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

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

FHA in the FHB

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

During fuel movement in the FHB:

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

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

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

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

It has been determined that for the FHA in the FHB, the actual release rate lambda based on the FHBVS exhaust (i.e., 8.7 hr^{-1}) is larger than the release rate applicable to "a 2-hr release" per Regulatory Guide 1.183 (i.e., 3.45 hr^{-1}). Thus the larger exhaust rate lambda associated with FHBVS operation plus the exhaust rate lambda for the 500 cfm outleakage is utilized in the analysis.

FHA in the Containment

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

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

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

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

15.5.22.2.3 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. Since the FHA is based on a 2-hour release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

The bounding EAB and LPZ dose following a FHA at either location and at either unit is presented in Table 15.5-47.

15.5.22.2.4 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. Provided below are the critical FHA-specific assumptions associated with control room response and activity transport.

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

Credit is taken for PG&E Design Class I area radiation monitors located at the control roomcontrol room normal intakes (1-RE-25/26, 2-RE-25/26) to initiate CRVS Mode 4 (filtered / pressurized accident ventilation) upon detection of high radiation levels at the control room normal intakes as a result of an FHA.

An analytical safety limit of 1 mR/hr for the gamma radiation environment at the control room normal operation air intakes has been used in the FHA analyses to initiate CRVS Mode 4. Note that the actual monitor trip setpoint is lower to include the instrument loop uncertainty.

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

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

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

It is determined that a DBA FHA (i.e., occurs at 72 hrs post shutdown) will result in a radiation environment at the control room normal operation intakes that greatly exceed the analytical limit of 1 mR/hr for initiating CRVS Mode 4. This will result in an almost instantaneous generation of a radiation monitor signal to initiate CRVS Mode 4 (radiation monitor response time is estimated to be < 1 sec). For purposes of conservatism, and since the delay in isolation of the normal intake has a significant impact on the estimated dose consequences, the analysis conservatively assumes a monitor response time to the HASP of 10 secs.

As discussed in Section 15.5.1.2, when crediting CRVS Mode 4, the FHA dose consequence analyses is not required to address the potential effects of a LOOP. Thus delays associated with diesel generator sequencing are not addressed.

Therefore, the time delay between the arrival of radioactivity released due to a DBA FHA at both the control room normal Intakes (assumed to be instantaneous) and CRVS Mode 4 operation is estimated to be the sum total of the monitor response time (10 secs), the signal processing time (2 Secs) and the damper closure time (10 secs) for a total delay of 22 seconds.

<u>Delayed FHA:</u>

It is recognized that the response time for radiation monitors are dependent on the magnitude of the radiation level / energy spectrum of the airborne cloud at the location of the detectors, which in turn are dependent on the fuel assembly decay time. Thus an additional case is considered for each of the two FHA scenarios described above

(i.e., a FHA in the FHB and a FHA in Containment) when determining the dose to the control room operator; i.e., a case that reflects a delayed FHA at Fuel Offload or a FHA during Reload, occurring at a time when the fuel has decayed to such an extent that the radiation environment at the control room normal intake radiation monitors is just below the setpoint; thus the control room remains in normal operation mode and CRVS Mode 4 is not initiated.

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

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to an FHA at either location, and at either unit, are provided in Table 15.5-47B. The χ/Q values presented in Table 15.5-47B take into consideration the various release points-receptors applicable to the FHA to identify the bounding χ/Q values applicable to a FHA at either unit and at either location, and reflect the allowable adjustments / reductions in the values as discussed in 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 FHA at either location and at either unit is presented in Table 15.5-47.

15.5.22.~~1.~~3 Conclusions

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

~~(1) An individual located at any point on the boundary of the exclusion area for any two hour period following the onset of the postulated fission product release shall not receive a total radiation dose in excess of 0.063 Sv (6.3 rem) total effective dose equivalent (TEDE) as shown in Table 15.5-47.~~

~~(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 0.063 Sv (6.3 rem) total effective dose equivalent (TEDE) as shown in Table 15.5-47.~~

~~(3) The dose to the control room operator under accident conditions shall not be in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident as shown in Table 15.5-47.~~

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

(1) The radiation dose to 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 is within 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5-47.

(2) The radiation dose to 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 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5-47.

(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 15.5-47.

~~15.5.22.2 Fuel Handling Accident Inside Containment~~

~~15.5.22.2.1 Acceptance Criteria~~

~~(1) The radiological consequences of a fuel handling accident inside containment 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 6 rem or a total radiation dose in excess of 75 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 6 rem, or a total radiation dose in excess of 75 rem to the thyroid from iodine exposure.~~

~~(2) 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.~~

~~15.5.22.2.2 Identification of Causes and Accident Description~~

~~The offsite radiological consequences of a postulated fuel handling accident inside the containment are mitigated by containment closure. The following evaluation shows that in all cases the calculated exposures would be well below limits specified in 10 CFR 100.11.~~

~~During fuel handling operations, containment closure is not required. Generally, the containment ventilation purge system is operational and exhausts air from the containment through two 48-inch containment isolation valves. These two valves are connected in series. This flow of air from the containment is discharged to the environment via the plant vent.~~

