 3.4 rem.

9.0 CONCLUSIONS

The Alternative Source Term as defined in Regulatory Guide 1.183 has been incorporated into the DCPP site boundary and control room dose re-analyses discussed herein. In accordance with current licensing basis, the dose to the Technical Support Center has been evaluated for the DBA that has the worst case radioactivity release, i.e., the LOCA. The estimated DCPP dose consequences for all design basis events meet the acceptance criteria specified in 10CFR50.67 and RG 1.183. This represents a full implementation of the Alternative Source Terms in which the RG 1.183 source term will become the licensing basis for DCPP.

10.0 REFERENCES

1. Code of Federal Regulations 10CFR50.67, "Accident Source Term".

2. NUREG-0800, Standard Review Plan 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Revision 0.

3. Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

4. NRC Safety Evaluation Report Related to License Amendment No. 163 to Facility Operating License No. DPR-80 and License Amendment No 165 to Facility Operating License No. DPR-82, PG&E, Diablo Canyon Power Station, Units 1 and 2, Docket Nos 50-275 and 50-323, dated February 27, 2004.

5. Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", Revision 1.

6. 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.

7. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", 1962.

8. Regulatory Guide 1.4, Revision 1, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors".

9. Code of Federal Regulations 10CFR50.44, "Combustible Gas Control for Nuclear Power Reactors".

10. NRC Safety Evaluation Report Related to License Amendment No. 168 to Facility Operating License No. DPR-80 and License Amendment No 169 to Facility Operating License No. DPR-82, PG&E, Diablo Canyon Power Station, Units 1 and 2, Docket Nos 50-275 and 50-323, dated May 4, 2004.

11. Code of Federal Regulations, 10CFR100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

12. Code of Federal Regulations, 10CFR50, Appendix A, GDC 19, "Control Room".

13. NUREG-0800, SRP 6.4, Revision 3, "Control Room Habitability System".

14. Code of Federal Regulations, 10CFR50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

15. NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

16. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants", February 1995.

17. SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," June 30, 1998.

18. NRC Safety Evaluation Report Related to License Amendment No. 201 to Facility Operating License no. DPR-40, OPPD, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated December 5, 2001.

19. Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion factors for Inhalation, Submersion, and Ingestion"

20. Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil"

21. Regulatory Guide 1.194, June 2003, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power plants.

22. NUREG 0800, Standard Review Plan 6.2.4, Revision 2, "Containment Isolation System".

23. Safety Guide 25, March 23, 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.

24. NUREG/CR 5009, Assessment of the Use of Extended Burn Fuel in LWRs, Jan 1988.

25. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors", Revision 1.

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

27. NUREG-0737, Supplement 1, Clarification of TMI Action Plan Requirements, January 1983.

28. Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants", Revision 1, March, 2007.

29. Bellinger, Thomas E., "The Impact of Nearby Structures and Trees on Sigma Theta Measurements", Illinois Department of Nuclear Safety, presented at the May 2002 NUMUG meeting, St. Charles, Illinois.

30. Call, Jennifer, "Evaluation of Obstruction Impacts on Wind Flow at Clinch River Nuclear Plant", February 2013.

31. ANSI/ANS 6.1.1-1977, "Neutron and Gamma-ray Flux-to-Dose-Rate Factors"

32. NUREG-0800, Standard Review Plan 15.2.8, Revision 2, "Feedwater System Pipe Break Inside and Outside Containment (PWR)".

33. NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays", June 1993.

34. Elia, Frank A. Jr. and Lischer, D. Jeffrey, Advanced Method for Calculating the Removal of Airborne Particles with Sprays, 1993, ASME paper no. 93-WA/SERA-5.

35. NUREG-0772, June 1981, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents."

36. Battelle Columbus Laboratories, BMI-2104, Vol. III, draft report, 1984, "Radionuclide Release Under Specific LWR Accident Conditions."

37. Walton, W. H., and Woolcock, A., 1960, "The Suppression of Airborne Dust by Water Spray," Interm. J. Air Pollution 3, 129-153.

38. Calvert, S., 1970, "Venturi and Other Atomizing Scrubbers Efficiency and Pressure Drop," AIChE Journal 16, 392-396.

39. Fuchs, N.A., 1964, "The Mechanics of Aerosols," revised and enlarged edition, Dover Publications, Inc.

40. Bunz, H., Kayro, M., Schöck, W., 1982, NAUA/MOD4 - A Code for Calculating Aerosol Behaviour in LWR Core Melt Accidents, Code Description and User Manual, KfK.

41. NRC Generic Letter No. 99-02, Laboratory Testing of Nuclear Grade Activated Charcoal, June 3, 1999.

42. NUREG-0800, 1988, Standard Review Plan, "Containment Spray as a Fission Product Cleanup System", Section 6.5.2, Revision 4.

43. Prentice-Hall W. M. Rohsenow and Harry Choi, "Heat, Mass and Momentum Transfer", 1961

44. NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents – Final Report," April 1992.

45. NUREG-0800, Standard Review Plan, Section 6.1.1, "Engineered Safety Features Materials", Revision 2

46. NUREG/CR-5950, "Iodine Evolution and pH Control", December 1992.

47. IE Bulletin No. 79-01B, Environmental Qualification of Class IE Equipment, January 14, 1980, including Enclosure 4, Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors.

48. Information Notice 91-56, September 19, 1991, "Potential Radioactive Leakage to Tank Vented to Atmosphere".

49. Joon Cho, Joseph Baron, Keith Ferguson, ""Modeling Radioactive Leakage from Atmospheric Tank Vents Following a LOCA", ANS Summer Conference in 2007, published in Transactions of the American Nuclear Society Volume 96, Radiation Protection and Shielding Session I, pg 441.

50. NUCON International Inc., "Control room Habitability Tracer Gas Leak Testing at Diablo Canyon Power Plant", December 2012.

51. NUREG-0800, Section 15.6.5, Appendix B, Revision 1, "Radiological Consequences of a Design Basis LOCA: Leakage from engineered Safety Feature Components outside Containment".

52. NUREG-0800, Standard Review Plan (SRP) Sections 15.2.1-15.2.5, "Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)", Revision 1.

53. NUREG-0800, SRP Section 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries", Revision 1.

54. NRC Safety Evaluation Report Related to License Amendment No. 139 to Facility Operating License No. DPR-80 and No. DPR-82, PG&E, Diablo Canyon Power Station, Units 1 and 2, Docket Nos 50-275 and 50-323, dated February 9, 2000.

55. NUREG-0933, "Resolution of Generic Safety Issues", Item 187, closed June 30, 2000

56. Code of Federal Regulations, 10CFR20.1003, "Definitions".

57. ANSI/ANS 6.1.1-1991, "Neutron and Gamma-ray Fluence-to-dose Factors".

58. NRC SER Related to License Amendment No. 8 and 6 to Facility Operating License No. DPR-80 and. DPR-82, PG&E, Diablo Canyon Power Station, Units 1 and 2, dated May 30, 1986.

59. Safety Guide 23, "Onsite Meteorological Programs", February 17, 1972

60. SCALE 4.3, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations And Personal Computers," Control Module SAS2 – CB&I S&W Inc. QA Category I Computer Code NU-230, V04, L03.

61. ACTIVITY2, "Fission Products in a Nuclear Reactor" – CB&I S&W Inc. Proprietary QA Category I Computer Code NU-014, V01, L03.

62. IONEXCHANGER, - CB&I S&W Inc. Proprietary QA Category I Computer Code NU-009, Ver. 01, Lev. 03.

63. EN-113, "Atmospheric Dispersion Factors" – CB&I S&W Inc. Proprietary QA Category I Computer Code EN-113, V06, L08.

64. ARCON96, "Atmospheric Relative Concentrations in Building Wakes" – CB&I S&W Inc. QA Category I Computer Code EN-292, V00, L00.

65. SWNAUA, "Aerosol Behavior in Condensing Atmosphere", CB&I S&W Inc. Proprietary QA Category I Computer Code NU-185, V02, L0.

66. RADTRAD 3.03 "A Simplified Model for RADionuclide Transport and Removal And Dose Estimates" – CB&I S&W Inc. QA Category I computer code No. NU-232, Version 3.03, Level (NA).

67. PERC2, "Passive Evolutionary Regulatory Consequence Code" – CB&I S&W Inc. Proprietary QA Category I Computer Code, NU-226, V00, L02.

68. SW-QADCGGP, "A Combinatorial Geometry Version of QAD-5A" – CB&I S&W Proprietary QA Category I Computer Code, NU-222, V00, L02.

69. GOTHIC, "Generation of Thermal-Hydraulic Information for Containments", CB&I S&W QA Category I computer code No. ME-376, Version 8.0, Lev (NA).

70. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms", March 7, 2006.

71. Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, October 2009.

72. NUREG/CR 2858, November 1982, PAVAN: An Atmospheric-Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations.

APPENDIX A: DCPP ARCON96 ATMOSPHERIC DISPERSION FACTOR INPUTS

Appendix A provides the DCPP Unit 1 and Unit 2 release point and receptor configuration information (e.g., height, velocity, distances, direction with respect to true north, etc.), release mode (e.g., ground, elevated, surface), and meteorological sensor configuration, used as input into ARCON96.

Also included as Figure A-1, is a site building layout and arrangement drawing that depicts the locations of the postulated release points and receptors. Figure A-1 shows plant north. The directions provided in the Unit 1 and Unit 2 release point and receptor configuration Tables presented herein account for the 23-degree azimuth clockwise offset of true north from plant north.

