15.5.8.8 ACTIVITY2

ACTIVITY2 (Reference 65) 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 reactor coolant activity inventory (design and as limited by the plant Technical Specifications) utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.9 IONEXCHANGER

ION EXCHANGER (Reference 66) 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 secondary coolant activity inventory (design and as limited by the plant Technical Specifications) utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.10 EN 113, Atmospheric Dispersion Factors

EN-113 Atmospheric Dispersion Factors (Reference 73) calculates χ/Q values at the EAB and LPZ following the methodology and logic outlined in Regulatory Guide 1.145, Revision 1. 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 utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.11 ARCON96

ARCON96 (Reference 74) was developed by Pacific Northwest National Laboratory (PNNL) for the 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 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 control room and TSC atmospheric dispersion factors utilized to assess the design basis accidents excluding the tank ruptures.

15.5.8.12 SWNAUA

SWNAUA (Reference 67) 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. NAUA/Mod 4 was modified 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 containment.

15.5.8.13 RADTRAD 3.03

RADTRAD 3.03 (Reference 68) 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). 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 Thyroid and TEDE dose (rem) to the public located offsite and to onsite personnel located in the control room 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 control room due to inhalation and submersion in airborne radioactivity following design basis accidents excluding tank ruptures

15.5.8.14 PERC2

PERC2 (Reference 69) is a multi-region activity transport and radiological dose consequence program. It includes the following major features:

(1) Provision of time-dependent releases from the reactor coolant system to the containment atmosphere.

(2) Provision for airborne radionuclides for both TID and AST release assumptions, including daughter in growth.

(3) Provision for calculating the CEDE to individual organs as well as EDE from inhalation, DDE and beta from submersion, and TEDE.

(4) Provisions for tracking time-dependent inventories of all radionuclides in all control regions of the plant model.

(5) 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 control room 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 post-LOCA external and contained radiation sources. This source term information is input into SW_QADCGGP to develop the direct shine dose to the control room. PERC2 is also used to develop the decay heat in the RWST and MEDT and develop the TEDE dose to personnel located in the TSC due to inhalation and submersion in airborne radioactivity following LOCA.

15.5.8.15 SW-QADCGGP

SW-QADCGGP (Reference 70) 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.

SW-QADCGGP is used to develop the direct shine dose to the operator in the control room, TSC and EAB.

15.5.8.16 GOTHIC

GOTHIC (Reference 71) is developed and maintained by Numerical Applications Incorporated (NAI) and 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 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.

15.5.9 CONTROL ROOM DESIGN AND TRANSPORT MODEL

The control room 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 as the control room 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 control room Mechanical Equipment and HVAC room is located adjacent to the control room (east side), at El 154'-6".

The control room 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 control room

pressurization air intakes have dual ventilation outside air intake design as defined by Regulatory Position C.3.3.2 of Regulatory Guide 1.194, June 2003 (refer to Section 2.3.5.2.2)

During normal operation (CRVS Mode 1), both control room normal intakes are operational. Redundant PG&E Design Class I radiation monitors located at each control room normal intake have the capability of isolating the control room normal intakes on detection of high radiation and switching the control room ventilation system (CRVS) to Mode 4 operation (i.e., control room filtered intake and pressurization).

CRVS Mode 4 operation utilizes redundant PG&E Design Class I radiation monitors located at each control room 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 in accordance with Regulatory Guide 1.194, June 2003. 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 (refer to Section 2.3.5.2.2).

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 operation.

During normal operations, unfiltered air is drawn into the control room envelope (refer to Table 15.5-81) from the Unit 1 and Unit 2 normal intakes. In response to a control room radiation monitor or SIS, the control room 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 control room pressurization air intakes. The control room 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 unfiltered due to backdraft damper leakage, all pressurization flow is filtered.

The allowable methyl iodide penetration and filter bypass for the CRVS Mode 4 Charcoal Filter is controlled by Technical Specifications and the VFTP, and is 2.5% and <1%, respectively. In accordance with Generic Letter 99-02, June 1999 a safety factor of 2 is used in determining the charcoal filter efficiency for use in safety analyses (refer to Section 9.4.1 and Table 9.4-2. Thus the control room 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 Technical Specifications is a "penetration plus system bypass" < 1.0%. 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 control room air is also recirculated and a portion of the

recirculation flow filtered through the same filtration unit as the pressurization flow. Refer to Table 15.5-81 for a summary of recirculation flow rates.

Unfiltered inleakage into the control room during Mode 1 and Mode 4 is provided in Table 15.5-86 and includes ingress/egress inleakage based on the guidance provided in SRP 6.4.

For purposes of estimating the post-accident dose consequences, the control room 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 control room through the filtered emergency intakes; b) enters the control room as infiltration, c) enters the control room during operator egress/ingress, and d) enters the control room 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 control room.

The dose consequence analyses for the LOCA, MSLB, SGTR and the CREA, assume a LOOP concurrent with reactor trip.

In addition, and as noted in Section 15.5.1.2, 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 accident specific:

a. Automatic isolation of the control room 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 SIS processing time and a 10 second damper closure time.

b. Automatic isolation of the control room 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 control room ventilation dampers to re-align. The 2 second SIS 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 control room 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.

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

Table 15.5-81 lists key assumptions / parameters associated with control room design.

~~The information previously in this section has been moved to Section 15.5.8.4.~~

15.5.10 RADIOLOGICAL CONSEQUENCES OF CONDITION II FAULTS

15.5.10.1 Acceptance Criteria

The radiological consequences of accidents analyzed in Section 15.2 (or from other events involving insignificant core damage, but requiring atmospheric steam releases) shall not exceed the dose limits of 10 CFR ~~100.11~~50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

EAB and LPZ Dose Criteria

Regulatory Guide 1.183 does not specifically address Condition II scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 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 will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

(1) An individual located at any point on the boundary of the exclusion area for ~~the two hours immediately~~ any 2-hour period following the onset of the postulated fission product release shall not receive a ~~total~~ radiation dose ~~to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure~~in excess of 0.025 Sv (2.5 rem) TEDE.

(2) 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 total radiation dose ~~to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~ in excess of 0.025 Sv (2.5 rem) TEDE.

Control Room Dose Criteria

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

15.5.10.2 Identification of Causes and Accident Description

15.5.10.2.1 Activity Release Pathways

As reported in Section 15.2, Condition II faults are not expected to cause breach of any of the fission product barriers, thus preventing fission product release from the core or plant. Under some conditions, however, small amounts of radioactive isotopes could be released to the atmosphere following Condition II events as a result of atmospheric steam dumps required for plant cooldown. The particular Condition II events that are expected to result in some atmospheric steam release are:

(1) Loss of electrical load and/or turbine trip

(2) Loss of normal feedwater

(3) Loss of offsite power to the station auxiliaries

(4) Accidental depressurization of the main steam system

The amount of steam released following these events depends on the time relief valves remain open and the availability of condenser bypass cooling capacity.

The mass of environmental steam releases for the Loss of Load Event bound all Condition II events.

A LOL event is different from the Loss of Alternating Current (AC) power condition, 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 LOL event incorporates the assumption of Loss of offsite power to the station auxiliaries.

Although Regulatory Guide 1.183 does not provide specific guidance with respect to scenarios to be assumed to determine radiological dose consequences from Condition II events, the scenario outlined below for the LOL analysis is based on the conservative assumptions outlined in Regulatory Guide 1.183 for the MSLB, and was analyzed to bound all Condition II events that result in environmental releases.

Table 15.5-9A 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.

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.

15.5.10.2.2 Activity Release Transport Model

No melt or clad breach is postulated for the LOL (refer to Section 15.2.7). Thus, and in accordance with Regulatory Guide 1.183, Appendix E, item 2, the activity released is based on the maximum coolant activity allowed by the plant Technical Specifications, which focus on the noble gases and iodines. In accordance with Regulatory Guide 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.

Plant Technical Specification limits primary to secondary steam generator (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 Regulatory Guide 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 Regulatory Guide 1.183, the iodines are released to the environment via the via the main steam safety valves (MSSVs) and 10% atmospheric dump valves (ADVs) in proportion to the steaming rate and the inverse of a partition coefficient of 100. The iodine releases 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 coincident loss of offsite power. Consequently, the radioactivity release resulting from a LOL event is discharged to the environment from the steam generators via the MSSVs / 10% ADVs. The SG releases continue for 10.73 hours, at which time shutdown cooling is initiated via operation of the Residual Heat Removal (RHR) system, and environmental releases are terminated.