~~This exhaust stream is monitored for activity by monitors in the plant vent. In the event of a postulated fuel handling accident, the plant vent monitors will alarm and result in the automatic closure of containment ventilation isolation valves. This activity release may result in offsite radiological exposures.~~

~~Containment penetrations are allowed to be open during fuel handling operations. The most prominent of these penetrations are the equipment hatch and the personnel airlock. Closure of these penetrations is achieved by manual means as discussed in Section 15.4.5. The closure of these penetrations is not credited in the design basis fuel handling accident inside containment.~~

~~The FHA analysis assumes that the control room ventilation system of each unit remains in the normal mode of operation following the FHA. Thus, the design basis FHA does not credit charcoal filtration of the control room atmosphere intake flow or recirculation flow.~~

~~The evaluation of potential offsite exposures was performed for a design basis case, assuming plant parameters as limited by Technical Specifications. The assumptions of Safety Guide 25, March 1972, were used as guidance with the exceptions detailed below.~~

~~15.5.22.2.2.1 Activity Released to Containment Atmosphere~~

~~The assumptions made in determining the quantity of activity available for release from the containment refueling pool following the postulated accident are identical to those discussed in Section 15.5.22.1. For the DBA case, these assumptions are consistent with those in Safety Guides 25, March 1972, and 1.183, July 2000.~~

~~Consistent with the guidance of Safety Guide 25, March 1972, it was assumed that all the gap activity in the damaged rods is released and consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85, and 10 percent of the total radioactive iodine in the rods at the time of the accident.~~

~~An effective DF of 200 for the iodines was assumed for the water in the refueling cavity. This DF is consistent with the current guidance provided in Regulatory Guide 1.183, July 2000.~~

~~The dose conversion factors used are from ICRP Publication 30 (Reference 45). The use of these dose conversion factors is consistent with the current guidance provided in Regulatory Guide 1.183, July 2000.~~

~~15.5.22.2.2.2 Containment Closure~~

~~Following the postulated accident, airborne activity evolves from the surface of the pool where it mixes with air above the pool. Airborne activity is then assumed to be discharged to the environment via the open penetrations. The duration of the release was assumed to be within two seconds.~~

~~In addition to radiation monitor indications, a fuel handling accident would immediately be known to refueling personnel at the scene of the accident. These personnel would initiate containment closure actions and are required by an Equipment Control Guideline to be in constant communication with control room personnel. The plant intercom system is described in Section 9.5.2.~~

~~15.5.22.2.2.3 Activity Released to Environment~~

~~The containment refueling pool is approximately rectangular in shape with approximate dimensions of 25 by 70 feet. The pool has a surface area of about 1750 square feet.~~

~~It was assumed that activity evolved from the pool was instantaneously mixed and retained within the approximately 33,600 cubic foot rectangular parallelepiped formed by the 25- by 70-foot pool and the 40-foot-high steam generators. Where the steam generators do not surround the pool, the radioactivity would actually be dispersed into a larger volume of air which would have the effect of reducing the dose. However, for conservatism, it was assumed that all the radioactivity remained within this 33,600-cubic-foot volume and was then transported to the environment within a two second time period through the open equipment hatch.~~

~~15.5.22.2.2.4 Offsite Exposures~~

~~The integrated release of activity to the environment and the resulting offsite radiological exposures were calculated for the postulated fuel handling accident inside containment using the LOCADOSE computer program.~~

~~Table 15.5-48 itemizes the DBA assumptions and numerical values used to calculate fuel handling accident radiological exposures. The calculated releases of activity to the atmosphere are listed in Table 15.5-49. The DBA exposures resulting from the postulated fuel handling accident inside containment are presented in Table 15.5-50. These exposures are well within the 10 CFR 100.11 limits.~~

~~15.5.22.2.2.5 Action Following Containment Isolation~~

~~Following manual containment closure after the fuel handling accident, activity can be removed from the containment atmosphere by the redundant PG&E Design Class II Iodine Removal System (two trains at 12,000 cfm per train), which consists of~~

~~HEPA/charcoal filters. This system is described in Section 9.4.5. There are no Technical Specification requirements for this filtration system.~~

~~The containment can also be purged to the atmosphere at a controlled rate of up to 300 cfm per train through the HEPA/charcoal filters of the hydrogen purge system. This system is described in Section 6.2.5.~~

~~15.5.22.2.3 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 below the dose limits of 10 CFR 100.11 as shown in Table 15.5-50.~~

~~(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 below the dose limits of 10 CFR 100.11 as shown in Table 15.5-50.~~

~~(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 as shown in Table 15.5-50.~~

~~15.5.22.3 Conclusion, Fuel Handling Accidents~~

~~In the preceding sections the potential offsite exposures from major fuel handling accidents have been calculated. The analyses have been carried out using the models and assumptions specified in pertinent regulatory guides. In all analyses the resulting potential exposures to individual members of the public and the general population have been found to be lower than the applicable guidelines and limits specified in 10 CFR 100.11. (FHA in Containment) and 10 CFR 50.67 (FHA in FHB).~~

~~On this basis, it can be concluded that the occurrence of a major fuel handling accident in a DCPP unit would not constitute an undue risk to the health and safety of the public.~~

~~Additionally, it can be concluded that the ESF provided for the mitigation of the consequences of a major fuel handling accident are adequate.~~

15.5.23 RADIOLOGICAL CONSEQUENCES OF A CONTROL ROD EJECTION ACCIDENT

15.5.23.1 Acceptance Criteria

The radiological consequences of a CREA 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:

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 0.063 Sv (6.3 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 0.063 Sv (6.3 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.