The on-site meteorological data input to ARCON96 (January 1, 2007 through December 31, 2011), in the ARCON96 input data format, is embedded below

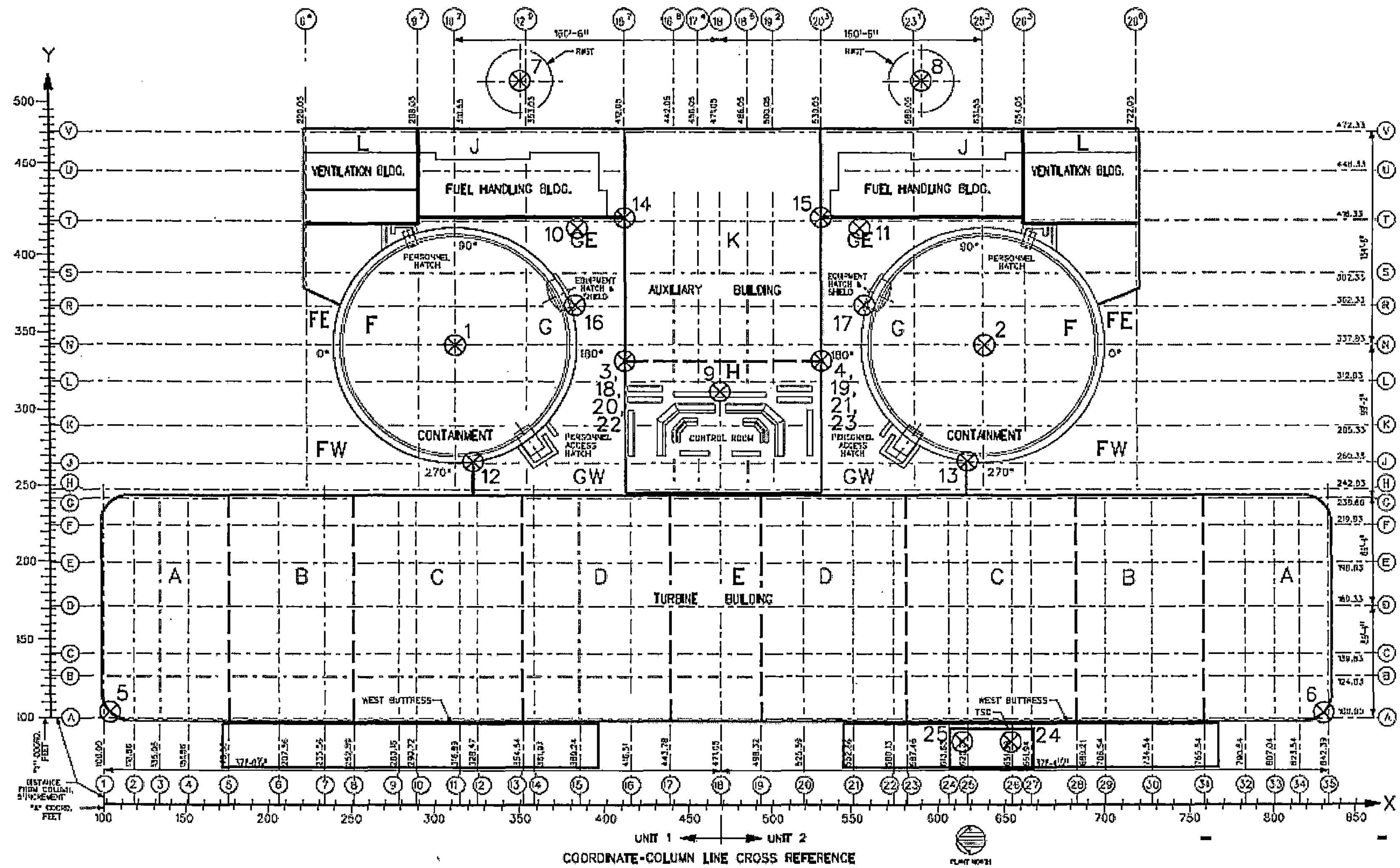
Attachment	DCPP ARCON96 Atmospheric Dispersion Factor Inputs
	On-site meteorological data in to ARCON96 (1/1/2007 – 12/31/2007)
	On-site meteorological data in to ARCON96 (1/1/2008 – 12/31/2008)
	On-site meteorological data in to ARCON96 (1/1/2009 – 12/31/2009)
	On-site meteorological data in to ARCON96 (1/1/2010 – 12/31/2010)
	On-site meteorological data in to ARCON96 (1/1/2011 – 12/31/2011)

Table A-1
On-Site Atmospheric Dispersion Factor Evaluation
Post-accident Release Point and Receptor Description / Location

ID (See Figure A-1)	Release/Receptor	Description
Note 1	Release Point	Unit 1 Containment Building (CB) edge
Note 1	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: Though not depicted in Figure A-1, 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.

Figure A-1 Diablo Canyon Power Plant – Site Layout and Arrangement
Post-Accident Environmental Release Point / Receptor Locations



Unit 1 LOCA Releases to CR Receptors

ARCON96 Parameter	Release Point / Receptor			
	U1 CB / U1 CR Normal Intake	**U1 CB / U2 CR Normal Intake**	**U1 CB / U1 CR Emergency Intake**	**U1 CB / U2 CR Emergency Intake**
Case No.	1	39	2	3
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - diffuse	ground - diffuse	ground - diffuse	ground - diffuse
Release Height (m)	22.0	22.0	32.0	32.0
Building Area (m^2)	2,744.5	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	9.4	44.6	71.8	151.4
Intake Height (m)	22.0	22.0	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	345	339	115	003
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	7.42, 11.07	7.42, 11.07	7.42, 11.07	7.42, 11.07

Unit 1 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 CB / CR Center	U1 PV / U1 CR Normal Intake	U1 PV / U2 CR Normal Intake	U1 PV / U1 CR Emergency Intake
Case No.	4	37	41	5
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - diffuse	ground - point	ground - point	ground - point
Release Height (m)	24.4	74.1	74.1	74.1
Building Area (m^2)	2,744.5	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	27.4	31.7	66.8	93.6
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	347	345	339	115
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	7.42, 11.07	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 PV / U2 CR Emergency Intake	U1 PV / CR Center	U1 RWST/ U1 CR Emergency Intake	U1 RWST / U2 CR Emergency Intake
Case No.	6	7	8	9
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	74.1	74.1	26.8	26.8
Building Area (m^2)	2,744.5	2,744.5	215.5	215.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	173.2	50.7	141.2	189.6
Intake Height (m)	32.0	24.4	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0[1]	0.0[1]
Direction to Source (degrees azimuth)	003	348	100	019
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Difference in local grade elevation between the RWST (115 ft) and the CR (85 ft) is accounted for in the release height such that source/receptor elevation difference is set to zero.

Unit 1 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 RWST / CR Center	U1 Cont. Pen GE/ U1 CR Normal Intake	U1 Cont. Pen GE / U2 CR Normal Intake	U1 Cont. Pen GE / U1 CR Emergency Intake
Case No.	10	21	43	22
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground -point	ground -point	ground -point	ground -point
Release Height (m)	26.8	16.8	16.8	16.8
Building Area (m^2)	215.5	530.4	530.4	2,744.5[2]
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	74.9	26.7	51.2	128.0
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0[1]	0.0	0.0	0.0
Direction to Source (degrees azimuth)	037	047	007	112
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Difference in local grade elevation between the RWST (115 ft) and the CR (85 ft) is accounted for in the release height such that source/receptor elevation difference is set to zero.
2. Release/receptor path goes around the Unit 1 CB.

Unit 1 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 Cont. Pen GE / U2 CR Emergency Intake	U1 Cont. Pen GE / CR Center	U1 PL GW/FW / Unit 1 CR Normal Intake	U1 PL GW/FW / U2 CR Normal Intake
Case No.	23	24	25	44
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground -point	ground -point	ground -point	ground -point
Release Height (m)	16.8	16.8	16.8	16.8
Building Area (m^2)	530.4	530.4	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	165.0	41.0	34.0	67.1
Intake Height (m)	32.0	24.4	22.0	22.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	012	027	300	319
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 PL GW/FW / U1 CR Emergency Intake	U1 PL GW/FW / U2 CR Emergency Intake	U1 PL GW/FW / CR Center	
Case No.	26	27	28	
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	
Lower Measurement Height (m)	10	10	10	
Upper Measurement Height (m)	76	76	76	
Wind Speed Units	m/sec	m/sec	m/sec	
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	
Source Information:				
Release Type	ground - point	ground - point	ground - point	
Release Height (m)	16.8	16.8	16.8	
Building Area (m^2)	2,744.5	2,744.5	2,744.5	
Vertical Velocity (m/sec)	0.0	0.0	0.0	
Stack Flow (m^3/sec)	0.0	0.0	0.0	
Stack Radius (m)	0.0	0.0	0.0	
Receptor Information:				
Distance to Receptor (m)	83.0	166.0	48.0	
Intake Height (m)	32.0	32.0	24.4	
Elevation Difference (m)	0.0	0.0	0.0	
Direction to Source (degrees azimuth)	124	354	317	
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	
Wind Direction Window (degrees azimuth)	90	90	90	
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	
Averaging Sector Width Constant	4.3	4.3	4.3	
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	

Unit 1 LOCA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U1 CB / TSC Normal Intake	U1 CB / TSC Center	U1 PV / TSC Normal Intake	U1 PV / TSC Center
Case No.	1	2	3	4
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - diffuse	ground - diffuse	ground - point	ground - point
Release Height (m)	11.0	10.7	74.1	74.1
Building Area (m^2)	2,744.5	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	109.5	97.4	131.7	119.7
Intake Height (m)	11.0	10.7	11.0	10.7
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	013	017	013	017
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	7.42, 11.07	7.42, 11.07	0.0, 0.0	0.0, 0.0

Unit 1 LOCA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U1 RWST/ TSC Normal Intake	U1 RWST/ TSC Center	U1 Cont. Pen GE / TSC Normal Intake	U1 Cont. Pen GE / TSC Center
Case No.	5	6	7	8
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	26.8	26.8	16.8	16.8
Building Area (m^2)	215.5	215.5	530.4	530.4
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	159.4	150.8	130.6	121.3
Intake Height (m)	11.0	10.7	11.0	10.7
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	030	035	027	032
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 LOCA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U1 PL GW/FW / TSC Normal Intake	**U1 PL GW/FW / TSC Center**		
Case No.	9	10		
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011		
Lower Measurement Height (m)	10	10		
Upper Measurement Height (m)	76	76		
Wind Speed Units	m/sec	m/sec		
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met		
Source Information:				
Release Type	ground - point	ground - point		
Release Height (m)	16.8	16.8		
Building Area (m^2)	2,744.5	2,744.5		
Vertical Velocity (m/sec)	0.0	0.0		
Stack Flow (m^3/sec)	0.0	0.0		
Stack Radius (m)	0.0	0.0		
Receptor Information:				
Distance to Receptor (m)	116.7	103.3		
Intake Height (m)	11.0	10.7		
Elevation Difference (m)	0.0	0.0		
Direction to Source (degrees azimuth)	004	008		
Default Information:				
Surface Roughness Length (m)	0.20	0.20		
Wind Direction Window (degrees azimuth)	90	90		
Minimum Wind Speed (m/sec)	0.5	0.5		
Averaging Sector Width Constant	4.3	4.3		
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0		

Unit 2 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 CB/ U1 CR Normal Intake	U2 CB/ U2 CR Normal Intake	U2 CB/ U1 CR Emergency Intake	U2 CB/ U2 CR Emergency Intake
Case No.	40	11	12	13
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - diffuse	ground - diffuse	ground - diffuse	ground - diffuse
Release Height (m)	22.0	22.0	32.0	32.0
Building Area (m^2)	2,744.5	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	44.9	7.8	150.6	71.0
Intake Height (m)	22.0	22.0	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	154	150	132	027
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	7.42, 11.07	7.42, 11.07	7.42, 11.07	7.42, 11.07

Unit 2 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 CB / CR Center	U2 PV / U1 CR Normal Intake	U2 PV/ U2 CR Normal Intake	U2 PV/ U1 CR Emergency Intake
Case No.	14	38	42	15
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - diffuse	ground - point	ground - point	ground - point
Release Height (m)	24.4	74.1	74.1	74.1
Building Area (m^2)	2,744.5	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	27.3	67.1	30.1	173.2
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	140	154	150	132
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	7.42, 11.07	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 PV/ U2 CR Emergency Intake	U2 PV/CR Center	U2 RWST/ U1 CR Emergency Intake	U2 RWST/ U2 CR Emergency Intake
Case No.	16	17	18	19
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	74.1	74.1	26.8	26.8
Building Area (m^2)	2,744.5	2,744.5	215.5	215.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	93.7	49.9	190.4	140.5
Intake Height (m)	32.0	24.4	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0[1]	0.0[1]
Direction to Source (degrees azimuth)	027	146	118	035
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Difference in local grade elevation between the RWST (115 ft) and the CR (85 ft) is accounted for in the release height such that source/receptor elevation difference is set to zero.