15.5.10.2.3 Offsite Dose Assessment

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 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 event at either unit is presented in Table 15.5-9.

15.5.10.2.4 Control Room Dose Assessment

The parameter values utilized for the control room in the accident dose transport model are discussed in Section 15.5.9. A summary of the critical assumptions associated with control room response and activity transport for the LOL event is provided below:

<u>Control Room Ventilation</u>

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

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

Due to the proximity of the MSSVs/10% ADVs to the control room normal intake of the affected unit, and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the affected unit (closest to the release point) will be insignificant. Therefore, only the unaffected unit's control room normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs (refer to Section 2.3.5.2.2 for detail).

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an LOL event at either unit are provided in Table 15.5-9B. The χ/Q values presented in Table 15.5-9B take into consideration the various release points-receptors applicable to the LOL to identify the bounding χ/Q values applicable to a LOL event at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2 and summarized in the notes of Tables 2.3-147 and 2.3-148.

The bounding Control Room dose following a LOL event at either unit is presented in Table 15.5-9.
~~The amount of radioactive iodine released depends on the amount of steam released and the iodine concentration in the steam generator water prior to the accident. An analysis of potential thyroid doses has been made over the full range of possible values of these two key parameters; the results are presented in Figures 15.5-2 through 15.5-5. As shown on the figures, the potential thyroid doses are higher with increasing steam releases and iodine concentrations. Figures 15.5-2 and 15.5-3 are results that assume Regulatory Guide 1.4, Revision 1, assumptions for post-accident meteorology and breathing rates (Design Basis Case Assumptions). As shown in Figure 15.5-2, approximately 1.6 x 10^6 lbm of steam is the maximum steam release expected for a full cooldown without any condenser availability, and a steam release of approximately 1 x 10^5 lbm would result from releasing only the contents of one steam generator due to a safety valve release or steam line break with condenser cooling available.~~

~~Figures 15.5-2 through 15.5-5 illustrate the range of possible thyroid doses from Condition II events. The highest anticipated doses would result from an event such as loss of electrical load, and the potential thyroid and whole body doses from this particular event have been analyzed using the EMERALD program. For both the design basis case and the expected case, it was assumed that 656,000 lbm of steam would be released to the atmosphere during the first 2 hours, and an additional 1,035,000 lbm would be released during the following 6 hours for a limiting total release of about 1.7E+06 lbm (see Table 6.4.2-1 of Reference 49 for a summary of OSG and RSG Condition II event steam releases). The assumptions used for meteorology, breathing rates, population density, and other common factors were described in earlier paragraphs. Note that the preceding steam release quantities are associated with the original steam generator (OSG) loss of load (LOL) analysis which provides the basis for the dose analysis of record. These values are greater than the replacement steam generator (RSG) LOL with Tavg and Tfeed Range analysis releases (651,000 lbm and 1,023,000 lbm, respectively) and are therefore bounding since total dose is proportional to total steam release.~~

~~For the design basis case, it was assumed that the plant had been operating continuously with 1 percent fuel cladding defects and 1 gpm primary-to-secondary leakage. For the expected case calculation, operation at 0.2 percent defects and 20 gallons per day to the secondary was assumed. In both cases, leakage of water from primary to secondary was assumed to continue during cooldown at 75 percent of the pre-accident rate during the first 2 hours and at 50 percent of the pre-accident rate~~

~~during the next 6 hours. These values were derived from primary-to-secondary pressure differentials during cooldown.~~

~~It was also conservatively assumed for both cases that the iodine partition factor in the steam generators releasing steam was 0.01, on a mass basis. In addition, to account for the effect of iodine spiking, fuel escape rate coefficients for iodines of 30 times the normal operation values given in Table 11.1-8 were used for a period of 8 hours following the start of the accident. Other detailed and less significant modeling assumptions are presented in Reference 4.~~

~~The resulting potential exposures from this type of accident are summarized in Table 15.5-9 and are consistent with the parametric analyses presented in Figures 15.5-2 through 15.5-5.~~

15.5.10.3 Conclusions

It can be concluded from the results discussed that the occurrence of any of the events analyzed in Section 15.2 (or from other events involving insignificant core damage, but requiring atmospheric steam releases) will result in insignificant radiation exposures and are bounded by the LOL event.

Additionally, the analysis demonstrates that the acceptance criteria are met as follows:

(1) The radiation dose to ~~the whole body and to the thyroid of~~ an individual located at any point on the boundary of the exclusion area for the ~~two-hours~~any 2-hour period ~~immediately~~ following the onset of the postulated fission product release is within 0.025 Sv (2.5 rem) TEDE ~~are insignificant~~ as shown in Table 15.5-9.

(2) The radiation dose to ~~the whole body and to the thyroid~~ of 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), is within 0.025 Sv (2.5 rem) TEDE ~~are insignificant~~ as shown in Table 15.5-9.

~~(2)~~(3) The radiation dose to an individual in the control room for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-9.

15.5.11 RADIOLOGICAL CONSEQUENCES OF A SMALL-BREAK LOCA

15.5.11.1 Acceptance Criteria

The radiological consequences of a small-break loss-of-coolant-accident (SBLOCA) shall not exceed the dose limits of 10 CFR ~~100.11~~ 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 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 will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

<u>EAB and LPZ Dose Criteria</u>

(1) 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 0.025 Sv (2.5 rem) TEDE.

(2) 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 total radiation dose -in excess of 0.025 Sv (2.5 rem) TEDE.

~~i. An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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 total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~(1) In accordance with the requirements of GDC 19, 1971, the dose to the control room operator under accident conditions shall not be in excess of 5 rem whole body or its equivalent to any part of the body (i.e., 30 rem thyroid and beta skin, Reference 51) for the duration of the accident.~~

15.5.11.2 Identification of Causes and Accident Description

As discussed in Section 15.3.1, a SBLOCA (defined in UFSAR Chapter 15.3.1 as a break that is large enough to actuate the emergency core cooling system), is not expected to cause fuel cladding failure. For this reason, the only activity release to the containment will be the dissolved noble gases and iodine in the reactor coolant water expelled from the pipe rupture. Some of this activity could be released to the containment atmosphere as the water flashes, and some of this amount could leak from the containment as a result of a rise in containment pressure.

The possible radiological consequence of this event is expected to be bounded by the "containment release" scenario of the CREA discussed in Section 15.5.23.

The dose consequences following a SBLOCA will be significantly less than a CREA since the CREA is postulated to result in 10% fuel damage, whereas the SBLOCA has no fuel damage.

As demonstrated in Table 15.5-52, the dose consequences at the EAB and LPZ following a CREA is within the acceptance criteria applicable to the SBLOCA.

~~The detailed description of the models used in calculating the potential exposures from a small LOCA is contained in Reference 4, and a general description is contained in Section 15.5.17 of this report. The specific assumptions used in the analysis are as follows:~~

~~(1) The fission product inventories, meteorological data, breathing rates, and population data are described in Sections 15.5.3, 15.5.5, 15.5.6, and 15.5.7, respectively. Other common assumptions are described in the~~

~~previous sections of 15.5.~~

~~(2) It has been assumed that all of the water contained in the RCS is released to the containment. For the design basis case, the reactor coolant activities associated with 1 percent defective cladding were used; and for the expected case, the reactor coolant activities associated with 0.2 percent defective cladding were used. These activities and concentrations are listed in Tables 11.1-11 and 11.1-12, and all models and assumptions used in determining these values are described in Section 11.1.~~

~~(3) Of the amounts of noble gases contained in the primary coolant, 100 percent is assumed to be released to the containment atmosphere at the time of the accident. For the iodines, it is assumed that only 10 percent of the dissolved iodine in the coolant is released to the containment atmosphere, due to the solubility of the iodine. It is assumed that the amounts of iodine in chemical forms that are not affected by the spray system are negligible. These release fractions are used for both the design basis case and the expected case.~~

~~(4) In addition, to account for the effect of iodine spiking, all of the activity released from the fuel up to 8 hours after the accident is assumed to be released to the containment. Of the amounts of noble gases released to the containment, 100 percent is assumed to be released to the containment atmosphere. For the iodines, it is assumed that only 10 percent of the iodines released to the containment are released to the containment atmosphere.~~

~~(5) The spray removal rates for the SBLOCA are assumed to be the same as those applicable for the large break LOCA as described in Section 15.5.17.~~

~~(6) The containment leakage rates in this analysis are also assumed to be the same as for the large break LOCA and are discussed in Section 15.5.17.~~