~~(1) The radiological consequences of a rod ejection accident 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.~~

(2) ~~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) for the duration of the accident.~~

15.5.23.2 Identification of Causes and Accident Description

15.5.23.2.1 Activity Release Pathway

As discussed in Section 15.4.6, this event consists of an uncontrolled withdrawal of a control rod from the reactor core. The CREA results in reactivity insertion that leads to a core power level increase, and under adverse combinations of circumstances, fuel failure, and a subsequent reactor trip. In this case, some of the activity in the fuel rod gaps would be released to the coolant and in turn to the inside of the containment building. As a result of pressurization of the containment, some of this activity could leak to the environment.

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

Regulatory requirements provided for the CREA in pertinent sections of Regulatory Guide 1.183 including Appendix H is used to develop the dose consequence model. Table 15.5-52A lists the key assumptions / parameters utilized to develop the radiological consequences following a CREA.

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

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

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

~~under adverse combinations of circumstances, some fuel cladding failures could occur following a rod ejection accident. In this case, some of the activity in the fuel rod gaps would be released to the coolant and in turn to the inside of the containment building. As a result of pressurization of the containment, some of this activity could leak to the environment.~~
Computer code RADTRAD 3.03, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a CREA.

~~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.~~

~~Following a postulated rod ejection accident, activity released from the fuel pellet-cladding gap due to failure of 10 percent of the fuel rods is assumed to be instantaneously released to the primary coolant. Releases to the primary coolant are assumed to be immediately and uniformly mixed throughout the coolant.~~

~~The activity released to the containment from the primary coolant through the ruptured control rod mechanism pressure housing is assumed to be mixed instantaneously throughout the containment and is available for leakage to the atmosphere.~~

~~It has been assumed for both the design basis and expected cases that 10 percent of the elemental iodine leaked to the coolant is released to the containment atmosphere as a result of flashing of some of the primary coolant water. Of the amounts of noble gases released to the primary coolant, 100 percent is assumed to be released to the containment atmosphere at the time of the accident. It is assumed that the amount 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.~~

~~Following the release to the containment, the fission products are assumed to leak from the containment at the same rates assumed for the LBLOCA, discussed in Section 15.5.17. In addition, the spray system is assumed to be in operation and acts to remove the iodines from the containment atmosphere at the same rates assumed for the LBLOCA.~~

~~The assumptions used for meteorology, breathing rates, population density, and other common factors were also described in earlier sections. Both the primary and secondary coolant activities prior to the accident are given in Section 15.5.3. The gap activities are listed in Table 11.1-7.~~

~~All of the data and assumptions listed above were used with the EMERALD computer program to calculate the activity releases and potential doses following the accident. The calculated activity releases are listed in Table 15.5-51, and the potential doses are given in Table 15.5-52. Thyroid doses that would result from secondary steam releases can be determined from Figures 15.5-2 and 15.5-3 for the DBA conditions and Figures 15.5-4 and 15.5-5 for the expected conditions.~~

~~If atmospheric steam releases occur following this accident, there will be some additional exposures via this pathway. The detailed assumptions used in estimating mode of exposure are described in Section 15.5-21. The results are given parametrically in Figures 15.5-14 and 15.5-15. It should be noted that these figures are based on the assumptions of a full plant cooldown with no condenser capacity available, a condition that would not be expected to occur following a rod ejection accident.~~

~~From these analyses, it can be concluded that offsite exposures from this accident will be well below the guideline levels specified in 10 CFR 100.11, and that the occurrence of such accidents would not result in undue risk to the public. A detailed evaluation of potential exposures to control room personnel is made in Section 15.5.17 for conditions following a LOCA.~~

15.5.23.2.2 Activity Release Transport Model

The CREA dose consequence analysis evaluates the following two scenarios.

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

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

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

Scenario 1: Transport From Containment

The failed fuel activity released due to a CREA into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the technical specification leak rate of 0.10% per day for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen or noble gas concentrations airborne in the containment. Per Regulatory Guide 1.183, the chemical composition of the iodine in the gap fuel is 95% particulate (CsI), 4.85% elemental and 0.15% organic. However, since no credit is taken for the actuation of sprays or pH control, the iodine released via containment leakage pathway is assumed to have the same composition as iodine activity released to the environment from the secondary coolant; i.e.; 97% elemental and 3% organic. Environmental releases due to containment leakage can occur unfiltered as a diffuse source from the containment wall, and as a point source via the containment penetration areas or the Plant Vent. The dose consequences are estimated based on the worst case atmospheric dispersion factors, i.e., an assumed environmental release via the containment penetration areas.

Scenario 2: Transport from Secondary System

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

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

The effect of SG tube uncovery in intact SGs (for SGTR and non-SGTR events), has been evaluated for potential impact on dose consequences as part of a WOG Program and demonstrated to be insignificant; therefore, the gap iodines have a partition coefficient of 100 in the SG. The gap noble gases are released freely to the

environment without retention in the SG.