Unit 2 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 RWST/CR Center	U2 Cont. Pen GE / U1 CR Normal Intake	U2 Cont. Pen GE / U2 CR Normal Intake	U2 Cont. Pen GE / U1 CR Emergency Intake
Case No.	20	45	29	30
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground -point	ground -point	ground -point	ground -point
Release Height (m)	26.8	16.8	16.8	16.8
Building Area (m^2)	215.5	530.4	530.4	530.4
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	74.9	50.9	26.7	168.0
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0[1]	0.0	0.0	0.0
Direction to Source (degrees azimuth)	099	127	084	124
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Difference in local grade elevation between the RWST (115 ft) and the CR (85 ft) is accounted for in the release height such that source/receptor elevation difference is set to zero.

Unit 2 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 Cont. Pen GE / U2 CR Emergency Intake	**U2 Cont. Pen GE / CR Center**	**U2 PL GW/FW / U1 CR Normal Intake**	**U2 PL GW/FW / U2 CR Normal Intake**
Case No.	31	32	46	33
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground -point	ground -point	ground -point	ground -point
Release Height (m)	16.8	16.8	16.8	16.8
Building Area (m^2)	2,744.5[2]	530.4	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	125.5	41.0	67.1	34.0
Intake Height (m)	32.0	24.4	22.0	22.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	025	107	175	194
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Release/receptor path goes around the Unit 2 CB.

Unit 2 LOCA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 PL GW/FW / U1 CR Emergency Intake	U2 PL GW/FW / U2 CR Emergency Intake	U2 PL GW/FW / CR Center	
Case No.	34	35	36	
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	
Lower Measurement Height (m)	10	10	10	
Upper Measurement Height (m)	76	76	76	
Wind Speed Units	m/sec	m/sec	m/sec	
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	
Source Information:				
Release Type	ground - point	ground - point	ground - point	
Release Height (m)	16.8	16.8	16.8	
Building Area (m^2)	2,744.5	2,744.5	2,744.5	
Vertical Velocity (m/sec)	0.0	0.0	0.0	
Stack Flow (m^3/sec)	0.0	0.0	0.0	
Stack Radius (m)	0.0	0.0	0.0	
Receptor Information:				
Distance to Receptor (m)	165.5	83.0	48.0	
Intake Height (m)	32.0	32.0	24.4	
Elevation Difference (m)	0.0	0.0	0.0	
Direction to Source (degrees azimuth)	140	012	175	
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	
Wind Direction Window (degrees azimuth)	90	90	90	
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	
Averaging Sector Width Constant	4.3	4.3	4.3	
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	

Unit 2 LOCA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U2 CB/ TSC Normal Intake	U2 CB/ TSC Center	U2 PV/ TSC Normal Intake	U2 PV/ TSC Center
Case No.	11	12	13	14
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - diffuse	ground - diffuse	ground - point	ground - point
Release Height (m)	11.0	10.7	74.1	74.1
Building Area (m^2)	2,744.5	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	55.1	54.8	77.4	77.1
Intake Height (m)	11.0	10.7	11.0	10.7
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	060	072	060	072
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	7.42, 11.07	7.42, 11.07	0.0, 0.0	0.0, 0.0

Unit 2 LOCA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U2 RWST/ TSC Normal Intake	**U2 RWST/ TSC Center**	**U2 Cont. Pen GE / TSC Normal Intake**	**U2 Cont. Pen GE / TSC Center**
Case No.	15	16	17	18
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	26.8	26.8	16.8	16.8
Building Area (m^2)	215.5	215.5	530.4	530.4
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	129.4	127.8	104.4	100.8
Intake Height (m)	11.0	10.7	11.0	10.7
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	058	064	049	057
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 LOCA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U2 PL GW/FW / TSC Normal Intake	**U2 PL GW/FW / TSC Center**		
Case No.	19	20		
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011		
Lower Measurement Height (m)	10	10		
Upper Measurement Height (m)	76	76		
Wind Speed Units	m/sec	m/sec		
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met		
Source Information:				
Release Type	ground - point	ground - point		
Release Height (m)	16.8	16.8		
Building Area (m^2)	2,744.5	2,744.5		
Vertical Velocity (m/sec)	0.0	0.0		
Stack Flow (m^3/sec)	0.0	0.0		
Stack Radius (m)	0.0	0.0		
Receptor Information:				
Distance to Receptor (m)	54.5	53.1		
Intake Height (m)	11.0	10.7		
Elevation Difference (m)	0.0	0.0		
Direction to Source (degrees azimuth)	054	070		
Default Information:				
Surface Roughness Length (m)	0.20	0.20		
Wind Direction Window (degrees azimuth)	90	90		
Minimum Wind Speed (m/sec)	0.5	0.5		
Averaging Sector Width Constant	4.3	4.3		
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0		

Unit 1 FHA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 FHB/ U1 CR Normal Intake	U1 FHB/ U2 CR Normal Intake	U1 FHB/ U1 CR Emergency Intake	U1 FHB/ U2 CR Emergency Intake
Case No.	1	2	17	3
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	18.3	18.3	18.3	18.3
Building Area (m^2)	530.4	530.4	2,744.5[1]	530.4
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	26.5	44.7	135.1	165.0
Intake Height (m)	22.0	22.0	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	067	013	112	013
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Release/receptor path goes around the Unit 1 CB.

Unit 1 FHA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 FHB/CR Center	U1 EH/ U1 CR Normal Intake	U1 EH/ U2 CR Normal Intake	U1 EH/ U1 CR Emergency Intake
Case No.	4	5	6	19
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	18.3	20.1	20.1	20.1
Building Area (m^2)	530.4	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	37.5	14.0	46.8	116.7
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	038	023	349	114
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 FHA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 EH/ U2 CR Emergency Intake	U1 EH/CR Center		
Case No.	7	8		
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011		
Lower Measurement Height (m)	10	10		
Upper Measurement Height (m)	76	76		
Wind Speed Units	m/sec	m/sec		
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met		
Source Information:				
Release Type	ground - point	ground - point		
Release Height (m)	20.1	20.1		
Building Area (m^2)	2,744.5	2,744.5		
Vertical Velocity (m/sec)	0.0	0.0		
Stack Flow (m^3/sec)	0.0	0.0		
Stack Radius (m)	0.0	0.0		
Receptor Information:				
Distance to Receptor (m)	161.9	32.3		
Intake Height (m)	32.0	24.4		
Elevation Difference (m)	0.0	0.0		
Direction to Source (degrees azimuth)	006	008		
Default Information:				
Surface Roughness Length (m)	0.20	0.20		
Wind Direction Window (degrees azimuth)	90	90		
Minimum Wind Speed (m/sec)	0.5	0.5		
Averaging Sector Width Constant	4.3	4.3		
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0		

Unit 1 FHA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U1 FHB/ TSC Normal Intake	U1 FHB/ TSC Center	U1 EH/ TSC Normal Intake	U1 EH/ TSC Center
Case No.	25	26	29	30
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	18.3	18.3	20.1	20.1
Building Area (m^2)	530.4	530.4	530.4	530.4
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	126.2	117.6	120.3	109.9
Intake Height (m)	11.0	10.7	11.0	10.7
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	030	036	021	027
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 FHA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 FHB/ U1 CR Normal Intake	**U2 FHB/ U2 CR Normal Intake**	**U2 FHB/ U1 CR Emergency Intake**	**U2 FHB/ U2 CR Emergency Intake**
Case No.	9	10	11	18
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	18.3	18.3	18.3	18.3
Building Area (m^2)	530.4	530.4	530.4	2,744.5[1]
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	44.7	26.5	162.5	135.1
Intake Height (m)	22.0	22.0	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	121	067	121	022
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

1. Release/receptor path goes around the Unit 2 CB.

Unit 2 FHA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 FHB/CR Center	U2 EH/ U1 CR Normal Intake	U2 EH/ U2 CR Normal Intake	U2 EH/ U1 CR Emergency Intake
Case No.	12	13	14	15
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	18.3	20.1	20.1	20.1
Building Area (m^2)	530.4	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	37.5	46.5	13.8	161.6
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	097	145	110	128
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 FHA Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 EH/ U2 CR Emergency Intake	U2 EH/CR Center		
Case No.	20	16		
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011		
Lower Measurement Height (m)	10	10		
Upper Measurement Height (m)	76	76		
Wind Speed Units	m/sec	m/sec		
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met		
Source Information:				
Release Type	ground - point	ground - point		
Release Height (m)	20.1	20.1		
Building Area (m^2)	2,744.5	2,744.5		
Vertical Velocity (m/sec)	0.0	0.0		
Stack Flow (m^3/sec)	0.0	0.0		
Stack Radius (m)	0.0	0.0		
Receptor Information:				
Distance to Receptor (m)	116.9	32.0		
Intake Height (m)	32.0	24.4		
Elevation Difference (m)	0.0	0.0		
Direction to Source (degrees azimuth)	020	126		
Default Information:				
Surface Roughness Length (m)	0.20	0.20		
Wind Direction Window (degrees azimuth)	90	90		
Minimum Wind Speed (m/sec)	0.5	0.5		
Averaging Sector Width Constant	4.3	4.3		
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0		