~~The resulting potential exposures are listed in Table 15.5-10 and demonstrate that all calculated doses are well below the guideline values specified in 10 CFR 100.11. Since the activity releases from this type of event will be significantly lower than those from a large break LOCA, any control room exposure which might occur would be well within the established criteria discussed in Section 15.5.17. In addition, because of significantly lower fission product releases to the sump and the absence of any zirconium-water reaction, the amounts of free hydrogen produced by sump radiolysis following a small LOCA would not be of concern.~~

15.5.11.3 Conclusions

~~The analysis demonstrates that the acceptance criteria are met as follows:~~

(1) ~~The radiation dose to the whole body and to the thyroid of an individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release are well within the dose limits of 10 CFR 100.11 as shown in Table 15.5-10.~~

(2) ~~The radiation dose to the whole body and to the thyroid of 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), are well within the dose limits of 10 CFR 100.11 as shown in Table 15.5-10.~~

(3) ~~Since the activity releases from the SBLOCA are less than those from a large-break LOCA (LBLOCA), any control room dose which might occur would be well within the established criteria discussed in Section 15.5.17.~~

On the basis of this conservative comparison approach, it is concluded that the dose consequences at the EAB and LPZ following a SBLOCA will remain within the acceptance criteria listed in Section 15.5.11.1.

15.5.12 RADIOLOGICAL CONSEQUENCES OF MINOR SECONDARY SYSTEM PIPE BREAKS

15.5.12.1 Acceptance Criteria

The radiological consequences of accidents analyzed in Section 15.3 such as minor secondary system pipe breaks shall not exceed the dose limits of ~~10 CFR 100.11 as outlined below:~~10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 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 will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

EAB and LPZ Dose Criteria

(1) 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 0.025 Sv (2.5 rem) TEDE.

(2) 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 total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

~~An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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 total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

15.5.12.2 Identification of Causes and Accident Description

The effects on the core of sudden depressurization of the secondary system caused by an accidental opening of a steam dump, relief or safety valve were described in Section 15.2 and apply also to the case of minor secondary system pipe breaks. As shown in that analysis, no core damage or fuel rod failure is expected to occur. In Section 15.~~5.18~~4.2, analyses are presented that show the effects on the core of a major steam line break, and, in this case also, no fuel rod failures are expected to occur.

The analyses presented in Section 15.3.2 demonstrate that a departure from nucleate boiling ratio (DNBR) of less than the safety analysis limit will not occur anywhere in the core in the event of a minor secondary system pipe rupture.

The steam releases following a minor secondary line break is expected to be significantly less than that associated with a main steam line break.

As demonstrated in Table 15.5-34, the dose consequences at the EAB and LPZ following a MSLB is within the acceptance criteria applicable to the minor secondary line break.

~~The possible radiological consequences of this event, due to the release of some steam that might contain radioactive iodines, are discussed in Section 15.5.10. The resulting thyroid doses are presented parametrically in Figures 15.5-2 through 15.5-5 as a function of quantity of steam released and secondary system activity. In the event that a complete plant cooldown without condenser cooling capacity is necessary following the break, the potential exposures would be the same as those reported in Table 15.5-9 for loss of electrical load.~~

15.5.12.3 Conclusions

On the basis of this conservative comparison approach, it is concluded that the dose consequences at the EAB and LPZ following a minor secondary system pipe rupture will remain within the acceptance criteria listed in Section 15.5.12.1.

~~On the basis of the discussed results, it can be concluded that the potential exposures following a minor secondary system pipe rupture would be insignificant.~~

~~Additionally, the analysis demonstrates that the acceptance criteria are met as follows:~~

~~The radiation dose to the whole body and to the thyroid of an individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release are insignificant as shown in Table 15.5-9.~~

~~The radiation dose to the whole body and to the thyroid of 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), are insignificant as shown in Table 15.5-9.~~

15.5.13 RADIOLOGICAL CONSEQUENCES OF INADVERTENT LOADING OF A FUEL ASSEMBLY INTO AN IMPROPER POSITION

15.5.13.1 Acceptance Criteria

Fuel assembly loading errors shall be prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses supporting Section 15.3.3 shall confirm that no events leading to radiological consequences shall occur as a result of loading errors.

15.5.13.2 Identification of Causes and Accident Description

Fuel and core loading errors such as inadvertently loading one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or loading a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. The inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods is also included among possible core loading errors. Because of margins present, as discussed in detail in Section 15.3.3, no events leading to radiological consequences are expected as a result of loading errors.

15.5.13.3 Conclusions

Because of margins present, as discussed in detail in Section 15.3.3, no events leading to radiological consequences are expected as a result of loading errors.

15.5.14 RADIOLOGICAL CONSEQUENCES OF COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.5.14.1 Acceptance Criteria

The radiological consequences of small amounts of radioactive isotopes that could be released to the atmosphere as a result of atmospheric steam dumping required for plant cooldown following a complete loss of forced reactor coolant flow shall not exceed the dose limits of 10 CFR ~~100.11 as outlined below:~~50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 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 will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

EAB and LPZ Dose Criteria

(1) 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 0.025 Sv (2.5 rem) TEDE.

(2) 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 total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

~~An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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 total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

15.5.14.2 Identification of Causes and Accident Description

As discussed in Section 15.3.4, a complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps (RCPs). If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature.

The analysis performed and reported in Section 15.3.4 has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit during the transient, and thus there is no cladding damage or release of fission products to the RCS. ~~For this reason, this accident has no significant radiological effects.~~

The possible radiological consequence of a complete loss of forced reactor coolant flow is expected to be bounded by the conservative Loss-of-Load scenario with a coincident Loss of offsite power described in Section 15.5.10.

As demonstrated in Table 15.5-9, the dose consequences at the EAB and LPZ following a Loss of Load is within the acceptance criteria applicable to the complete loss of forced reactor coolant flow.

15.5.14.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ following a complete loss of forced reactor coolant flow will remain within the acceptance criteria listed in Section 15.5.14.1.~~The analysis described in Section 15.3.4 demonstrates that there are no significant environmental effects of the Complete Loss of Forced Reactor Coolant Flow event. Therefore, the acceptance criteria are met as follows:~~

~~The radiation dose to the whole body and to the thyroid of an individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release are insignificant.~~

~~The radiation dose to the whole body and to the thyroid of 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), are insignificant.~~

15.5.15 RADIOLOGICAL CONSEQUENCES OF AN UNDERFREQUENCY ACCIDENT

15.5.15.1 Acceptance Criteria

The radiological consequences of small amounts of radioactive isotopes that could be released to the atmosphere as a result of atmospheric steam dumping required for plant cooldown following an underfrequency accident shall not exceed the dose limits of 10 CFR ~~100.11 as outlined below:~~ 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 as outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 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 will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

EAB and LPZ Dose Criteria

(1) 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 0.025 Sv (2.5 rem) TEDE.

(2) 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 total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

~~An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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 total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

15.5.15.2 Identification of Causes and Accident Description

A transient analysis for this unlikely event ~~has been carried out~~is discussed in Section 15.3.4. The analysis demonstrates that for an underfrequency accident, the DNBR does not decrease below the safety analysis limit during the transient, and thus there is no cladding damage or release of fission products to the RCS. However, small amounts of radioactive isotopes could be released to the atmosphere as a result of atmospheric steam dumping required for plant cooldown.

The possible radiological consequence of this event is expected to be bounded by the conservative Loss-of-Load scenario with a coincident Loss of offsite power described in Section 15.5.10.