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

15.5.23.2.3 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 Scenario 1 (release via Containment leakage), the worst case 2-hour period occurs during the first 2 hours). For Scenario 2 (release via secondary side), the worst two hour period can occur either during the 0-2 hr period when the noble gas release rate is the highest, or during the t=8.73 hr to 10.73 hr period when the iodine and particulate level in the SG liquid peaks (SG releases are terminated at T=10.73 hrs). Regardless of the starting point of the worst 2 hr window, the 0-2 hr EAB χ/Q is utilized.

The bounding EAB and LPZ dose following a CREA at either unit for both scenarios are presented in Table 15.5-52.

15.5.23.2.4 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. Provided below are the critical CREA-specific assumptions associated with control room response and activity transport.

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

The time to generate a signal to switch CRVS operation from Mode 1 to Mode 4 is based on the containment pressure response following a 2 inch small-break LOCA (SBLOCA), and the fact that at DCPP, a Containment High Pressure signal will initiate a SIS which will automatically initiate CRVS Mode 4 pressurization. The containment pressure response analysis for a 2 inch SBLOCA shows that the 3 psig setpoint for Containment High Pressure is reached in 150 seconds after the SBLOCA. As indicated earlier, releases to the containment following a CREA are through a faulted control rod drive mechanism housing. The control rod shaft diameter is 1.840 inches and the RCCA housing penetration opening is 4 inches in diameter. Based on the above and for the purposes of conservatism, the time to generate the Containment High Pressure SIS following a CREA is assumed to be double the value applicable to the 2 inch SBLOCA, or 300 seconds.

Based on the above, following a CREA,

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

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

As noted in Section 2.3.5.2.2, because of the proximity of the MSSV/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 faulted unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's control room normal intake is assumed to be contaminated by a release from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a CREA at either unit are provided in Table 15.5-52B. The χ/Q values presented in Table 15.5-52B take into consideration the various release points-receptors applicable to the CREA to identify the bounding χ/Q values applicable to a CREA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 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 CREA at either unit is presented in Table 15.5-52.

15.5.23.3 Conclusions

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

(1) The radiation dose to 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 is within 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5-52.

(2) The radiation dose to 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 0.063 Sv (6.3 rem) TEDE as shown in Table 15.5-52.

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 15.5-52 ~~comparing the activity releases following a rod ejection accident, given in Table 15.5-51, with the activity releases calculated for a LOCA, given in Tables 15.5-13 and 15.5-14, it can be concluded that any control room exposures following a rod ejection accident will be well below the GDC 19, 1971, criterion level.~~

~~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 immediately following the onset of the postulated fission product release are well below the dose limits of 10 CFR 100.11 as shown in Table 15.5-52.~~

~~(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 below the dose limits of 10 CFR 100.11 as shown in Table 15.5-52.~~

~~(3) Since the activity releases from the rod ejection accident given in Table 15.5-51 are less than those from a LBLOCA (see Table 15.5-13 and 15.5-14), any control room dose which might occur would be well below the established criteria of GDC 19, 1971, and discussed in Section 15.5.17.~~

15.5.24 RADIOLOGICAL CONSEQUENCES OF A RUPTURE OF A WASTE GAS DECAY TANK

15.5.24.1 Acceptance Criteria

The radiological consequences of a rupture of a waste gas decay tank shall not exceed the dose limits of 10 CFR 100.11 as outlined below:

(1) 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.

15.5.27 REFERENCES

1. Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plant, N18.2, American Nuclear Society, 1972.

2. Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, US Atomic Energy Commission (AEC), Rev. 1, October 1972.

3. Regulatory Guide 4.2, Preparation of Environmental Reports for Nuclear Power Plants, Directorate of Regulatory Standards, AEC, March 1973.

4. W. K. Burnot, et al, EMERALD (REVISION I) - A Program for the Calculation of Activity Releases and Potential Doses, Pacific Gas and Electric Company, March 1974.

5. S. G. Gillespie and W. K. Brunot, EMERALD NORMAL - A Program for the Calculations of Activity Releases and Doses from Normal Operation of a Pressurized Water Plant, Program Description and User's Manual, Pacific Gas and Electric Company, March 1973.

6. Regulatory Guide Number 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors, AEC, Rev. 1, June 1973.

7. D. H. Slade, ed., Meteorology and Atomic Energy 1968, AEC Report Number TID-24190, July 1968.

8. International Commission on Radiological Protection (ICRP) Publication 2, Report of Committee II, Permissible Dose for Internal Radiation, 1959.

9. Deleted~~R. L. Engel, et al, ISOSHLD - A Computer Code for General Purpose Isotope Shielding Analysis, BNWL-236, UC-34, Physics, Pacific Northwest Laboratory, Richland, WA, June 1966~~.

10. R. K. Hilliard, et al, "Removal of Iodine and Particles by Sprays in the Containment Systems Experiment," Nuclear Technology, April 1971.

11. G. L. Simmons, et al, ISOSHLD-II: Code Revision to Include Calculation of Dose Rate from Shielded Bremsstrahlung Sources, BNWL-236-SUP1, UC-34, Physics, Pacific Northwest Laboratory, Richland, WA, March 1967.