Unit 2 FHA Releases to TSC Receptors

ARCON96 Parameter	Receptor			
	U2 FHB/ TSC Normal Intake	**U2 FHB/ TSC Center**	**U2 EH/ TSC Normal Intake**	**U2 EH/ TSC Center**
Case No.	27	28	31	32
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	18.3	18.3	20.1	20.1
Building Area (m^2)	530.4	530.4	530.4	530.4
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	108.3	103.6	89.6	85.5
Intake Height (m)	11.0	10.7	11.0	10.7
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	045	053	047	057
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 MSSV/ U1 CR Normal Intake	**U1 MSSV / U2 CR Normal Intake**	**U1 MSSV / U1 CR Emergency Intake**	**U1 MSSV / U2 CR Emergency Intake**
Case No.	1	2	25	3
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	27.1	27.1	27.1	27.1
Building Area (m^2)	0.0	2,744.5	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	1.5	37.5	116.6	149.8
Intake Height (m)	22.0	22.0	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	247	337	121	005
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 MSSV /CR Center	**U1 10% ADVs / Unit 1 CR Normal Intake**	**U1 10% ADVs / U2 CR Normal Intake**	**U1 10% ADVs / U1 CR Emergency Intake**
Case No.	4	9	10	27
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	27.1	26.5	26.5	26.5
Building Area (m^2)	2,744.5	0.0	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	20.5	1.5	37.5	116.6
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	355	247	337	121
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 10% ADVs / U2 CR Emergency Intake	**U1 10% ADVs /CR Center**	**U1 MSLB/ U1 CR Normal Intake**	**U1 MSLB/ U2 CR Normal Intake**
Case No.	11	12	17	18
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	26.5	26.5	17.7	17.7
Building Area (m^2)	2,744.5	2,744.5	0.0	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	149.8	20.5	1.5	37.5
Intake Height (m)	32.0	24.4	22.0	22.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	005	355	247	337
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 1 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U1 MSLB / U1 CR Emergency Intake	U1 MSLB / U2 CR Emergency Intake	U1 MSLB/CR Center	
Case No.	29	19	20	
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	
Lower Measurement Height (m)	10	10	10	
Upper Measurement Height (m)	76	76	76	
Wind Speed Units	m/sec	m/sec	m/sec	
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	
Source Information:				
Release Type	ground - point	ground - point	ground - point	
Release Height (m)	17.7	17.7	17.7	
Building Area (m^2)	2,744.5	2,744.5	2,744.5	
Vertical Velocity (m/sec)	0.0	0.0	0.0	
Stack Flow (m^3/sec)	0.0	0.0	0.0	
Stack Radius (m)	0.0	0.0	0.0	
Receptor Information:				
Distance to Receptor (m)	116.6	149.8	20.5	
Intake Height (m)	32.0	32.0	24.4	
Elevation Difference (m)	0.0	0.0	0.0	
Direction to Source (degrees azimuth)	121	005	355	
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	
Wind Direction Window (degrees azimuth)	90	90	90	
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	
Averaging Sector Width Constant	4.3	4.3	4.3	
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	

Unit 1 MSL Break Location Releases to TSC Receptors (Conservative for releases from MSSVs / 10% ADVs)

ARCON96 Parameter	Receptor			
	U1 MSL break / TSC Normal Intake	U1 MSL break / TSC Center		
Case No.	21	22		
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011		
Lower Measurement Height (m)	10	10		
Upper Measurement Height (m)	76	76		
Wind Speed Units	m/sec	m/sec		
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met		
Source Information:				
Release Type	ground - point	ground - point		
Release Height (m)	17.7	17.7		
Building Area (m^2)	2,744.5	2,744.5		
Vertical Velocity (m/sec)	0.0	0.0		
Stack Flow (m^3/sec)	0.0	0.0		
Stack Radius (m)	0.0	0.0		
Receptor Information:				
Distance to Receptor (m)	107.4	96.9		
Intake Height (m)	11.0	10.7		
Elevation Difference (m)	0.0	0.0		
Direction to Source (degrees azimuth)	021	027		
Default Information:				
Surface Roughness Length (m)	0.20	0.20		
Wind Direction Window (degrees azimuth)	90	90		
Minimum Wind Speed (m/sec)	0.5	0.5		
Averaging Sector Width Constant	4.3	4.3		
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0		

Unit 2 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 MSSV/ U1 CR Normal Intake	**U2 MSSV / U2 CR Normal Intake**	**U2 MSSV / U1 CR Emergency Intake**	**U2 MSSV / U2 CR Emergency Intake**
Case No.	5	6	7	26
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	27.1	27.1	27.1	27.1
Building Area (m^2)	2,744.5	0.0	2,744.5	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	37.5	1.5	149.8	116.6
Intake Height (m)	22.0	22.0	32.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	157	247	130	013
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 MSSV /CR Center	**U2 10% ADVs / U1 CR Normal Intake**	**U2 10% ADVs / U2 CR Normal Intake**	**U2 10% ADVs / U1 CR Emergency Intake**
Case No.	8	13	14	15
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	27.1	26.5	26.5	26.5
Building Area (m^2)	2,744.5	2,744.5	0.0	2,744.5
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	20.5	37.5	1.5	149.8
Intake Height (m)	24.4	22.0	22.0	32.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	139	157	247	130
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 10% ADVs / U2 CR Emergency Intake	U2 10% ADVs /CR Center	U2 MSLB/ U1 CR Normal Intake	U2 MSLB/ U2 CR Normal Intake
Case No.	28	16	21	22
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	2007 - 2011
Lower Measurement Height (m)	10	10	10	10
Upper Measurement Height (m)	76	76	76	76
Wind Speed Units	m/sec	m/sec	m/sec	m/sec
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met
Source Information:				
Release Type	ground - point	ground - point	ground - point	ground - point
Release Height (m)	26.5	26.5	17.7	17.7
Building Area (m^2)	2,744.5	2,744.5	2,744.5	0.0
Vertical Velocity (m/sec)	0.0	0.0	0.0	0.0
Stack Flow (m^3/sec)	0.0	0.0	0.0	0.0
Stack Radius (m)	0.0	0.0	0.0	0.0
Receptor Information:				
Distance to Receptor (m)	116.6	20.5	37.5	1.5
Intake Height (m)	32.0	24.4	22.0	22.0
Elevation Difference (m)	0.0	0.0	0.0	0.0
Direction to Source (degrees azimuth)	013	139	157	247
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	0.20
Wind Direction Window (degrees azimuth)	90	90	90	90
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	0.5
Averaging Sector Width Constant	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	0.0, 0.0

Unit 2 MSSVs / 10% ADVs / MSL Break Location Releases to CR Receptors

ARCON96 Parameter	Receptor			
	U2 MSLB / U1 CR Emergency Intake	**U2 MSLB / U2 CR Emergency Intake**	**U2 MSLB/CR Center**	
Case No.	23	30	24	
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011	2007 - 2011	
Lower Measurement Height (m)	10	10	10	
Upper Measurement Height (m)	76	76	76	
Wind Speed Units	m/sec	m/sec	m/sec	
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met	
Source Information:				
Release Type	ground - point	ground - point	ground - point	
Release Height (m)	17.7	17.7	17.7	
Building Area (m^2)	2,744.5	2,744.5	2,744.5	
Vertical Velocity (m/sec)	0.0	0.0	0.0	
Stack Flow (m^3/sec)	0.0	0.0	0.0	
Stack Radius (m)	0.0	0.0	0.0	
Receptor Information:				
Distance to Receptor (m)	149.8	116.6	20.5	
Intake Height (m)	32.0	32.0	24.4	
Elevation Difference (m)	0.0	0.0	0.0	
Direction to Source (degrees azimuth)	130	013	139	
Default Information:				
Surface Roughness Length (m)	0.20	0.20	0.20	
Wind Direction Window (degrees azimuth)	90	90	90	
Minimum Wind Speed (m/sec)	0.5	0.5	0.5	
Averaging Sector Width Constant	4.3	4.3	4.3	
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0	0.0, 0.0	

Unit 2 MSL Break Location Releases to TSC Receptors (Conservative for releases from MSSVs / 10% ADVs)

ARCON96 Parameter	Receptor			
	U2 MSL break / TSC Normal Intake	U2 MSL break / TSC Center		
Case No.	23	24		
Meteorological Information:				
Period of Meteorological Data	2007 - 2011	2007 - 2011		
Lower Measurement Height (m)	10	10		
Upper Measurement Height (m)	76	76		
Wind Speed Units	m/sec	m/sec		
Meteorological Data File Names:	dcpprg07.met – dcpprg11.met	dcpprg07.met – dcpprg11.met		
Source Information:				
Release Type	ground - point	ground - point		
Release Height (m)	17.7	17.7		
Building Area (m^2)	0.0	0.0		
Vertical Velocity (m/sec)	0.0	0.0		
Stack Flow (m^3/sec)	0.0	0.0		
Stack Radius (m)	0.0	0.0		
Receptor Information:				
Distance to Receptor (m)	83.6	77.7		
Intake Height (m)	11.0	10.7		
Elevation Difference (m)	0.0	0.0		
Direction to Source (degrees azimuth)	039	049		
Default Information:				
Surface Roughness Length (m)	0.20	0.20		
Wind Direction Window (degrees azimuth)	90	90		
Minimum Wind Speed (m/sec)	0.5	0.5		
Averaging Sector Width Constant	4.3	4.3		
Initial Diffusion Coefficients (m)	0.0, 0.0	0.0, 0.0		

APPENDIX B: CHANGES TO KEY DESIGN INPUT VALUES (BY ACCIDENT): CLB VS AST

As noted in Section 1.0, with this application, and in the interest of evaluating DCPP design against a more realistic accident sequence, as well as in gaining dose analysis margin, the methodology / scenarios used in the following design basis accident (DBA) analyses discussed in the DCPP UFSAR are being updated to reflect the AST guidance provided in RG 1.183.

1. Loss of Coolant Accident (LOCA)
2. FHA in the Containment or in the Fuel Handling Building (FHA)
3. Locked Rotor Accident (LRA)
4. Control Rod Ejection Accident (CREA)
5. Main Steam Line Break (MSLB)
6. Steam Generator Tube Rupture (SGTR)
7. Loss-of Load (LOL) Event

Appendix B provides a comparison between the design input values used in the current licensing basis (CLB) dose consequence analyses supporting DCPP Units 1 and 2, to those utilized in the AST analyses supporting this application. The basis for the change from the CLB value is also included, as applicable. It is noted that the DCPP CLB assesses Control Room habitability for the LOCA, MSLB, SGTR and FHA. The methodology used to assess the CLB analyses supporting the CREA, LRA and LOL event are DCPP-specific with pre-NUREG-0800 assumptions. In addition, the CLB analyses for the CREA, LRA and LOL only address offsite dose consequences.

The AST vs CLB information is provided by accident and in tabular format. The control room parameter values and the offsite atmospheric dispersion factors, both of which are generally applicable to all accidents, are summarized separately.