As demonstrated in Table 15.5-9, the dose consequences at the EAB and LPZ following a Loss of Load is within the acceptance criteria applicable to an underfrequency accident.
~~A detailed discussion of the potential environmental consequences of accidents involving atmospheric steam dumping is presented in Section 15.5.10. From the parametric analyses presented in that section, the potential exposures from an underfrequency accident are given in Table 15.5-11. On the basis of these potential exposures, it can be concluded that, although very unlikely, the occurrence of this accident would not cause undue risk to the health and safety of the public.~~

15.5.15.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ following a complete loss of forced reactor coolant flow will remain within the acceptance criteria listed in Section 15.5.15.1. ~~On the basis of the potential~~

~~exposures discussed, it can be concluded that, although very unlikely, the occurrence of this accident would not cause undue risk to the health and safety of the public.~~

~~Additionally, the analysis demonstrates that the acceptance criteria are met as follows:~~

~~The radiation dose to the whole body and to the thyroid of an individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release are insignificant as shown in Table 15.5-11.~~

~~The radiation dose to the whole body and to the thyroid of 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), are insignificant as shown in Table 15.5-11.~~

15.5.16 RADIOLOGICAL CONSEQUENCES OF A SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT FULL POWER

15.5.16.1 Acceptance Criteria

The radiological consequences of a single rod cluster control assembly withdrawal shall not exceed the dose limits of 10 CFR ~~100.11 as outlined below:~~50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below:

Regulatory Guide 1.183 does not specifically address Condition III scenarios. However, per Regulatory Guide 1.183, Section 1.2.1, a full implementation of AST allows a licensee to utilize the dose acceptance criteria of 10 CFR 50.67 in all dose consequence analyses. In addition, Section 4.4 of Regulatory Guide 1.183 indicates that for events with a higher probability of occurrence than those listed in Table 6 of Regulatory Guide 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 will be limited to the lowest value reported in Table 6, i.e., a small fraction (10%) of the limit imposed by 10 CFR 50.67.

EAB and LPZ Dose Criteria

(1) 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 0.025 Sv (2.5 rem) TEDE.

(2) 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 total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

~~An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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 total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

15.5.16.2 Identification of Causes and Accident Description

A complete transient analysis of this accident is presented in Section 15.3.5. For the condition of one rod cluster control assembly (RCCA) fully withdrawn with the rest of the bank fully inserted, at full power, an upper bound of the number of fuel rods experiencing DNBR less than the safety analysis limit is 5 percent of the total fuel rods in the core.

The possible radiological consequence of this event is expected to be bounded by the CREA discussed in Section 15.5.23.

The dose consequences following 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.

As demonstrated in Table 15.5-52, the dose consequences at the EAB and LPZ following a CREA is within the acceptance criteria applicable to the condition of one rod cluster control assembly fully withdrawn with the rest of the bank fully inserted, at full power. ~~A detailed discussion of the potential radiological consequences of accidents involving small amounts of fuel rod failure is included in Section 15.5.21. From the parametric analyses presented in that section, the potential exposures from an RCCA withdrawal at full power resulting in 5 percent fuel failure are given in Table 15.5-12.~~

15.5.16.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ following the condition of one rod cluster control assembly fully withdrawn with the rest of the bank fully inserted, at full power will remain within the acceptance criteria listed in Section 15.5.16.1.
~~On the basis of the potential exposures discussed, it can be concluded that the occurrence of this accident would not cause undue risk to the health and safety of the public.~~

~~Additionally, the analysis demonstrates that the acceptance criteria are met as follows:~~

~~The radiation dose to the whole body and to the thyroid of an individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release are insignificant as shown in Table 15.5-12.~~

~~The radiation dose to the whole body and to the thyroid of 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), are insignificant as shown in Table 15.5-12.~~

15.5.17 RADIOLOGICAL CONSEQUENCES OF MAJOR RUPTURE OF PRIMARY COOLANT PIPES

~~Various aspects of the radiological consequences of a large-break loss-of-coolant accident (LBLOCA) are presented in this section.~~

15.5.17.1 Acceptance Criteria

The radiological consequences of a LOCA shall not exceed the dose limits of 10 CFR 50.67, and will meet the dose acceptance criteria of Regulatory Guide 1.183, July 2000 and outlined below:

<u>EAB and LPZ Dose Criteria</u>

(1) 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 0.25 Sv (25 rem) TEDE.

(2) 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 total radiation dose in excess of 0.25 Sv (25 rem) TEDE.

Control Room Dose Criteria

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

Technical Support Center Dose Criteria

The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, as amended by Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident.

~~(1) The radiological consequences of a major rupture of primary coolant pipes shall take into consideration fission product releases due to leakage from the containment, post-LOCA recirculation Loop leakage in the Auxiliary Building (inclusive of a residual heat removal (RHR) pump seal failure resulting in a 50 gpm leak for 30 minutes starting at T=24 hrs post-LOCA), and containment shine.~~

~~(2) The radiological consequences of a major rupture of primary coolant pipes shall not exceed the dose limits of 10 CFR 100.11 as outlined below:~~

~~i. An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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 total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~(3) In accordance with the requirements of GDC 19, 1971, the dose to the control room operator under accident conditions shall not be in excess of 5 rem whole body or its equivalent to any part of the body (i.e., 30 rem thyroid and beta skin, Reference 51) for the duration of the accident.~~

~~(4) In the event controlled venting of the containment is implemented post-LOCA using the containment hydrogen purge system (serves as a back-up capability for hydrogen control to the hydrogen recombiners), an individual located at any point on the boundary of the exclusion area, 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 total radiation dose to the whole body in excess of 0.5 rem/year in accordance with 10 CFR Part 20.~~

15.5.17.2 Identification of Causes and Accident Description

15.5.17.2.1 ~~Basic Events and Release Fractions~~Activity Release Pathways

The accidental rupture of a main coolant pipe is the event assumed to initiate a ~~LB~~large break LOCA. Analyses of the response of the reactor system, including the emergency core cooling system (ECCS), to ruptures of various sizes have been presented in 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. As a result of the increase in cladding temperature and the rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Following the cladding failure, some activity would be released to the primary coolant and subsequently to the inside of the containment building. Active mechanisms include radioactive particulate and iodine removal by the containment sprays inclusive of the containment air mixing provided by the CFCUs. Section 6.2 describes the design and operation of the CSS and the CFCUs. ~~Because of the pressurization of the containment building by the primary coolant water escaping from the pipe break, some of the volatile radioactive iodines and noble gases could leak from the containment building to the atmosphere.~~

~~It is not expected that a significant amount of organic iodine would be liberated from the fuel as a result of a LBLOCA. This conclusion is based on the results of fuel meltdown experiments conducted by the Oak Ridge National Laboratory. The fraction of the total iodine that is released in organic forms is expected to be on the order of 0.2 percent, or less, since the rate of thermal radiolytic decomposition would exceed the rate of production.~~

~~Organic compounds of iodine can be formed by reaction of absorbed elemental iodine on surfaces of the containment vessel. Experiments have shown that the rate of formation is dependent on specific conditions such as the concentration of iodine, concentration of impurities, radiation level, pressure, temperature, and relative humidity. The rate of conversion of airborne iodine is proportional to the surface-to-volume ratio of the enclosure, whether the process is limited to diffusion to the surface or by the reaction rate of the absorbed iodine. The observed yields of organic iodine as a function of aging time in various test enclosures, with various volume-to-surface area ratios, were extrapolated to determine the values for the DCPP containment vessel. The iodine conversion rates predicted in this manner did not exceed 0.0005 percent of the atmospheric iodine per hour.~~

~~The potential exposures following the postulated sequence of events in LBLOCAs have been analyzed for two cases. In the expected case, it has been assumed that the entire inventory of volatile fission products contained in the pellet-cladding gap spaces is~~

~~released to the coolant during the time the core is being flooded by the ECCS. Of this gap inventory, 25 percent of the iodines and 100 percent of the noble gases are considered to be released to the containment atmosphere immediately following the pipe rupture. In this respect, the expected case does contain some degree of conservatism since the ECCS is designed to prevent gross cladding damage. In accordance with the experimental data reported in the previous paragraph, the fraction of iodine that is released in organic form is assumed to be 0.2 percent, and the production rate of organic forms is considered negligible. The iodine plateout rates are negligible (Reference 10) compared to the spray washout rates and are assumed to be zero. The particulate fraction of iodine is also assumed to be zero for the expected case since this fraction is small and the spray removal rates for particulates is large as shown in Reference 10.~~

~~For the design basis LOCA, it has been assumed that 25 percent of the equilibrium radioactive iodine inventory in the core is immediately available for leakage from the reactor containment. Ninety-one percent of this 25 percent is assumed to be in the form of elemental iodine, 4 percent of this 25 percent is in the form of organic iodides, and 5 percent of this 25 percent is in the form of particulate iodine. In addition, 100 percent of the noble gas inventory in the core is assumed to be immediately released to the containment building. As discussed in earlier paragraphs, releases of these magnitudes are not expected to occur, even if the ECCS does not perform as expected. An analysis using these assumptions is presented because these values are considered acceptable for a design basis analysis in Regulatory Guide 1.4, Revision 1.~~

Regulatory Guide 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.