12. L. F. Parsly, Calculation of Iodine - Water Partition Coefficients, ORNL-TM-2412, Part IV, January 1970.

13. Westinghouse, Radiological Consequences of a Fuel Handling Accident, December 1971.

14. Deleted. ~~F. J. Brutschy, et al, Behavior of Iodine in Reactor Water During Plant Shutdown and Startup, General Electric Co. Atomic Power Equipment Department Report, NEDO-10585, August 1972~~.

15. Deleted in Revision 16.

16. Proposed Addendum to ANS Standard N18.2, Single Failure Criteria for Fluid Systems, American Nuclear Society, May 1974.

17. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19," 13th AEC Air Cleaning Conference, August 1974.

18. M. L. Mooney and H. E. Cramer, Meteorological Study of the Diablo Canyon Nuclear Power Plant Site, Meteorological Office, Gas Control Department, PG&E, 1970 (see also Appendix 2.3A in Reference 27 of Section 2.3 in this FSAR Update).

19. M. L. Mooney, First Supplement, Meteorological Study of the Diablo Canyon Nuclear Power Plant Site, Meteorological Office, Gas Control Department, PG&E, 1971 (see also Appendix 2.3C in Reference 27 of Section 2.3 in this FSAR Update).

20. M. L. Mooney, Second Supplement, Meteorological Study of the Diablo Canyon Nuclear Power Plant Site, Meteorological Office, Gas Control Department, PG&E, 1972 (see also Appendix 2.3D in Reference 27 of Section 2.3 in this FSAR Update).

21. International Commission on Radiological Protection Publication 30, Limits for Intakes of Radionuclides by Workers, 1979.

22. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR 82, as amended.

23. Safety Guide 25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, USNRC, March 1972.

24. Safety Guide 24, Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure, USNRC, March 1972.

25. Deleted.~~Start, G. E., J. H. Cate, C. R. Dickson, N. R. Ricks, G. H. Ackerman, and J. F. Sagendorf, "Rancho Seco Building Wake Effects on Atmospheric Diffusion, NOAA Technical Memorandum, ERL ARL-69, 1977~~.

26. Deleted~~Walker, D. H., R. N. Nassano, M. A. Capo, "Control Room Ventilation Intake Selection for the Floating Nuclear Power Plant," 14th ERDA Air Cleaning Conference, 1976~~.

27. Deleted~~Hatcher, R. N., R. N. Meroney, J. A. Peterka, K. Kothari, "Dispersion in the Wake of a Model Industrial Complex," NUREG-0373, 1978~~.

28. Deleted~~Meroney, R. N., and B. T. Yang, Wind Tunnel Study on Gaseous Mixing due to Various Stack Heights and Injection Rates Above an Isolated Structure, FDDL Report CER 71-72 RNM-BTY16, Colorado State University, 1971~~.

29. Deleted~~R. P. Hosker, Jr., "Dispersion in the Vicinity of Buildings," Preprints of Second Joint Conference on Applications in Air Pollution Meteorology and Second Conference on Industrial Meteorology, New Orleans, LA, March 24-28, 1980, pp. 92-107, American Meteorological Society, Boston, Mass. Also in "Flow and Diffusion Near Obstacles," Chapter 7 of Atmospheric Sciences and Power Production, D. Randerson, ed., USDOE~~.

30. Deleted~~D. J. Wilson, Contamination of Air Intakes from Roof Exhaust Vents, ASHRAE Trans. 82, Part 1, pp. 1024-1038, 1976~~.

31. Deleted~~R. J. B. Bouwmeester, K. M. Kothari, R. N. Meroney, An Algorithm to Estimate Field Concentrations Under Nonsteady Meteorological Conditions from Wind Tunnel Experiments, NUREG/CR-1474, USNRC, September 1980~~.

32. R. Bhatia, J. Dodds, and J. Schulz, Building Wake χ/Qs for Post-LOCA Control Room Habitability, Bechtel Power Corporation, San Francisco, CA.

33. Report on the Methodology for the Resolution on the Steam Generator Tube Uncovery Issue, WCAP-13247, March 1992.

34. Deleted in Revision 18.

35. Deleted.~~Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, AEC, Revision 2, June 1974.~~

36. T. R. England and R. E. Schenter, ENDF-223, ENDF/B-IV Fission Product Files: Summary of Major Nuclide Data, October 1975.

37. Standard Review Plan, Section 15.6.3, Radiological Consequences of Steam Generator Tube Failure (PWR), NUREG-0800, USNRC, July 1981.

38. Deleted in Revision 18.

39. Westinghouse Letter PGE-91-533, Safety Evaluation for Containment Spray Flow Rate Reduction, February 7, 1991.

40. Westinghouse Letter PGE-93-652 dated October 5, 1993, transmitting NSAL-93-016, Revision 1.

41. K. F. Eckerman et. al., Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

42. K. F. Eckerman and J. C. Ryman, External Exposure to Radionuclides in Air, Water, and Soil, Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

43. Deleted in Revision 12.

44. ~~Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," 05/2003.~~Deleted

~~44.~~45. Deleted~~International Commission on Radiological Protection (ICRP) Publication 30, Limits For Intakes of Radionuclides by Workers, 07/1978~~.