Table No.	Subject
Table B.1-1	Control Room (CR)
Table B.1-2	Site Boundary Atmospheric Dispersion Factors (χ/Q)
Table B.2-1	Loss of Coolant Accident (LOCA)
Table B.2-2A	Limiting CR χ/Q for LOCA – AST Values
Table B.2-2B	CR χ/Q's for LOCA – CLB Values
Table B.3-1	Fuel Handling Accident (FHA)
Table B.3-2A	Limiting CR χ/Q's for FHA – AST Values
Table B.3-2B	CR χ/Q's for FHA – CLB Values
Table B.4-1	Locked Rotor Accident (LRA)
Table B.5-1	Control Rod Ejection Accident (CREA)
Table B.6-1	Main Steam Line Break (MSLB)
Table B.6-2A	Limiting CR χ/Q's for MSLB – AST Values
Table B.6-2B	CR χ/Q's for MSLB – CLB Values
Table B.7-1	Steam Generator Tube Rupture (SGTR)
Table B.7-2A	Limiting CR χ/Q's for SGTR – AST Values
Table B.7-2B	CR χ/Q's for SGTR – CLB Values
Table B.8-1	Loss of Load (LOL) Event

Table B.1-1 Control Room[1]
Changes to Key Input Parameter Values: AST vs CLB

Parameter	AST Value	CLB Value	Remark
Free Volume	170,000 ft^3		
Unfiltered Normal Operation Intake	Unit 1: 2100 cfm ± 10% Unit 2: 2100 cfm ± 10%	4200 cfm total (SGTR) 2100 cfm (LOCA / MSLB / FHA)	AST: addresses flow uncertainty
Emergency Pressurization Flow Rate	650 – 900 cfm	2100 cfm	AST: Based on results of 2012 Tracer Gas Test
Maximum Unfiltered Backdraft Damper Leakage during CR Pressurization Operation	100 cfm	None	AST: Based on results of 2012 Tracer Gas Test CLB: Analysis predates modification
Carbon / HEPA Filter Flow during CR Pressurization Operation	1800 – 2200 cfm	2100 cfm	AST: Based on results of 2012 Tracer Gas Test
Emergency Filtered Recirculation Rate	1250 cfm (minimum)	2100 cfm	AST: Based on results of 2012 Tracer Gas Test
Pressurization Intake and Recirculation Carbon/HEPA Filter Efficiency (includes filter bypass)	93% (elemental and organic iodine) 98% (particulates)	95% (particulates & all iodine)	AST: Based on test acceptance criteria (see Section 7.1)
Unfiltered Inleakage	70 cfm (maximum)	10 cfm	AST: Includes 10 cfm for ingress / egress; Conservative value based on results of 2012 Tracer Gas Test of 37 cfm.
Occupancy Factors	0-24 hr (1.0) 1 - 4 d (0.6) 4-30 d (0.4)		
Operator Breathing Rate	0-30 d (3.5E-04 m^3/sec)	0-30 d (3.47E-04 m^3/sec)	
Inhalation Dose Conversion Factors	Federal Guidance Report No.11	Various: ICRP-30 (FGR 11), RG 1.109, ICRP 2	
Computer Codes used for CR Dose Calculations	RADTRAD 3.03, PERC2	RADTRAD 3.03 - SGTR LOCADOSE - LOCA / FHA / MSLB	AST: For full list of codes used see Section 3.0. CLB - the site boundary dose for other accidents was developed using EMERALD

Note 1:
CLB: Control Room dose only assessed for LOCA / MSLB / SGTR / FHA; the FHA dose consequence analysis does not take credit for CRVS Mode 4 operation
AST: Control Room dose estimated for all analyzed accidents; i.e., LOCA / MSLB / SGTR / FHA / LRA / CREA / LOL event

Table B.1-2
Site Boundary Atmospheric Dispersion Factors (χ/Q)
Changes to Key Input Parameter Values: AST vs CLB

	χ/Q (sec/m^3)				
Receptor	**0 - 2 hours**	**2 - 8 hours**	**8 - 24 hours**	**1 - 4 days**	**4 - 30 days**
AST					
Unit 1 EAB (NW)	2.50E-04				
Unit 2 EAB (SSE)	2.17E-04				
Unit 1/2 LPZ (NW)	2.00E-05	8.94E-06	6.14E-06	2.72E-06	8.48E-07
CLB					
EAB	5.29E-04				
LPZ	2.20E-05	2.20E-05	4.75E-06	1.54E-06	3.40E-07

Note:

AST: Based on RG 1.145, Revision 1 methodology and use of hourly meteorological data; see Section 5.1 for detail.

CLB: Based on RG 1.4, Revision 1 methodology.

Table B.2-1 Loss of Coolant Accident (LOCA) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	AST Value	CLB Value	Remark
Core Power Level (105% of the rated power of 3411 MWth)	3580 MWt		
Fuel Activity Release Fractions into containment	Per Reg. Guide 1.183 (see Section 7.2.3.2.6)	Per Reg. Guide 1.4: 100% core noble gases, 25% of core iodine.	
Fuel Release Timing (gap)	Onset: 30 sec Duration: 0.5 hr	Instantaneous release	
Fuel Release Timing (Early-In-Vessel)	Onset: 0.5 hr Duration: 1.3 hr		
Fuel Activity Release Fractions into sump liquid	Per Reg. Guide 1.183: Same as that released to containment with the exception of the noble gases which escape into the containment atmosphere	10% of core iodine (i.e., 100% of fuel gap activity) are released to the sump water	
Core Activity	Composite core source calculated by SCALE4.3 SAS2/ ORIGEN-S based on a range of 4.2% to 5.0% U-235 enrichment, a 19-month average fuel cycle, and a maximum core average burnup of 50 GWD/MTU. See Table 4.1-1	Composite core source calculated for 3.5% and 4.5% enrichments, and a burnup range from 0.1 to 50 GWD/MTU, by ORIGEN2 code; Activities of some isotopes are listed in Remark column for comparison.	Comparison of some core isotopes (Ci) AST / CLB I-131 9.90E7 9.760E7 I-133 2.01E8 1.992E8 Kr-85 1.11E6 1.226E6 Kr-88 6.43E7 1.045E8 Xe-133 2.01E8 1.929E8 Xe-135 4.92E7 5.570E7
Chemical Form of Iodine released from fuel to containment atmosphere	4.85% elemental 95% particulate 0.15% organic	91% elemental 5% particulate 4% organic	AST: Per RG 1.183 CLB: Per RG 1.4, R1
Chemical Form of Iodine Released from RCS and sump water	97% elemental 3% organic	99.75% elemental 0.25% organic	AST: Based on RG 1.183 CLB: Based on SG 25
Containment Vacuum/Pressure Relief Parameters			
Minimum Containment Free Volume:	2.550E+06 ft^3	Not Evaluated	
Primary Coolant Tech Spec Activity	Isotopic activity concentrations corresponding to 1 µCi/gm DE I-131 and 270 µCi/gm DE Xe-133		

Table B.2-1 Loss of Coolant Accident (LOCA) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	**AST Value**	**CLB Value**	**Remark**
Chemical Form of Iodine Released	97% elemental; 3% organic		
Maximum RCS flash fraction after LOCA Noble Gases Halogens	 100% 40%		
Maximum containment pressure relief line air flow rate	218 acfs		
Maximum duration of release via containment pressure relief line	13 sec		
Release Point	Plant Vent		
Containment Leakage Parameters			
Containment Spray Coverage –: Sprayed Volume Unsprayed Volume	Injection and Recirculation Spray Modes 82.5% (sprayed fraction) of Total volume of 2.55E+06 ft^3 2.103E+06 ft^3 4.470E+05 ft^3	Injection Spray Mode 82.5% (sprayed fraction) of Total volume : 2.55E+06 ft^3 2.103E+06 ft^3 4.470E+05 ft^3	
Minimum mixing flow rate from unsprayed to sprayed region: Before actuation of CFCUs After actuation of CFCUs	 2 unsprayed regions/hr (based on natural convection) 9.13 unsprayed regions/hr (based on 68,000cfm CFCU flow to address surveillance margins and uncertainty)	94,000 cfm between sprayed and unsprayed regions	AST: CFCU flowrate consistent with current containment analysis
Minimum duration of mixing via CFCUs	Start = 86 sec End = 30 days	Start = 0 sec End = 30 days	AST: CFCU start and termination time consistent with current containment analysis for one train operation.

Table B.2-1 Loss of Coolant Accident (LOCA)
Changes to Key Input Parameter Values: AST vs CLB

Parameter	AST Value	CLB Value	Remark
Containment spray in injection mode Initiation time Termination time	111 sec 3798 sec Spray removal credit ends for elemental iodine when the maximum DF is reached	86.5 sec Not used Spray removal credit ends when the maximum DF for elemental iodine is reached	AST: Injection spray start and termination times consistent with current containment analysis for one train operation.
Maximum delay between end of injection spray and initiation of recirculation spray	12 min (based on manual operator action)	Not Applicable	
Containment spray in recirculation mode Initiation time Termination time	4518 sec 22,518 sec	Not applicable	AST: Spray initiation / termination in recirculation mode consistent with analysis performed to obtain containment conditions if recirculation CS is initiated 12 minutes after injection spray is terminated (1 train operation).
Long-term Sump Water pH	≥ 7.5 (includes acid production), thus no iodine revolution	No iodine re-evolution	
Maximum allowable DF for fission product removal	Elemental Iodine: 200 Others: not applicable	Elemental Iodine: 100 Others: not applicable	CLB: The value of 100 is applied to the initial release of 25% core iodine to the containment assuming an immediate 50% plateout. The effective DF is 200 if the initial release fraction of 50% was addressed.
Elemental iodine and particulate spray removal coefficients in sprayed region during both injection spray and recirculation spray modes	See Table 7.2-2	Constant spray removal coefficients as given below Elemental iodine:31 hr^{-1} Particulate iodine: 0 hr^{-1}	
Elemental iodine removal coefficients due to wall deposition	See Table 7.2-2	None	
Particulate removal coefficients in unsprayed region due to gravitational settling	See Table 7.2-2	None	
Containment Leak rate (0-24 hr)	0.1% weight fraction per day		

Table B.2-1 Loss of Coolant Accident (LOCA)
Changes to Key Input Parameter Values: AST vs CLB

Parameter	AST Value	CLB Value	Remark
Containment Leak rate (1-30 day)	0.05% weight fraction per day		
Containment Leakage Release Point (Unfiltered)	From the worst case release point of the following: • Diffuse source via the containment wall • Via Plant Vent • Via Containment Pen Area GE • Via Containment Pen Areas GW & FW	Top of Containment.	
ESF System Environmental Leakage Parameters			
Minimum post-LOCA containment water volume sources	480,015 gal.	469,000 gal for the ESF leakage source 373,220 gal for the RHR pump seal failure source	AST: Based on current analysis (Min volume of RWST, RCS, SAT and Accumulators) CLB: Based on RWST, RCS and Accumulators; volumes used are different from current analysis.
Minimum time after LOCA when recirculation is initiated	829 sec	0 sec	AST: Based on current containment analysis
Duration of leakage	30 days		
Maximum ESF fluid temperature after initiation of recirculation (used to establish iodine airborne fraction)	259.9 °F	See DF values below.	AST: Based on current containment analysis assuming operation of CS in the recirculation mode.
Maximum ESF leak	Unfiltered via plant vent = 240 cc/min Unfiltered via Containment Penetration Areas GE or GW & FW = 12 cc/min	Unfiltered via plant vent = 1910 cc/hr (~ 32 cc/min) A maximum additional filtered / unfiltered leak via the plant vent of 1.85 gpm (7003 cc/min) / 0.186 gpm (or 704 cc/min)	AST: Based on current operational data with margin. Listed values include a factor of 2. CLB: The maximum additional filtered / unfiltered leak rates of 1.85 gpm (7003 cc/min) / 0.186 gpm (or 704 cc/min) are derived in the CLB analysis and are each based on using all of the dose margin to the regulatory limit. In addition, none of the CLB leakage values have a "safety factor of 2."