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 Standard Review Plan Section 15.6.5, Appendix B (Reference 77), as well as Regulatory Guide 1.183, July 2000, inclusion of this leakage path in dose consequences is not required. However, as noted in the following Sections, the RHR

pump seal failure resulting in a "filtered" release is DCPP's licensing basis with respect to passive single failure.

- 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. A review of the equipment in the RHR system loop and the CSS loop indicates that the largest leakage would result from the failure of an RHR pump seal. Evaluation of RHR pump seal leakage rate, assuming only the presence of a seal retention ring around the pump shaft, shows that flows less than 50 gpm would result (Chapter 6). Circulation loop piping leaks, valve packing leaks, and flange gasket leaks are much smaller and less severe than an RHR pump seal failure leak.

- 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 Miscellaneous Equipment Drain Tank (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 LOCA dose consequence analysis follows the requirements provided in the pertinent sections of Regulatory Guide 1.183 including Appendix A. Table 15.5-23A lists the key assumptions / parameters utilized to develop the radiological consequences following a LOCA at either unit.

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

15.5.17.2.2 Activity Release Transport Model~~Spray System Iodine Removal Rates~~

~~The containment spray system (CSS) is described in detail, along with a performance analysis, in Sections 6.2.2 and 6.2.3. The performance analysis includes the representation of the spatial distribution of droplets and iodine in the containment, as well as drop coalescence and other effects.~~

~~For the expected case analyses, the CSS is assumed to function with both spray pumps operating, giving an effective elemental iodine removal coefficient of 92 per hour. On the basis of experiments at Battelle, as described in Reference 10, the spray removal rate for organic iodides was assumed to be 0.058 per hour.~~

~~For the design basis case, it is assumed that one of the two spray pumps fails to operate, and the elemental iodine removal coefficient is reduced to 31 per hour. This assumption is consistent with the value of 32 per hour used in the PSAR analysis. It has also been assumed, for the design basis case, that the CSS has no effect on the organic and particulate iodines.~~

~~Although a subsequent safety evaluation showed that the Design Case coefficient of 31 per hour (for 2600 gpm spray header flow) should be reduced to approximately 29 per hour (for 2466 gpm spray header flow), the potential offsite dose increase due to this change is extremely small and can be considered insignificant (Reference 39).~~

15.5.17.2.2.1 Containment Pressure /Vacuum Relief Line Release

In accordance with Regulatory Guide 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 Technical Specifications primary coolant equilibrium activity (refer to Table 15.5-78). 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 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.

15.5.17.2.2.2 Containment Leakage

The inventory of fission products in the reactor core available for release into the containment following a LOCA is provided in Table 15.5-77 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 15.5-77 provide information on isotopes used to estimate the inhalation and submersion doses following a LOCA, vs isotopes that are considered to estimate the post-LOCA direct shine dose.

Per Regulatory Guide 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.

In accordance with Regulatory Guide 1.183:

a. 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).
b. The core inventory release fractions, by radionuclide groups, for the gap and early in-vessel damage are as follows:

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

Note: Footnote 10 criterion is met in that peak fuel rod burnup is limited to 62,000 MWD/MTU.

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 Regulatory Guide 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 6.2.3.3.7, the 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 and spray removal are credited to deplete the inventory of fission products airborne in containment.

Containment spray in the injection and recirculation mode 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 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.

a. Containment 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 within twelve (12) minutes after injection spray is terminated. Thus, based on single train operation, containment spray in the recirculation mode is initiated at 4518 seconds, and terminated 5 hours later at 22,518 seconds. In summary, containment spray operation (injection plus recirculation) is credited for 6.25 hrs post-LOCA, with a twelve minute gap after injection spray is terminated.

b. Containment Spray Coverage: As discussed in Section 6.2.3.3.7.1, the containment sprays are estimated to effectively cover 82.5% of the containment free volume during the containment spray injection as well as spray recirculation mode.

c. Mixing between Sprayed and Unsprayed Regions of Containment: 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 CFCU with a total volumetric flow rate that addresses surveillance margins and uncertainty, between the unsprayed regions and sprayed regions. 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 the major openings in the containment structure, and various active and passive mixing mechanisms, results in the conclusion that following a LOCA, credit can be taken for a) the entire flowrate provided by each operating CFCU to support mixing between the sprayed and unsprayed regions, and b) homogeneous mixing within the sprayed and unsprayed regions, of the volume of air transferred from one region to the other due to CFCU operation. In accordance with Regulatory Guide 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.

d. Fission Product Removal: The fission product removal coefficients developed for the LOCA reflect the following guidance documents:

    i. Elemental iodine removal coefficients are calculated using guidance provided in Standard Review Plan Section 6.5.2, Revision 4 (Reference 80) which is invoked by Regulatory Guide 1.183, Appendix A, Section 3.3

    ii. Time dependent particulate aerosol removal coefficients are estimated using Regulatory Guide 1.183, Appendix A, Section 3.3, which permits the use of time-dependent particulate aerosol removal coefficients by invoking NUREG/CR 5966, June 1993 (Reference 81), 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. There are several aerosol mechanics phenomena that promote the depletion of aerosols from the containment atmosphere. 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 sprays and diffusiophoresis (aerosol removal due to steam condensation). Computer program SWNAUA is used to develop the time dependent particulate aerosol removal coefficients which reflect the effect of diffusiophoresis and sprays. Gravitational settling is considered only in the unsprayed region.

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 6.2.3.3.7.2. The total elemental iodine and particulate removal coefficients in the sprayed and unsprayed region of the containment as a function of time are summarized in Table 6.2-32.

In summary, 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 during the containment spray injection and recirculation mode. 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 NUREG-0800, Standard Review Plan 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 in 6.2.3.3.7.2, the wall deposition removal coefficient for elemental iodine has been calculated with the model provided in NUREG-0800, 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 Standard Review Plan Section 6.5.2 and is limited to a DF of 200.

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').

~~15.5.17.2.3 Offsite Exposures from Containment Leakage~~

~~As a result of the pressurization of the containment following a LOCA, there is a possibility of containment leakage during the time that the containment pressure is above atmospheric. For the design basis case, the leakage rate has been assumed to be 0.1 percent per day for the first 24 hours following the accident, and 0.05 percent per day after the first day. These assumed rates are consistent with the Technical Specifications (Reference 22) limit, the assumed rates considered acceptable in Regulatory Guide 1.4, Revision 1, and the values assumed in the PSAR analyses.~~

~~For the expected case, the containment leakage rates used are 0.05 percent per day for the first day and 0.025 percent per day for the periods after 1 day. These rates were determined from averages of the actual predicted containment pressures presented in previous sections, with the assumption that some of the heat removal systems do not function at full capacity.~~

~~In this regard, the leakage rates assumed for the expected case analysis retain some degree of conservatism since the containment heat removal systems are designed to reduce the containment pressure to atmospheric following the initial pressure rise, thus terminating the leakage.~~

~~15.5.17.2.4 Containment Leakage Exposure Sensitivity Study~~

~~HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.~~

~~Sensitivity studies were performed to illustrate the dependence of the thyroid exposures on the spray system removal constant and the fraction of nonremovable iodines present in the containment. The results of these studies are shown in Figures 15.5-6, 15.5-7, and 15.5-8. The thyroid exposures, normalized to the exposure for zero spray removal constant, are shown as a function of spray constant in Figure 15.5-6, for a fixed fraction of nonremovable iodine forms of 15 percent. In Figures 15.5-7 and 15.5-8 thyroid exposures are plotted as a function of two parameters: the spray removal constant and the percent of nonremovable iodines. To determine an absolute exposure (rem) from Figure 15.5-7, the normalized exposure should be multiplied by 940.9 rem, which is the reference 2 hour-800 meter exposure for the design basis case with a zero spray constant and a zero nonremovable fraction. To determine an absolute exposure (rem) from Figure 15.5-8, the normalized exposure should be multiplied by 197.4 rem, which is the reference 30-day-10,000 meter exposure for the design basis case with a zero spray constant and a zero nonremovable fraction. As shown in these figures, combinations of these parameters that result in normalized exposures below the criterion line would result in a calculated absolute exposure less than the 300 rem~~

~~*guideline level specified in 10 CFR Part 100.*~~

~~15.5.17.2.5 Radiological Consequences with DF of 100~~

~~The design basis LOCA was reviewed to evaluate potential differences in the offsite radiological dose consequences using a containment decontamination factor of 100 and a containment mixing flowrate of 94,000 cfm.~~