~~45.~~46. Diablo Canyon Units 1 and 2 Replacement Steam Generator Program – NSSS Licensing Report, WCAP-16638 (Proprietary), September 2007.

~~46.~~47. LOCADOSE-NE319, A Computer Code System for Multi-Region Radioactive Transport and Dose Calculation, Release 6, Bechtel Corporation.

~~47.~~48. PG&E Calculation N-166, Small Break LOCA Doses, Revision 0, October 31, 1994.

~~48.~~49. Diablo Canyon Units 1 and 2 T_{avg} and T_{feed} Ranges Program NSSS Engineering Report, WCAP-16985 (Proprietary), April 2009.

~~49.~~50. Deleted~~A. G. Croff, ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621, Oak Ridge National Laboratory, July 1980~~.

~~50.~~51. NRC Letter, License Amendment No. 155/155, "Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment RE: Revision of Technical

Specifications Section 3.9.4, Containment Penetrations (TAC Nos. MB3595 and MB3596), Accession No. ML021010606, October 21, 2002.

52. ~~"RADTRAD: A Model for Radionuclide Transport, Removal and Dose", NUREG/CR-6604, SAND98-0272, April, 1998.~~Deleted

53. TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites”, 1962.

54. Not Used.

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

56. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors”, Revision 1.

57. NUREG 0737, Clarification of TMI Action Plan Requirements, November 1980.

58. NUREG 0737, Supplement 1, Clarification of TMI Action Plan Requirements, January 1983.

59. NUREG-0800, Standard Review Plan 15.0.1, “Radiological Consequence Analyses using Alternative Source Terms,” Revision 0.

60. Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”, Revision 1.

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

62. NUREG/CR 5009, Assessment of the Use of Extended Burn Fuel in LWRs, January 1988.

63. RADTRAD 3.03 (GUI Mode Version), A Simplified Model for RADionuclide Transport and Removal And Dose Estimation, NUREG/CR-6604, Users' Guide - Supplement 2, October 2002.

64. SCALE 4.3, “Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations And Personal Computers,” Control Module SAS2.

65. ACTIVITY2, “Fission Products in a Nuclear Reactor” - CB&I Proprietary Computer Code NU-014, V01, L03.

66. IONEXCHANGER, - CB&I Proprietary Computer Code NU-009, Ver. 01, Lev. 03.

67. SWNAUA, "Aerosol Behavior in Condensing Atmosphere", CB&I Proprietary Computer Code NU-185, V02, L0.

68. RADTRAD 3.03 "A Simplified Model for RADionuclide Transport and Removal And Dose Estimates.

69. PERC2, "Passive Evolutionary Regulatory Consequence Code" - CB&I Proprietary Computer Code, NU-226, V00, L02.

70. SW-QADCGGP, "A Combinatorial Geometry Version of QAD-5A" - CB&I Proprietary Computer Code, NU-222, V00, L02.

~~53.~~71. GOTHIC, "Generation of Thermal-Hydraulic Information for Containments".

72. Not Used

73. EN-113, "Atmospheric Dispersion Factors" - CB&I Computer Code EN-113, V06, L08.

74. ARCON96, "Atmospheric Relative Concentrations in Building Wakes.

75. NUREG-0800, Standard Review Plan (SRP) Sections 15.2.1-15.2.5, "Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)", Revision 1

76. NUREG-0800, SRP Section 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries", Revision 1.

77. USNRC Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG 0800, Section 15.6.5, Appendix B, Revision 1, "Radiological Consequences of a Design Basis LOCA: Leakage from engineered Safety Feature Components outside Containment".

78. NRC Generic Letter No. 99-02, Laboratory Testing of Nuclear Grade Activated Charcoal, June 3, 1999.

79. Not Used

80. NUREG 0800, 1988, Standard Review Plan, "Containment Spray as a Fission Product Cleanup System", Section 6.5.2, Revision 4.

81. NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays”, June 1993.

82. NUREG/CR-5732, “Iodine Chemical Forms in LWR Severe Accidents – Final Report,” April 1992.

83. ANSI/ANS 6.1.1-1977, “Neutron and Gamma-ray Flux-to-Dose Rate Factors”

84. ANSI/ANS 6.1.1-1991, "Neutron and Gamma-ray Fluence-to-dose Factors".

85. NRC Information Notice 91-56, September 19, 1991, “Potential Radioactive Leakage to Tank Vented to Atmosphere”.

86. NUREG 0800, Standard Review Plan 15.2.8, Revision 2, “Feedwater System Pipe Break Inside and Outside Containment (PWR)”

~~54.~~87. NRC SER Related to Amendment No. 8 and 6 to Facility Operating License No. DPR-80 and. DPR-82, PG&E, Diablo Canyon Power Station, Units 1 and 2, dated May 30, 1986.