Table B.2-1 Loss of Coolant Accident (LOCA) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	**AST Value**	**CLB Value**	**Remark**
RHR pump seal failure	Filtered via plant vent 50 gpm starting at t = 24 hrs for 30 min		AST: Releases from the RHR Pump Seal failure are filtered for CR dose evaluation and filtered for Site Boundary Dose Evaluation
Iodine Airborne Release Fraction	10% (for both RHR pump seal and ESF leakage)	DF for RHR pump seal failure release: 132	CLB: Iodine partition credited for RHR pump seal failure which is a large short-term release.
		DF for ESF leakage: 1.0	Iodine partition not credited for ESF leakage which is a small long-term release.
Auxiliary Building ESF Ventilation System filter efficiency	Elemental iodine: 88% Organic iodine: 88% (See Section 7.2.3.4)	Elemental iodine: 90% Organic iodine: 70% Particulate iodine: 90%	AST: AB Ventilation filter credited for RHR Pump Seal Failure. CLB: AB Ventilation filter credited for RHR Pump Seal Failure
Refueling Water Storage Tank (RWST) Back-Leakage Parameters			
Earliest initiation time of RWST back-leakage	829 sec	Not Evaluated	
Maximum ECCS / sump water back-leakage rate to RWST (includes safety factor of 2)	2 gpm		
RWST back-leakage iodine release fractions	See Table 7.2-3		
RWST back-leakage noble gas, as iodine daughters, release rate from the RWST vent	See Table 7.2-3		
Miscellaneous Equipment Drain Tank (MEDT) Leakage Parameters			
MEDT inflow rate (includes safety factor of 2)	1900 cc/min	Not Evaluated	
MEDT leakage Iodine release fractions	See Table 7.2-4		
MEDT leakage noble gas, as iodine daughters release rate from plant vent	See Table 7.2-4		
CR Emergency Ventilation: Initiation Signal/Timing			

Table B.2-1 Loss of Coolant Accident (LOCA)
Changes to Key Input Parameter Values: AST vs CLB

Parameter	AST Value	CLB Value	Remark
Initiation time (signal)	SI signal generated: 6 sec Non-Affected Unit NOP Intake Isolated: 18 sec Affected Unit NOP Intake Isolated and CRVS Mode 4 in full Operation: 44.2 sec	CLB assumes a 10-second closure time for the CRVS outside air isolation dampers.	
Bounding Control Room Atmospheric Dispersion Factors for LOCA	Table B.2-2A (Same as Table 7.2-5, Refer to Section 5.2 for detail)	Table B.2-2B Based on a modified Halitsky methodology	

Table B.2-2A Loss of Coolant Accident (LOCA) **AST Values**: Limiting Control Room Atmospheric Dispersion Factors (sec/m³)					
Release Location / Receptor	**0-2 hr**	**2-8 hr**	**8-24 hr**	**24-96 hr**	**96-720 hr**
Control Room Normal Intakes					
Plant Vent Release					
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	9.08E-04	-----	-----	-----	-----
Containment Penetration Areas					
- Affected Unit Intake	6.60E-03	-----	-----	-----	-----
-Non-Affected Unit Intake	2.08E-03	-----	-----	-----	-----
Control Room Infiltration					
Plant Vent	1.25E-03	8.93E-04	3.47E-04	3.46E-04	2.98E-04
Containment Penetration Areas	3.09E-03	1.83E-03	7.22E-04	7.13E-04	6.50E-04
RWST Vent	1.05E-03	5.55E-04	2.12E-04	2.12E-04	1.72E-04
Control Room Pressurization Intake					
Plant Vent	5.55E-05	3.68E-05	1.36E-05	1.38E-05	1.11E-05
Containment Penetration Areas	6.00E-05	3.98E-05	1.63E-05	1.37E-05	1.10E-05
RWST Vent	4.73E-05	2.93E-05	1.13E-05	1.08E-05	8.50E-06

Note 1: Release from the Containment penetration areas (i.e., areas GE or GW & FW): applicable to containment leakage and ESF system leakage that occurs in the Containment Penetration Area

Note 2: Release from Plant Vent: applicable to ESF system leakage that occurs in the Auxiliary building, MEDT releases, RHR Pump Seal Failure Release and Containment Vacuum/Pressure Relief Line Release

Table B.2-2B Loss of Coolant Accident (LOCA) **CLB Values[1]**: Control Room Atmospheric Dispersion Factors (sec/m³)				
Release Location / Receptor	**0-8 hr**	**8-24 hr**	**1-4 days**	**4-30 days**
Unfiltered inleakage/intake	1.96E-04	1.49E-04	1.08E-04	6.29E-05
Filtered pressurization intake	7.05E-05	5.38E-05	3.91E-05	2.27E-05

Note 1:

The above control room X/Q values are used for all postulated releases and are based on a release point on the top of containment.

Table B.3-1 Fuel Handling Accident in Fuel Handling Building or Containment (FHA) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	**AST Value**	**CLB Value**	**Remark**
Power Level	3580 MWt		
Number of Damaged Fuel Assemblies	1		
Total Number of Fuel Rods damaged	264 (i.e., all rods in damaged fuel assembly)		
Decay Time Prior to Fuel Movement	72 hours	100 hrs	
Radial Peaking Factor	1.65		
Fraction of Core Inventory in gap	I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%) Refer to Section 4.3	Kr-85 (30%) Other Noble Gases (10%) All Halides (10%) Based on SG 25	
Equilibrium Fuel Assembly Activity	Based on 72 hour decay. Refer to Table 7.3-2	Based on 100 hour decay. Activities of some isotopes after 100 hr decay are listed in Remark column for comparison.	Comparison of a single fuel assembly activity (Ci) AST(72 hr decay) CLB(100 hr decay) I-131 4.09E5 3.625E5 I-133 9.73E4 3.783E4 Kr-85 5.75E3 6.350E3 Xe-133 8.31E5 6.914E5
Iodine form of gap release before scrubbing in pool/reactor cavity	99.85% elemental 0.15% Organic	99.75% elemental 0.25% Organic	AST: Based on RG 1.183 CLB: Based on SG 25
Iodine form of gap release after scrubbing in pool/reactor cavity	57% elemental 43% Organic	75% elemental 25% Organic	AST: Based on RG 1.183 CLB: Based on SG 25
Pool / reactor cavity scrubbing Decontamination Factors	Iodine (200, effective) Noble Gas (1) Particulates (∞)		AST/CLB: Based on RG 1.183
Rate of Release from Fuel	Puff		
Environmental Release Points and Rates			
Environmental Release Rate	All airborne activity released within a 2 hour period (or less if the ventilation system promotes a faster release rate)	FHA in FHB: All airborne activity released within a 2 hour period FHA in Containment: Puff release	AST FHA: *In FHB*: Analysis uses the actual release rate lambda based on the FHBVS exhaust (i.e., 8.7 hr^{-1}) since it is larger than the release rate applicable to "a 2-hr release" per regulatory guidance (i.e., 3.45 hr^{-1}). *In Containment:* Analysis uses a release rate applicable to "a 2-hr release"
FHA in the FHB – Release flow rates	-Plant Vent – 46,000 cfm FHB Outleakage -Ingress/Egress – 30 cfm	Plant vent – 40,000 cfm	AST: Per Fan curve, 46,000 cfm is maximum flow; other out leakages are conservative assumptions, see Section 7.3

Table B.3-1 Fuel Handling Accident in Fuel Handling Building or Containment (FHA) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	AST Value	CLB Value	Remark
	-Miscellaneous gaps / openings – 470 cfm		
Minimum free volume in FHB above SFP	317,000 ft^3	435,000 ft^3	AST: Based on updated assessment of free volume above the SFP
FHA in Containment - Release flow rates	-Open Equipment Hatch – All airborne activity released in 2 hrs	Open containment - All airborne activity released in 1 second (puff release)	
Minimum Free Volume in Containment above Operating Floor	2,013,000 ft^3	33,600 ft^3 containment volume above fuel pool	AST: Based on updated assessment of volume available for dilution of FHA releases CLB: Conservative value based on a small rectangular parallelepiped of air space above the 25 ft x 70 ft square feet pool surface in reactor cavity.
CR Emergency Ventilation: Initiation Signal/Timing			
Signal(s) available to switch the CRVS from normal operation (NOP) Ventilation (Mode 1) to Pressurized Filtered Ventilation (Mode 4) following a FHA	Radiation signals from gamma sensitive intake monitors that initiate closure of the CR normal intake dampers and switch the CRVS from normal operation Ventilation Mode 1 to Pressurized Filtered Ventilation Mode 4. (Refer to Section 7.3) No LOOP (Refer to Section 7.1)	Assumes CR is in normal ventilation mode, with unfiltered inlet/inleakage and exhaust flow rate of 2110 cfm for the duration of the accident. No LOOP	
Delay time for CRVS Mode 4 operation, including monitor response, signal processing, and damper closure time	32 seconds (see below)	Not applicable	
Radiation Monitor Response Time	20 seconds (conservative assumption) -(Refer to Section 7.3)		
Radiation monitor signal processing time	2 seconds		
Damper Closure Time	10 seconds		
Bounding Control Room Atmospheric Dispersion Factors for FHA	Table B.3-2A (Same as Table 7.3-3; Refer to Section 5.2 for detail)	Table B.3-2B Based on a modified Halitsky methodology	

Table B.3-2A Fuel Handling Accident (FHA) **AST Values**: Limiting Control Room Atmospheric Dispersion Factors (sec/m³)						
Release Location / Receptor	**0-22 sec**	**22 sec - 2 hr**	**2-8 hr**	**8-24 hr**	**1-4 d**	**4-30 d**
Control Room Normal Intakes						
Containment Hatch Release						
- Affected Unit Intake	2.48E-02	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	2.67E-03	-----	-----	-----	-----	-----
Plant Vent Release						
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	9.08E-04	-----	-----	-----	-----	-----
FHB Out-leakage points						
- Affected Unit Intake	6.68E-03	-----	-----	-----	-----	-----
- Non-Affected Unit Intake	2.69E-03	-----	-----	-----	-----	-----
Control Room Infiltration						
Containment Hatch Release	5.09E-03	5.09E-03	-----	-----	-----	-----
Plant Vent	1.25E-03	1.25E-03	-----	-----	-----	-----
FHB Out-leakage points	3.61E-03	3.61E-03	-----	-----	-----	-----
Control Room Pressurization Intake						
Containment Hatch Release	-----	6.15E-05	-----	-----	-----	-----
Plant Vent	-----	5.55E-05	-----	-----	-----	-----
FHB Out-leakage points	-----	6.13E-05	-----	-----	-----	-----

Table B.3-2B Fuel Handling Accident (FHA) **CLB Values[1]**: Control Room Atmospheric Dispersion Factors (sec/m³)				
Release Location / Receptor	**0-8 hr**	**8-24 hr**	**1-4 days**	**4-30 days**
Unfiltered inleakage/intake	1.96E-04	-	-	-

Note 1:

The above control room X/Q values are used for all postulated releases and are based on a release point on the top of containment.