~~A containment mixing rate of 94,000 cfm corresponds with our current minimum design basis operation of two containment fan cooler units (CFCU). Calculations were based on reload fuel. A containment spray delay of 80 seconds was used. The radionuclide inventory source terms for the various fuel conditions were calculated using the ORIGEN-2 computer code with a power level of 3580 MWt. The radionuclide atmospheric releases and offsite doses were calculated with the LOCADOSE computer code.~~

~~Calculations were made relative to 10 CFR 100.11 requirements for offsite doses, at the 800 meter exclusion area boundary (EAB) at 2 hours and the 10,000 meter low population zone (LPZ) at 30 days, from post-LOCA containment leakage.~~

~~Table 15.5-75 presents the calculated offsite dose consequences from post-LOCA from various pathways. The limiting dose is the thyroid at the EAB. For the containment leakage pathway, the maximum thyroid dose of 107.06 rem exceeds the original design basis LOCA thyroid dose of 95.9 rem in Table 15.5-23. For the pre-existing small leakage, the EAB thyroid dose is 8.22 rem. These doses are comparable with the corresponding original design basis LOCA large leakage and small leakage cases doses.~~

~~All doses are within the 10 CFR 100.11 guidelines.~~

~~15.5.17.2.6 Offsite Exposures from Containment Shine~~

~~The site boundary 30-day DBA exposure from direct containment gamma radiation (containment shine) is estimated to be 0.0048 rem. Containment shine is a function of the activity present in the containment atmosphere. The EMERALD computer code was used to calculate the post-accident containment activity time-history, and the ISOSHLD II computer code was then used to calculate the containment shine exposure. The shine exposure model assumes 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 site boundary receptor point is assumed to be 800 meters from the containment structure.~~

~~15.5.17.2.7 Offsite Population Exposures from Containment Leakage~~

~~The calculated population exposures for the design basis case assumptions, and for the expected case, are summarized in Table 15.5-23. These whole body population~~

~~exposures do not include the effects of any population redistribution due to evacuation. These exposures were calculated using the EMERALD computer code. The atmospheric dilution factors and population distribution utilized in the population exposure calculations are discussed in Section 15.5.5.~~

~~15.5.17.2.8 Offsite Exposures from Post-LOCA Recirculation Loop Leakage in the Auxiliary Building Reactor coolant water that collects in the containment recirculation sump after a LOCA would contain radioactive fission products.~~

15.5.17.2.2.3 ESF System Leakage Outside Containment

The fluid that collects in the containment recirculation sump after a LOCA (i.e., the fluid contents of reactor coolant system, the RWST, the NaOH tank and the accumulators) contain radioactive fission products that has been released from the core as a result of the LOCA. ~~Because containment recirculation sump water is circulated outside the containment, problems of potential exposure due to post-LOCA operation of external circulation loops with leakage have been evaluated.~~

~~Reactor coolant water, ECCS injection water, and containment spray water accumulate in the containment recirculation sump following a LOCA. Containment~~ The containment recirculation sump water is circulated by the RHR pumps, cooled via the RHR heat exchangers, returned to the containment via the RHR system piping and the CSS piping ~~(if recirculation spray is used)~~, passed through the RCS and the containment spray nozzles ~~(if recirculation spray is used)~~, and finally returned to the containment recirculation sump. In the event of circulation loop leakage in the auxiliary building, post-LOCA activity has a pathway to the atmosphere.

~~An illustration of this pathway for a small leak is given in Figure 15.5-9. For the small leakage situation, fission products in the leakage water are exposed to auxiliary building ventilation air flow for a long period of time. Thus, for the small leakage situation, all activity released to the auxiliary building would be released to the auxiliary building air, i.e., no credit for liquid-gas partitioning.~~

~~An illustration of post-LOCA activity pathway for a large leak is given in Figure 15.5-10. For the large leakage situation, fission products in the leakage water are exposed to auxiliary building ventilation air flow for a short period of time. Thus, most of the activity released to the auxiliary building would be transferred to the floor drain receiver tank, i.e., credit for liquid-gas partitioning.~~

The complete RHR system and CSS description, including ~~estimates of leakage~~, detection of leakage, equipment isolation, and corrective maintenance, are contained in Sections 5.5.6 and 6.2.2, respectively.

In accordance with Regulatory Guide 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 Guide 1.183, the 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.

ESF leakage is assumed to occur at initiation of the recirculation mode for safety injection. Since the maximum temperature of the recirculation fluid supports a flash fraction less than 10%, per Regulatory Guide 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.

The environmental release of ESF system leakage can occur via the 2 pathways listed below.

a. <u>Environmental release of ESF System leakage via the plant vent</u>: 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 requirements of Regulatory Guide 1.183, the analysis addresses an ESF leakage of 240 cc/min in the auxiliary building. The areas where these components are located are covered by the PG&E Design Class I ABVS 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.

b. <u>Environmental release of ESF System leakage via Containment Penetration Area GE and Areas GW & FW</u>: 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 requirements of Regulatory Guide 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 & FW.

~~Post-LOCA auxiliary building loop leakage exposures were calculated for four different leakage cases:~~

~~(1) Expected small leakage case~~

~~(2) Expected large leakage case~~

~~(3) DBA small leakage case~~

~~(4) DBA large leakage case~~

~~Assumptions and numerical values used to calculate loop leakage exposures are listed in Table 15.5-24. Table 15.5-63 shows the results of the calculations based on these assumptions. Because an insignificant amount of noble gases would be in the containment recirculation sump water, the whole body exposures are negligible.~~

~~One possible approach to the evaluation of offsite exposures from post-LOCA recirculation loop leakage would include the following assumptions:~~

~~(1) A LOCA as an initiating event~~

~~(2) Failure of two ECCS trains resulting in gross fuel damage: Release of 50 percent of core iodine inventory and 100 percent of core noble gas inventory to the containment~~

~~(3) Failure of an RHR pump seal, resulting in the release of a significant amount of the above containment activity to the auxiliary building~~

~~(4) Failure of the passive auxiliary building charcoal filters resulting in the unfiltered release of iodine fission products to the environment~~

~~The assumption of this sequence of failures for analysis of offsite exposures, however, would be requiring plant design features in excess of the current guides and regulations, and in particular the requirements of ANS Standard N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Power Plants. (See proposed addendum to ANS Standard N18.2, Single Failure Criteria for Fluid Systems (Reference 16)).~~

~~Applying the proposed standard to post-LOCA recirculation loop leakage the LBLOCA was assumed as the initiating event:~~

> ~~"The unit shall be designed to tolerate an initiating event which may be a single active or passive failure in any system intended for use during normal operation."~~

~~The ECCS was assumed to function properly, as required by the ECCS acceptance criteria, preventing gross fuel damage. Although meeting these criteria is expected to preclude gross cladding damage, it was assumed for this analysis that 100 percent of~~

~~the gap iodine and noble gas inventories were released to the containment recirculation sump.~~

~~For the large leakage cases,~~

15.5.17.2.2.4 RHR Pump Seal Failure

~~failure~~ Failure of an RHR pump seal was assumed to be ~~as~~ the worst case single failure ~~and can~~to be tolerated without loss of the required functioning of the RHR system, as was required by the following clauses in the ~~proposed~~ addendum to the ANS Standard N18.2 proposed at the time of original license:

> "Fluid systems provided to mitigate the consequences of Condition III and Condition IV events shall be designed to tolerate a single failure in addition to the incident which requires their function, without loss of the function to the unit.
>
> "A single failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions when called upon. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against a single failure if neither (a) a single failure of any active component (assuming passive components function properly); nor (b) a single failure of a passive component (assuming active components function properly) results in a loss of the safety function to the nuclear steam electric generating unit.
>
> "An active failure is a malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function upon demand.
>
> "Examples of active failures include the failure of a valve or a check valve to move to its correct position, or the failure of a pump, fan or diesel generator to start.
>
> "Spurious action of a powered component originating within its actuation system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious action.
>
> "A passive failure is a breach of the fluid pressure boundary or blockage of a process flowpath."

~~For the expected and DBA large leakage cases, the~~The failure of auxiliary building charcoal filters, a second failure, was not assumed, in accordance with the standard.