~~TABLE 15.5-1~~

~~REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES DURING STEADY STATE OPERATION AND PLANT SHUTDOWN OPERATION~~

	~~Operating PWR Plant (HISTORICAL)~~		~~Diablo Canyon - Design Basis Case~~	
~~Isotope~~	~~*Measured Activity Before Shutdown, mCi/gm*~~	~~*Measured Peak Shutdown Activity, mCi/gm*~~	~~Calculated Activity Before Shutdown, mCi/gm~~	~~Expected Peak Shutdown Activity, mCi/gm~~
~~I-131~~	~~*0.83*~~	~~*14.9*~~	~~2.45~~	~~43.9~~
~~Xe-133~~	~~*127.00*~~	~~*65.0*~~[(a)]	~~255.8~~	~~130.9~~[(a)]
~~Cs-134~~	~~*1.29*~~	~~*1.7*~~	~~0.198~~	~~0.26~~
~~Cs-137~~	~~*1.67*~~	~~*2.14*~~	~~0.31~~	~~0.39~~
~~Ce-144~~	~~*0.00068*~~	~~*0.0058*~~	~~0.00034~~	~~0.0029~~
~~Sr-89~~	~~*0.0033*~~	~~*0.40*~~	~~0.0026~~	~~0.32~~
~~Sr-90~~	~~*0.00057*~~	~~*0.013*~~	~~0.00013~~	~~0.003~~
~~Co-58~~	~~—~~	~~*0.95*~~	~~0.026~~	~~1.04~~

~~(a) Activity reduced from steady state level by approximately 1 day of system degasification prior to plant shutdown.~~

~~TABLE 15.5-2 (HISTORICAL)~~

~~*RESULTS OF STUDY OF EFFECTS OF PLUTONIUM ON ACCIDENT DOSES*~~

~~*Type of Accident*~~	~~*Change in 30-day Thyroid Dose, %*~~	~~*Change in 30-day Whole Body Dose, %*~~	~~*Change in 2-hour Thyroid Dose, %*~~	~~*Change in 2-hour Whole Body Dose, %*~~
~~*Release from gas decay tank*~~	~~*0*~~	~~*-4*~~	~~*0*~~	~~*-4*~~
~~*Fuel handling accident*~~	~~*+6*~~	~~*-3*~~	~~*0*~~	~~*0*~~
~~*Loss of reactor primary coolant - large break*~~	~~*+6*~~	~~*-3*~~	~~*+5*~~	~~*-7*~~
~~*Steam generator tube rupture accident*~~	~~*+6*~~	~~*-2*~~	~~*+4*~~	~~*-2*~~
~~*Steam line rupture accident*~~	~~*+5*~~	~~*-2*~~	~~*+5*~~	~~*-2*~~

~~TABLE 15.5-4~~

~~EXPECTED POST ACCIDENT ATMOSPHERIC DILUTION FACTORS (SEC/M^3)~~

~~Period, hrs~~	~~Distance from Release Point, meters[(a)]~~						
~~0-8~~	~~$5.29x10^{-5}$~~	~~$3.40x10^{-5}$~~	~~$1.87x10^{-5}$~~	~~$7.78x10^{-5}$~~	~~$3.59x10^{-6}$~~	~~$2.20x10^{-6}$~~	~~$8.85x10^{-7}$~~
~~8-24~~	~~$2.15x10^{-5}$~~	~~$1.40x10^{-5}$~~	~~$5.00x10^{-6}$~~	~~$1.75x10^{-6}$~~	~~$7.50x10^{-7}$~~	~~$4.75x10^{-7}$~~	~~$1.75x10^{-7}$~~
~~24-96~~	~~$7.70x10^{-6}$~~	~~$3.90x10^{-6}$~~	~~$1.75x10^{-6}$~~	~~$5.70x10^{-7}$~~	~~$2.50x10^{-7}$~~	~~$1.54x10^{-7}$~~	~~$5.50x10^{-8}$~~
~~96-720~~	~~$1.75x10^{-6}$~~	~~$8.20x10^{-7}$~~	~~$3.70x10^{-7}$~~	~~$1.35x10^{-7}$~~	~~$5.20x10^{-8}$~~	~~$3.40x10^{-8}$~~	~~$1.20x10^{-8}$~~

~~(a) Minimum exclusion area boundary radius is 0.5 miles (approximately 800 m). Radius of low population zone is 6.2 miles (approximately 10,000 m).~~

~~TABLE 15.5-5~~

~~ATMOSPHERIC DILUTION FACTORS~~
~~$\chi/Q \times 10^{8}$ sec-m^{-3}~~

~~Onshore Sector Midpoint Directions~~	~~Sector Midpoint Downwind Distance, miles~~					
	~~5~~	~~15~~	~~25~~	~~35~~	~~45~~	~~55~~
~~SSE~~	~~1.61~~	~~0.54~~	~~0.32~~	~~0.23~~	~~0.18~~	~~0.15~~
~~S~~	~~1.44~~	~~0.48~~	~~0.29~~	~~0.21~~	~~0.16~~	~~0.13~~
~~SSW~~	~~0.79~~	~~0.26~~	~~0.16~~	~~0.11~~	~~0.09~~	~~0.07~~
~~SW~~	~~0.54~~	~~0.18~~	~~0.11~~	~~0.08~~	~~0.06~~	~~0.05~~
~~WSW~~	~~0.65~~	~~0.22~~	~~0.13~~	~~0.09~~	~~0.07~~	~~0.06~~
~~W~~	~~1.08~~	~~0.36~~	~~0.22~~	~~0.15~~	~~0.12~~	~~0.10~~
~~WNW~~	~~1.19~~	~~0.40~~	~~0.24~~	~~0.17~~	~~0.13~~	~~0.11~~
~~NW~~	~~5.39~~	~~1.80~~	~~1.08~~	~~0.77~~	~~0.60~~	~~0.49~~
~~NNW~~	~~1.94~~	~~0.65~~	~~0.39~~	~~0.28~~	~~0.22~~	~~0.18~~