Table B.4-1 Locked Rotor Accident (LRA) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	AST Value	CLB Value	Remark
Power Level	3580 MWt	3568 MWt	
Reactor Coolant Mass	446,486 lbm	Not Available	
Primary to Secondary SG tube leakage	0.75 gpm at STP (total for all 4 SGs)	1.0 gpm at STP (total for all 4 SGs)	
Melted Fuel Percentage	0%		
Failed Fuel Percentage	10%		
Equilibrium Core Activity	Composite core source calculated by SCALE4.3 SAS2/ ORIGEN-S based on current enrichment and burnup. (Refer to Section 4.1 & Table 4.1-1)	Calculated by EMERALD NORMAL computer program based on 3568 MWt core power, 3.18% enrichment, and 12 month fuel cycle. (Refer to UFSAR Table 11.1-2 & 11.1-4)	
Radial Peaking Factor	1.65	Not used	
Fraction of Core Inventory in Fuel gap	I-131 (8%) I-132 (23%) Kr-85 (35%) Other Noble Gases (4%) Other Halides (5%) Alkali Metals (46%)	DCPP specific: Gap fractions are isotope dependent. Provided below are some of the values (based on hot channel factor of 1.70): I-131 - 0.822% Kr-85 - 16.7% Xe-133 - 0.667% (Refer to UFSAR Table 11.1-7)	
Iodine Chemical Form in Gap	4.85% elemental 95% Particulate 0.15% organic	100% elemental	
Secondary Side Parameters			
Initial and Minimum SG Liquid Mass	92,301 lbm/SG	Not Available	
Iodine Species Released to Environment	97% elemental; 3% organic	100% elemental	
Time period of tubes being uncovered	insignificant		

Table B.4-1 Locked Rotor Accident (LRA)
Changes to Key Input Parameter Values: AST vs CLB

Parameter	AST Value	CLB Value	Remark
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs	0-2 hrs: 656,000 lbm 2-8 hrs: 1,035,000 lbm	AST: Based on RSG and current allowable T_{avg} and T_{feed} range
Iodine Partition Coefficient in SGs	100		
Particulate Carry-Over Fraction in SGs	0.0005 by weight	Not Applicable	
Fraction of Noble Gas Released	1.0 (Released without holdup)		
Termination of releases from SGs	10.73 hours	8 hrs	
Environmental Release Point	MSSVs/10% ADVs		
Initial and Minimum SG Liquid Mass	92,301 lbm/SG	Not Available	

Note: No comparison is provided for Control Room parameters since the CLB does not include a dose assessment in the Control room following a LRA.

Table B.5-1 Control Rod Ejection Accident (CREA) Changes to Key Input Parameter Values: AST vs CLB			
<u>Parameters</u>	<u>AST Value</u>	<u>CLB Value</u>	<u>Remark</u>
Containment Leakage Pathway			
Power Level	3580 MWt	3568 MWt	
Equilibrium Core Activity	Composite core source calculated by SCALE4.3 SAS2/ ORIGEN-S based on current enrichment and burnup. (Refer to Section 4.1 & Table 4.1-1)	The core activity is calculated by EMERALD NORMAL computer program based on 3568 MWt core power, 3.18% enrichment, and 12 month fuel cycle. (Refer to UFSAR Table 11.1-2 & 11.1-4)	
Free Volume	2.55E+06 ft^3		
Containment leak rate (0 -24 hr)	0.1% vol. fraction per day		
Containment leak rate(1-30 day)	0.05% vol. fraction per day		
Failed Fuel Percentage	10%		
Percentage of Core Inventory in Fuel Gap	10% core noble gases 10% core halogens	DCPP specific: Gap fractions are isotopic dependent. Provided below are some of the values are (based on hot channel factor of 1.70): I-131 - 0.822% Kr-85 - 16.7% Xe-133 - 0.667% (Refer to UFSAR Table 11.1-7)	<u>AST</u>: Per RG 1.183 <u>CLB</u>: Plant specific
Percentage of fission products released to coolant that are released to the containment atmosphere	100% of the noble gases 100% of the iodines	100% of the noble gases 10% of the halogens	
Melted Fuel Percentage	0%		
Chemical Form of Iodine in Failed fuel	4.85% elemental; 95% Particulate; 0.15% organic	Not stated	<u>AST</u>: Per RG 1.183
Radial Peaking Factor	1.65	Not used	
Core Activity Release Timing	Puff		
Form of Iodine from failed fuel in the Containment Atmosphere	97% elemental; 3% organic	100% elemental	<u>AST</u>: Per RG 1.183

Table B.5-1 Control Rod Ejection Accident (CREA)
Changes to Key Input Parameter Values: AST vs CLB

Parameters	AST Value	CLB Value	Remark
Credit taken for containment sprays	No	Yes	
Iodine removal coefficients by sprays	Not applicable	Same as LOCA Elemental iodine:31 hr^{-1}	
Termination of Containment Release	30 days		
Environmental Release Point	Same as LOCA Containment Leakage pathway		
Secondary Side Pathway			
Reactor Coolant Mass	446,486 lbm	The assumptions and inputs for the secondary side pathway are the same as that described in Locked Rotor Accident (Table B.4-1) for CLB	
Primary-to-Secondary Leak rate	0.75 gpm for all 4 SGs		
Failed Fuel Percentage	Same as containment leakage pathway		
Percentage of Core Inventory in Fuel Gap	Same as containment leakage pathway		
Minimum Post-Accident SG Liquid Mass	92,301 lbm / SG		
Iodine Species released to Environment	97% elemental; 3% organic		
Time period when tubes not totally submerged	Insignificant		
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs.		
Iodine Partition Coefficient in SGs	100		
Fraction of Noble Gas Released	1.0 (Released without holdup)		
Termination of Release from SGs	10.73 hours		
Environmental Release Point	MSSVs/10% ADVs		

Note: No comparison is provided for Control Room parameters since the CLB does not calculate a dose assessment in the Control room following a CREA. A qualitative statement made in the current UFSAR indicates that since the activity releases following a CREA will be less than a LOCA, the CR dose will be below GDC 19.

TABLE B.6-1 Main Steam Line Break (MSLB) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	AST Value	CLB Value	Remark
Power Level	3580 MWt		
Reactor Coolant Mass	446,486 lbm	566,000 lbm	AST: A smaller RCS mass is more conservative for the accident-initiated iodine spike (AIS) case. RCS mass is not sensitive to the dose consequences for the pre-accident iodine spike (PIS) case because the amount of RCS loss due to SG leakage is small compared to the total inventory.
Leak rate to Faulted Steam Generator	0.75 gpm at STP (conservative assumption)	The calculated maximum allowable accident induced primary to secondary leak rate of 10.5 gpm based on Alternate Repair Criteria.	
Leak rate to Intact Steam Generators	0 gpm (all leakage assumed into faulted SG)	0.3125 gpm (total for 3 intact SGs)	CLB: TS 3.4.13d allows a maximum operational leakage of 150 gpd per SG, which is equivalent to 0.3125 gpm for 3 SGs.
Failed/Melted Fuel Percentage	0%		
RCS Tech Spec Iodine Conc.	1 µCi/gm DE I-131		
RCS Tech Spec Noble Gas Conc.	270 µCi/gm DE Xe-133	RCS NG activity is based on 1% fuel defects.	AST: The AST NG values correspond to 0.5% fuel defects.
RCS Equilibrium Iodine Appearance Rates	Fuel to RCS appearance rate that results in 1 µCi/gm DE I-131, based on 132 gpm letdown flow rate, 100% ion-exchanger efficiency and 11 gpm RCS leakage	Appearance rate that results in 1 µCi/gm DE I-131, based on 132 gpm letdown flow rate, 100% ion-exchanger efficiency, 11 gpm RCS leakage , 1 gpm boron control shim bleed, and 1 gpm tube leakage	Comparison of I-131 appearance rate: AST – 4.31E-01 Ci/min CLB – 4.29E-01 Ci/min
Pre-Accident Iodine Spike Concentrations	60 µCi/gm DE I-131		
Accident-Initiated Iodine Spike Appearance Rate	500 times equilibrium appearance rate		
Duration of Accident- Initiated Iodine Spike	8 hours		
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131		

TABLE B.6-1 Main Steam Line Break (MSLB) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	AST Value	CLB Value	Remark
Secondary System Release Parameters			
Iodine Species released to Environment	97% elemental; 3% organic	Not specified	Since the CR filter efficiency is the same for the elemental iodine and the organic iodine, this parameter value is inconsequential for CR dose.
Fraction of Iodine Released from Faulted SG	1.0 (Released to Environ without holdup)		
Fraction of Noble Gas Released from Faulted SG	1.0 (Released to Environ without holdup)		
Liquid mass in each SG	Faulted: 182,544 lbm (max.) Intact: 92,301 lbm (min. and initial)	Faulted: 162,784 lbm Intact: 81,500 lbm / SG	AST: Uses maximum mass for Faulted SG (Hot Zero Power value, RSG / current allowable T_{avg} and T_{feed} range) to maximize dose consequences of release from faulted SG. The initial SG liquid mass for the intact SGs based on RSG / current allowable T_{avg} and T_{feed} range. The initial SG liquid mass is used to determine the total iodine inventory in the SG liquid prior to the accident. The SG liquid mass increases following a reactor trip, so the minimum SG liquid mass post-accident is also the SG initial liquid mass. The minimum post-accident SG liquid mass is used to determine the iodine activity release rate during the accident.
Release Rate of SG liquid from Faulted SG	Dryout of SG liquid within10 seconds	Instantaneous dryout	AST: Dryout time based on RSG / current allowable T_{avg} and T_{feed} range.
Termination of 0.75 gpm leak primary to secondary leak from Faulted SG	30 hrs (when RCS reaches 212 °F)	8 hrs (assumption)	
Steam Releases from intact SGs	0-2 hrs: 384,000 lbm 2-8 hrs: 893,000 lbm 8-10.73 hrs: Same release rate as that for 2-8 hrs	0-2 hrs: 393,464 lbm 2-8 hrs: 915,000 lbm	AST: Based on RSG / current allowable T_{avg} and T_{feed} range
Iodine Partition Coefficient in Intact SG	100 (SGs fully covered)	No iodine partition factor is credited.	
Termination of release from Intact SG	10.73 hours (calculated time for initiation of shutdown cooling)	8 hrs (assumption)	
Release Point: Faulted SG	Outside containment, at the steam line break location		

TABLE B.6-1 Main Steam Line Break (MSLB) Changes to Key Input Parameter Values: AST vs CLB			
Parameter	AST Value	CLB Value	Remark
Release Point: Intact SG	MSSVs/10% ADVs		
CR Emergency Ventilation : Initiation Signal/Timing			
Initiation (signal)	SIS / Phase A		
Unaffected Unit CRVS inlet damper fully closed	Within 12.6 seconds	Conservatively assumes that the control room is isolated in 2 minutes	
Affected Unit CRVS inlet dampers fully closed	Within 38.8 seconds		
Control Room Atmospheric Dispersion Factors	Table B.6-2A (Same as Table 7.6-2; Refer to Section 5.2 for detail)	Table B.6-2B (based on a modified Halitsky methodology)	