~~For the expected and DBA small leakage cases, failure of auxiliary building charcoal filters was assumed as the single failure and can be tolerated without loss of the required function of the auxiliary building ventilation system, which provides cooling for~~

~~ECCS components.~~

~~For the long-term small leakage cases, the charcoal filters are not needed to reduce exposures below the guideline values given in 10 CFR Part 100. In any case, the fans in the ventilation system are redundant, and only the passive charcoal beds themselves are not redundant.~~

~~For the expected small and large leakage cases, it was assumed that two ECCS trains, five fan coolers, and two containment spray trains functioned. For the DBA small and large leakage cases, it is assumed that two ECCS trains, two fan coolers, and one containment spray train functioned. The DBA assumptions result in high containment recirculation sump water temperatures and minimum containment recirculation sump water pHs.~~

~~For all four circulation loop leakage cases it was assumed that 100 percent of the gap iodine inventory was deposited in containment recirculation sump water.~~

~~For the expected small and large leakage cases, the assumed gap iodine inventories are listed in Table 11.1-7. The expected case gap iodine was assumed to be only elemental iodine. For the DBA small and large leakage cases, the assumed gap iodine inventories are based on release fractions given in Safety Guide 25, March 1972 (Reference 23). The DBA case gap iodine was assumed to be 99.75 percent elemental iodine and 0.25 percent organic iodine per Safety Guide 25, March 1972.~~

~~Radiological decay of activity in the containment recirculation sump was assumed for all leakage cases for both the time periods before and during loop leakage. No credit was taken for cleanup of activity in the containment recirculation sump.~~

~~Reactor coolant water, accumulator water, and refueling water storage tank (RWST) water make up the total volume of water in which activity is deposited. Consideration of emergency core cooling injection flowrates and containment spray injection flowrates yields the volume of RWST water (Chapter 6). Table 15.5-24 lists the assumed volume of water in which activity is deposited for the four leakage cases. For the large leakage cases, the volume of diluting water was taken as the volume when the leakage began. No credit was taken for the extra diluting water added from the RWST during the 30-minute leakage period.~~

~~Sodium hydroxide spray additive will provide for an increased pH in the containment recirculation sump water. Consideration of emergency core cooling injection flowrates and containment spray injection flowrates yields the pH of the containment recirculation sump water (Chapter 6). Table 15.5-24 lists the assumed pH of recirculation loop leakage water for the four leakage cases. For the large leakage cases, the pH was taken as the pH when the leakage began. No credit was taken for the extra sodium hydroxide in the spray water added during the 30-minute leakage period.~~

~~The design evaluation conducted for the containment functional design yields the~~

~~temperature of containment recirculation sump water as a function of time (Chapter 6). Table 15.5-24 lists the assumed temperature of recirculation loop leakage water for the four leakage cases. For the large leakage cases, the water temperature was taken as the temperature when the leakage began. No credit was taken for the decrease of water temperature during the 30-minute leakage period.~~

A review of the equipment in the RHR system loop and the CSS loop indicates that the largest leakage would result from the failure of an RHR pump seal. Evaluation of RHR pump seal leakage rate, assuming only the presence of a seal retention ring around the pump shaft, shows that flows less than 50 gpm would result (refer to Section 3.1 and Chapter 6). Circulation loop piping leaks, valve packing leaks, and flange gasket leaks are much smaller and less severe than an RHR pump seal failure leak. ~~Leakage from these components during normal post-LOCA operation of the RHR system loop and the CSS loop is estimated to be 1910 cc/hr (Chapter 6).~~ On this basis, a 50 gpm leakrate was assumed for ~~both the expected large leakage case and the DBA large leakage~~LOCA. ~~case, and a 1910 cc/hr leakrate was assumed for both the expected small leakage case and the DBA small leakage case.~~

For the DBA ~~large leakage case, recirculation loop~~LOCA pump seal leakage was assumed to commence 24 hours after the start of the ~~LB~~LOCA. This assumption is consistent with the discussion in Sections 3.1.1.1 and 6.3.3.5.3, and with the guidance in Standard Review Plan 15.6.5, Appendix B. In this context, the limiting recirculation loop long term passive failure is 50 gpm leakage at 24 hours after the start of the LBLOCA.

Evaluation of an RHR pump seal failure shows that the failure could be detected and the pump isolated well within 30 minutes (Chapter 6). ~~A~~ Thus a leakage duration of 30 minutes is conservatively assumed for ~~both the expected and~~ the DBA LOCA~~large leakage cases~~.

~~A leakage duration of 30 days is assumed for both the expected and DBA small leakage cases. As discussed earlier, the auxiliary building DF is a function of the Partition Factor (PF) for a particular isotope (Equation 15.5-7).~~

~~For both the expected and DBA large leakage cases, it was assumed that leakage water was pumped away to the floor drain receiver tank. Iodine in the leakage water was assumed to be exposed to auxiliary building ventilation air flow for a short period of time (0.05-0.10 hours), and thus, liquid-gas partitioning was assumed for elemental iodine isotopes.~~

~~The large leakage case elemental iodine PFs were calculated using the previously presented PF expression. Because the circulation water will be above 212°F (Chapter 6), a flashing process must be considered. For heat energy conservation on the basis of 1 lb:~~

$$h_{f0} = h_f(1-x) + h_g x \qquad (15.5\text{-}11)$$

~~Rearranging yields~~

$$x = \frac{h_{f0} - h_f}{h_g - h_f} \qquad (15.5\text{-}12)$$

~~where:~~

h_{f0} = ~~initial enthalpy of liquid, Btu/lbm~~
h_f = ~~final enthalpy of liquid, Btu/lbm~~
h_g = ~~final enthalpy of vapor, Btu/lbm~~
x = ~~fraction of initial mass that became vapor~~

~~The end point of the flashing process is 212°F, and thus the final enthalpies are based on this temperature. The mass fraction, x, is the ratio of the final mass of vapor to the total initial mass of water, so the mass ratio at the end of the flashing process becomes:~~

$$\frac{M_{vapor}}{M_{liquid}} = \frac{x}{1-x} \qquad (15.5\text{-}13)$$

~~Figures 15.5-11 and 15.5-12 present the expected and DBA large leakage case elemental iodine PFs as a function of both temperature and pH. For small PFs, the DF (see Equation 15.5-7) is approximately equal to the reciprocal of the PF. Figures 15.5-11 and 15.5-12 illustrate that auxiliary building iodine PFs and resulting DFs are relatively insensitive to water temperature, but much more sensitive to pH. Table 15.5-24 lists the assumed temperatures and pHs along with the resulting elemental iodine PFs and auxiliary building decontamination factors for both the expected and DBA large leakage cases.~~

~~For both the expected and DBA small leakage cases, it was assumed that leakage water was not pumped away. Elemental iodine in the leakage water was assumed to be exposed to auxiliary building ventilation air flow for a long period of time (100-150 hours), and thus, liquid-gas partitioning for elemental iodine isotopes was not assumed. For the small leakage case all elemental iodine activity released to the auxiliary building was assumed to be released to the auxiliary building atmosphere, i.e., a DF of 1.~~

~~Liquid-gas partitioning for organic iodine isotopes was not assumed for any of the four leakage cases. All organic iodine activity released to the auxiliary building was assumed to be released to the auxiliary building atmosphere, i.e., a decontamination factor of 1.~~

~~For all four loop leakage cases, no credit was taken for auxiliary building radiological decay or fission product plateout.~~

~~For the expected and DBA large cases, credit for auxiliary building charcoal filters was taken, and for the expected and DBA small leakage cases, no credit for auxiliary building charcoal filters was taken (as previously discussed with reference to ANS Standard N18.2 single failure criteria). Table 15.5-24 lists the assumed iodine filter efficiencies for each loop leakage case.~~

~~From the calculated DBA case offsite exposures from post-LOCA recirculation loop leakage in the auxiliary building listed in Table 15.5-63, it can be concluded that any exposures that occur via this combination of unlikely events would be well below the guideline levels in 10 CFR Part 100. In addition, even if no consideration is given to the effectiveness of the auxiliary building charcoal filters for the DBA leakage cases, the calculated exposures would still be below guideline levels specified in 10 CFR Part 100.~~

~~15.5.17.2.8.1 Maximum Allowable Leakage From Post-LOCA Recirculation Loop~~

~~Calculations have been performed to determine the maximum allowable leakage from recirculation loop components that could occur during post-LOCA recirculation operations before offsite and control room operator design basis radiation doses would exceed regulatory limits. A computer code (LOCADOSE) was used to determine design basis EAB and low population zone outer boundary (LPZ) offsite radiation doses and control room operator airborne radiation dose from post-LOCA containment leakage, RHR pump seal leakage, and pre-existing leakage from recirculation loop components outside containment. The calculations determined the amount of pre-existing recirculation leakage which could exist before offsite exposures would exceed 10 CFR 100.11 limits or control room operator exposures would exceed GDC 19, 1971 limits, if a LOCA were to simultaneously occur.~~

~~Table 15.5-63 shows the results of the calculations based on the above assumptions which determined that the maximum allowable leakage (in addition to the RHR pump seal leakage) from the recirculation loop at post-LOCA conditions of pressure and~~

~~temperature was 1.85 gpm where the airborne activity is filtered by charcoal filters or 0.186 gpm where the airborne activity is unfiltered. The limitation is the GDC 19, 1971 allowable dose for the control room.~~
In summary, 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 15.5.17.2.2.3 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.~~Credit for filtration of the release of a RHR System pump seal failure by the auxiliary building ventilation system is taken in determining the dose consequences to the public at the EAB and LPZ, control room and TSC.~~

The efficiency of the auxiliary building charcoal filters is determined using methodology similar to that documented in Section 15.5.9 for the CRVS Mode 4 ventilation filters. The allowable methyl iodide penetration / filter bypass for the auxiliary building charcoal filter is controlled by DCPP Technical Specification 5.5.11; and are 5% and <1%, respectively. 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.

15.5.17.2.2.5 Refueling Water Storage Tank Back Leakage

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.

As discussed in NRC Information Notice 91-56, September 1991 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. The acceptance criteria in the DCPP administrative test procedures ensure that the total as-tested back leakage into the RWST from the containment recirculation sump is less than or equal to 1 gpm.

Dose consequences of RWST back-leakage assumes that leakage starts at the switchover to recirculation 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. 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 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 Unit 1 and Unit 2 RWSTs due to RWST back-leakage following switchover to the sump water recirculation mode of operation is summarized in Table 15.5-23C. As discussed earlier, the release fractions / rates presented in Table 15.5-23C 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 15.5-23C 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 15.5-23C 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.

15.5.17.2.2.6 Miscellaneous Equipment Drain Tank (MEDT) Leakage

The DCPP Unit 1 and Unit 2 MEDT is a covered rectangular stainless steel lined concrete tank located in the auxiliary building below El 60 foot. 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 (refer to Section 9.4.2).

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 / RWST leakage from the non-accident unit) which are hard-piped to the MEDT. The acceptance criteria in the DCPP administrative test procedures 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 Guide 1.183, 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. 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 is similar to that used to address RWST back-leakage. Adaptation of the methodology to address overflows/room ventilation releases is straightforward with the room ventilation rate being treated as the tank exhaust rate.

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, which ultimately overflows into the Unit 1/Unit 2 pipe tunnels. The auxiliary building sump is also a covered stainless steel lined concrete tank with a vent that is hard-piped to the auxiliary building ventilation system (ABVS) with a PG&E Design Class II auto transfer capability. The auxiliary building sump 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) Prior to MEDT overflow - 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) After MEDT overflow – The equilibrium iodine distribution balance is conservatively assumed to be between the iodine concentrations in the MEDT overflow liquid and the EDRT room (or Unit 1/Unit 2 pipe tunnels) ventilation flow (rather than the average concentration in the EDRT room (or Unit 1/Unit 2 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 Unit 1/Unit 2 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 Unit 1/Unit 2 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 exhausting to its unit specific plant vent. Thus, it is conservatively assumed that the

non-LOCA unit's ABVS is also operating, and together with the accident units' exhaust fans, are providing the motive force to exhaust the airborne releases to the respective unit 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 15.5-23D. As discussed earlier, the release fractions / rates presented in Table 15.5-23D 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 to 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.

15.5.17.2.3 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 in Section 15.5.17.2.3 was evaluated. The 0-2 hr EAB Atmospheric Dispersion Factor from Table 2.3-145 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.

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 Table 2.3-145.

~~**15.5.17.2.9 Offsite Exposures from Controlled Post-accident Containment**~~

~~Venting~~

~~Because of the potential release of significant amounts of hydrogen to the containment atmosphere following a LBLOCA, it is necessary to provide means of monitoring and controlling the post-accident concentration of hydrogen in the containment atmosphere. Redundant thermal hydrogen recombiners are the primary means of post-accident hydrogen control. As a backup, controlled containment venting (via the containment hydrogen purge system) with offshore flow, wind directions from northwest through east-southeast measured clockwise, provides hydrogen control with a high probability of no inland exposures. As shown in Table 15.5-26, offshore wind directions occur over 50 percent of the time regardless of the season and, as shown in Table 15.5-27, have a high degree of persistence. The large time period (312 hours for DBA case) between the proposed hydrogen venting level (3.5 v/o) and the hydrogen flammability level (4.0 v/o) is much greater than the longest recorded period (37 consecutive hours) of onshore winds in any 22.5° sector. These data ensure a very high probability that venting can be carried out during the occurrence of offshore winds.~~

~~Even though there is a high probability that containment venting can be carried out when the wind is blowing offshore, if necessary at all, an evaluation is presented in the following paragraphs to determine potential exposures if venting were carried out during onshore winds.~~

~~Section 6.2.5 contains the analysis of post-accident hydrogen production and accumulation in the containment atmosphere and its control. Containment venting is also described in Section 6.2.5. The purge stream is withdrawn from the containment through one of two penetration lines. The stream is routed through a flow-measuring device, charcoal filters, exhaust fans, the radiation monitors, and finally to the plant vent.~~

~~Post-accident containment venting activity releases are calculated with the following equation:~~

$$ACT(I) = \frac{[1.0 - 0.01FILEFF(I)]60}{VOLUME} \int_{T(1)}^{T(2)} VENRAT x AC(I) e^{-\lambda(I)t} dt$$

~~where:~~

~~ACT(I) =~~
~~AC(I) = activity of isotope I released to the containment atmosphere, Ci~~
~~VOLUME =~~
~~VENRAT =~~
~~$\lambda(I)$ = removal constant for isotope I, hr^{-1}~~
~~FILEFF(I) =~~
~~T(1) = time after LOCA that containment venting begins, hr~~
~~T(2) = time after LOCA that containment venting ends, hr~~
~~t = time, hr~~

~~60 = minutes per hr~~

~~The above equation considers radiological decay during the time period prior to containment venting and the time period during containment venting. It also assumes that the LBLOCA activity released to the containment atmosphere is homogeneously dispersed throughout the containment atmospheric volume. Exposures from activity released to the atmosphere were calculated using the EMERALD computer code. EMERALD assumes there is no radiological decay during the atmospheric dispersion. Containment venting exposures were calculated for both the expected case and the DBA case. Assumptions and numerical values used to calculate venting exposures are itemized in Table 15.5-28. Onshore controlled containment venting thyroid and whole body exposures are listed in Table 15.5-29.~~

~~Post-accident containment venting schedules are evaluated in Section 6.2.5. Assuming the venting system will operate an average 2 hours per day, the system flowrates during short venting periods are 120 cfm (expected) and 300 cfm (DBA). Equivalent continuous venting rates, 10 cfm and 25 cfm, were used to calculate venting activity releases.~~

~~In the event containment venting should be required during periods with onshore flow, the venting would be limited to those periods when Pasquill Stability Category D exists. Therefore, ground-level centerline atmospheric dispersion factors for Pasquill Stability Category D and an elevated release height of 70 meters were evaluated using a conventional Gaussian plume model and are listed in Table 15.5-30. The meteorological input parameters utilized were determined from onsite measurements, given in References 18, 19, and 20. Because an individual is assumed to be located on the plume centerline for the entire venting duration, exposures are centerline exposures and represent worst case conditions. The probability of an individual being located on the plume centerline for a 2-hour period is very small, and thus centerline exposures listed in Table 15.5-29 are very conservative.~~

~~During the time period prior to venting, activity released to the containment atmosphere is significantly reduced by both radiological decay and functioning of the safety features systems. The main contributors of radioactivity several hundred hours after the accident are the noble gases: Kr-85, Xe-133, and, to some extent, Xe-131m. Because Kr-85 has a half-life of 10.6 years, the exposures resulting from containment venting would not be significantly reduced if the venting could be further delayed for many months.~~

~~It can be concluded from the results presented in Table 15.5-29, along with the consideration of the very high probability of opportunities for offshore venting and the other favorable factors associated with the DCPP design and site, that, as a backup to the internal hydrogen recombiner system, controlled venting using the containment hydrogen purge system is an acceptable contingency method of post-accident hydrogen control for this plant. In addition, it can be concluded that the expected exposures due to venting, even using the assumptions in Safety Guide 7, will not exceed the annual~~