Table B.6-2A Main Steam Line Break (MSLB) AST Values: Limiting Control Room Atmospheric Dispersion Factors (sec/m^3)				
Receptor – Release Point	0-2hr	2-8hr	8-10.73hr	10.73-30hr
CR NOP Intake - Faulted SG (Break Location)	Note 1			
CR NOP Intake - Intact SG (MSSVs/10% ADVs)	8.12E-04			
CR Inleakage - Faulted SG (Break Location)	1.14E-02	7.22E-03	3.00E-03	3.00E-03
CR Inleakage - Intact SG (MSSVs/10% ADVs)	2.46E-03	1.59E-03	1.59E-03	-----
CR Emergency Intake & Bypass - Faulted SG (Break Location)	6.85E-05	4.70E-05	1.85E-05	1.85E-05
CR Emergency Intake & Bypass - Intact SG (MSSVs/10% ADVs)	1.40E-05	9.40E-06	9.40E-06	-----

Note 1: ARCON96 based χ/Q s are not applicable for this case

Table B.6-2B Main Steam Line Break (MSLB) CLB Values[1]: Control Room Atmospheric Dispersion Factors (sec/m^3)				
Receptor	0-8 hr	8-24 hr	1-4 days	4-30 days
Unfiltered inleakage/intake	1.96E-04	1.49E-04	1.08E-04	6.29E-05
Filtered pressurization intake	7.05E-05	5.38E-05	3.91E-05	2.27E-05

Note 1:

The above control room X/Q values are used for all postulated releases and are based on a release point on the top of containment

Table B.7-1 Steam Generator Tube Rupture (SGTR)
Changes to Key Input Parameter Values AST vs CLB

Parameter	AST Value	CLB Value	Remark
Power Level	3580 MWt		
Reactor Coolant Mass	446,486 lbm	499,500 lbm	AST: A smaller RCS mass is more conservative for the accident-initiated iodine spike (AIS) case. A larger RCS mass is more conservative for the pre-accident iodine spike (PIS) case because the integrated RCS break flow is significant compared to the initial inventory. AIS case is the more limiting case relative to the regulatory limits.
Time of Reactor Trip	179.0 sec		
Time of isolation of stuck-open 10% ADV on the Ruptured SG	2653 sec		
Termination of Break Flow from Ruptured SG that flashes	3402 sec		
Termination of Break Flow from Ruptured SG	5872 sec		
Time of manual depressurization of the Ruptured SG	2 hours		
Break Flow to Ruptured Steam Generator that flashes	Table 7.7-2, Column "A"		
Break Flow to Ruptured Steam Generator that does not flash	Table 7.7-2, Column "B"		
Tube Leakage rate to Intact Steam Generators	0.75 gpm at STP (for 3 intact SGs)	1.0 gpm at STP (for 3 intact SGs)	
Failed/Melted Fuel Percentage	0%		
RCS Tech Spec Iodine Concentration	1 μCi/gm DE I-131		
RCS Tech Spec Noble Gas Concentration	270 μCi/gm DE Xe-133	RCS NG activity is based on 1% fuel defects.	AST: The AST NG values correspond to 0.5% fuel defects.
RCS Equilibrium Iodine Appearance Rates	Fuel to RCS appearance rate that results in 1 μCi/gm DE I-131, based on 132 gpm letdown flow rate, 100% ion-exchanger efficiency and 11 gpm RCS leakage	Appearance rate that results in 1 μCi/gm DE I-131, based on 132 gpm letdown flow rate, 100% ion-exchanger efficiency and 11 gpm RCS leakage	Comparison of I-131 appearance rate: AST – 4.31E-01 Ci/min CLB – 4.41E-01 Ci/min
Pre-Accident Iodine Spike Concentration	60 μCi/gm DE I-131		

Table B.7-1 Steam Generator Tube Rupture (SGTR) Changes to Key Input Parameter Values AST vs CLB			
Parameter	AST Value	CLB Value	Remark
Accident-Initiated Iodine Spike Appearance Rate	335 times TS equilibrium appearance rate		
Duration of Accident-Initiated Iodine Spike	8 hours		
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131		
Secondary System Release Parameters			
Initial SG liquid mass	89,707 lbm/SG	106,000 lbm for ruptured SG 118,500 lbm for each of the 3 intact SGs	CLB: Values reflect average SG mass during the transient. AST: Value based on current analysis supporting RSG / current allowable T_{avg} and T_{feed} range. The initial SG liquid mass is used to determine the total iodine inventory in the SG liquid prior to the accident
Iodine Species released to Environment	97% elemental; 3% organic	Not specified	Since the CR filter efficiency is the same for the elemental iodine and the organic iodine, this parameter value is inconsequential for CR dose.
Steam flow rate to condenser from Ruptured SG before trip	63,000 lbm/min		
Steam flow rate to condenser from intact SGs before trip	189,000 lbm/min		
Partition Factor in Main Condenser	0.01 (elemental iodine)		
	1 (organic iodine and noble gases)		
Steam Releases from Ruptured SG	Table .7.7-2, Column "C"		
Steam Releases from intact SG	Table 7.7-2, Column "D"		
Post-accident minimum SG liquid mass for Ruptured SG	89,707 lbm	106,000 lbm	CLB value is the average of the initial SG mass (89,707 lbm) and the minimum mass (122,500 lbm) during the stuck open 10% ADV phase of the transient. AST: The minimum post-accident SG liquid mass is used to determine the iodine activity release rate during the accident. The SG liquid mass increases following a reactor trip, so the minimum SG liquid mass post-accident is also the SG initial liquid mass

Table B.7-1 Steam Generator Tube Rupture (SGTR) Changes to Key Input Parameter Values AST vs CLB			
Parameter	**AST Value**	**CLB Value**	**Remark**
Post-accident minimum SG liquid mass for intact SGs	89,707 lbm per SG	118,500 lbm per SG	CLB: value is the average of the initial SG mass (89,707 lbm) and the minimum mass (147,400 lbm) during the transient at the end of the cooldown. AST: The minimum post-accident SG liquid mass is used to determine the iodine activity release rate during the accident. The SG liquid mass increases following a reactor trip, so the minimum SG liquid mass post-accident is also the SG initial liquid mass
Time period of tube uncovery for intact SG	insignificant		
Fraction of Iodine Released (flashed portion)	1.0 (Released without holdup)		
Fraction of Noble Gas Released from all SGs	1.0 (Released without holdup)		
Iodine Partition Coefficient	100		
Termination of Release from intact SG	10.73 hrs (calculated time for initiation of shutdown cooling)	8 hrs (assumed)	
Environmental Release Points	Plant Vent : 0 – 179 sec MSSVs/10% ADVs: 179 sec – 10.73 hr	Condenser exhaust : 0 – 179 sec MSSVs/10% ADVs:179 sec – 8 hr	
CR emergency Ventilation : Initiation Signal/Timing			
Initiation time (signal)	SIS: 219 sec Unaffected Unit inlet damper closed: 231 sec Affected Unit inlet damper closed: 257.2 sec	SIS: 215 sec Unaffected Unit & Affected Unit inlet damper closed: 250 sec	
Control Room Atmospheric Dispersion Factors	Table B.7-2A (Same as Table 7.7-3; Refer to Section 5.2 for detail)	Table B.7-2B (based on a modified Halitsky methodology)	

TABLE B.7-2A Steam Generator Tube Rupture (SGTR) AST Values: Limiting Control Room Atmospheric Dispersion Factors (sec/m³)					
Release Location / Receptor	**0-179s**	**179-257.2s**	**257.2s- 2hr**	**2-8hr**	**8-10.73hr**
Control Room Normal Intakes					
Plant Vent	1.29E-03	-----	-----	-----	-----
MSSVs/10% ADVs (Note 1)	-----	8.12E-04	-----	-----	-----
Control Room Infiltration					
Plant Vent	1.25E-03	-----	-----	-----	-----
MSSVs/10% ADVs	-----	2.46E-03	2.46E-03	1.47E-03	1.47E-03
Control Room Pressurization Intake					
MSSVs/10% ADVs	-----	-----	1.40E-05	9.40E-06	9.40E-06

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR 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 normal operation CR intake of the affected unit. Thus the χ/Q s presented reflect those applicable to the CR intake of the unaffected unit.

Table B.7-2B Steam Generator Tube Rupture (SGTR) CLB Values[1]: Control Room Atmospheric Dispersion Factors (sec/m³)				
Receptor	**0-8 hr**	**8-24 hr**	**1-4 days**	**4-30 days**
Unfiltered inleakage/intake	1.96E-04	1.49E-04	1.08E-04	6.29E-05
Filtered pressurization intake	7.05E-05	5.38E-05	3.91E-05	2.27E-05

Note 1:

The above control room X/Q values are used for all postulated releases and are based on a release point on the top of containment.

Table B.8-1 Loss of Load (LOL) Event Changes to Key Input Parameter Values: AST vs CLB			
Parameter	**AST Value**	**CLB Value**	**Remark**
Power Level	3580 MWt	3568 MWt	
Reactor Coolant Mass	446,486 lbm	Not Available	
Primary to Secondary SG tube leakage	0.75 gpm at STP (Four SGs)	1.0 gpm at STP (Four SGs)	
Failed/Melted Fuel Percentage	0%		
RCS Technical Specification Iodine Levels	1 µCi/gm DE I-131	Design basis RCS (1% fuel cladding defects) noble gas and iodine activity	
RCS Technical Specification Noble Gas Levels	270 µCi/gm DE Xe-133		
RCS Equilibrium Iodine Appearance Rates	Fuel to RCS appearance rate that results in 1 µCi/gm DE I-131, based on 132 gpm letdown flow rate, 100% ion-exchanger efficiency and 11 gpm RCS leakage	DCPP specific	
Accident-Initiated Iodine Spike Appearance Rate	500 times TS equilibrium appearance rate	30 times the appearance rate of normal operation	
Pre-Accident Iodine Spike Concentration	60 µCi/gm DE I-131	Not evaluated	
Duration of Accident-Initiated Iodine Spike	8 hrs		
Initial Secondary Coolant Iodine Concentrations	0.1 µCi/gm DE I-131	Based on 1% fuel defects and 1 gpm Primary to secondary Leakage	
Initial and Minimum SG Liquid Mass	92,301 lbm/SG	Not Available	
Time period of tubes uncovered	insignificant		
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs	0-2 hrs: 656,000 lbm 2-8 hrs: 1,035,000 lbm	AST: Based on RSG and current allowable T_{avg} and T_{feed} range
Iodine Partition Coefficient in SGs	100		
Iodine Species Released to Environment	97% elemental; 3% organic	100% elemental	
Fraction of Noble Gas Released	1.0 (Released without holdup)		

Table B.8-1 Loss of Load (LOL) Event Changes to Key Input Parameter Values: AST vs CLB			
Parameter	**AST Value**	**CLB Value**	**Remark**
Termination of releases from SGs	10.73 hours	8 hrs	
Environmental Release Point	MSSVs/10% ADVs		

Note: No comparison is provided for Control Room parameters since the CLB does not include a dose assessment in the Control room following a Loss of Load Event