~~Atmospheric Dilution Factors~~
~~$\chi/Q \times 10^{6}$ sec-m^{-3}~~

	~~Downwind Distance, meters~~						
~~Direction~~	~~800~~	~~1200~~	~~2000~~	~~4000~~	~~7000~~	~~10,000~~	~~20,000~~
~~SE~~	~~0.75~~	~~0.47~~	~~0.19~~	~~0.087~~	~~0.050~~	~~0.035~~	~~0.018~~

~~TABLE 15.5-6~~

~~ASSUMED ONSITE ATMOSPHERIC DILUTION FACTORS (SEC/M^3)~~
~~FOR THE CONTROL ROOM~~

~~Period, hrs.~~	~~Base χ/Q[(a)] Sec/m^3~~	~~Modifying Factors f_1~~	~~f_2~~	~~f_3~~	~~f_4~~	~~f_5~~	~~f_6~~	~~Final χ/Q[(a)]~~
~~A. For The Pressurization Case~~								
~~0-8~~	~~1.084x10^{-3}~~	~~1~~	~~1~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~7.05x10^{-5}~~
~~8-24~~	~~1.084x10^{-3}~~	~~.83~~	~~.92~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~5.38x10^{-5}~~
~~24-96~~	~~1.084x10^{-3}~~	~~.66~~	~~.84~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~3.91x10^{-5}~~
~~96-720~~	~~1.084x10^{-3}~~	~~.48~~	~~.67~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~2.27x10^{-5}~~
~~B. For The Infiltration Case~~								
~~0-8~~	~~3.01x10^{-3}~~	~~1~~	~~1~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~1.96x10^{-4}~~
~~8-24~~	~~3.01x10^{-3}~~	~~.83~~	~~.92~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~1.49x10^{-4}~~
~~24-95~~	~~3.01x10^{-3}~~	~~.66~~	~~.84~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~1.08x10^{-4}~~
~~96-720~~	~~3.01x10^{-3}~~	~~.48~~	~~.67~~	~~.2~~	~~1~~	~~.5~~	~~.65~~	~~6.29x10^{-5}~~

~~(a) The χ/Q calculated above do not account for credit for dual pressurization inlet and occupancy factors.~~

~~TABLE 15.5-8~~

~~POPULATION DISTRIBUTION~~

~~Onshore Sector Midpoint Directions~~	~~Sector Midpoint Downwind Distance, miles~~						~~Total Sector Population~~
	~~5~~	~~15~~	~~25~~	~~35~~	~~45~~	~~55~~	
~~SSE~~	~~1,014~~	~~4,727~~	~~2,700~~	~~5,433~~	~~1,567~~	~~697~~	~~16,138~~
~~S~~	~~1,000~~	~~4,666~~	~~2,000~~	~~4,234~~	~~466~~	~~466~~	~~12,832~~
~~SSW~~	~~1,367~~	~~20,334~~	~~7,000~~	~~4,933~~	~~1,167~~	~~1,100~~	~~35,901~~
~~SW~~	~~366~~	~~15,666~~	~~5,000~~	~~700~~	~~700~~	~~634~~	~~23,066~~
~~WSW~~	~~840~~	~~26,000~~	~~6,600~~	~~1,767~~	~~1,433~~	~~1,533~~	~~38,173~~
~~W~~	~~474~~	~~10,334~~	~~1,600~~	~~1,066~~	~~734~~	~~900~~	~~15,108~~
~~WNW~~	~~1,843~~	~~20,033~~	~~22,933~~	~~19,734~~	~~16,066~~	~~6,900~~	~~87,509~~
~~NW~~	~~0~~	~~9,700~~	~~21,334~~	~~18,666~~	~~15,334~~	~~6,000~~	~~71,034~~
~~NNW~~	~~0~~	~~0~~	~~21,333~~	~~22,267~~	~~19,133~~	~~6,500~~	~~69,233~~
~~Total Radial Population~~	~~6,904~~	~~111,460~~	~~90,500~~	~~78,800~~	~~56,600~~	~~24,730~~	~~368,944~~

TABLE 15.5-9

SUMMARY OF OFFSITE AND CONTROL ROOM DOSES ~~FROM~~
LOSS OF ELECTRICAL LOAD

~~Thyroid Doses, rem~~

	Dose (TEDE, rem)~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~Regulatory Limit (TEDE, rem)
~~10 CFR Part 100~~	~~300~~	~~300~~
Maximum 2-hour Exclusion Area Boundary Dose[1]	~~0.028~~	~~0.0065~~
~~Design basis case~~		
- Pre-incident iodine Spike	<0.1	2.5
- Accident-Initiated Iodine Spike	<0.1	2.5
30-day Integrated Low Population Zone Dose		
- Pre-incident iodine Spike	<0.1	2.5
- Accident-Initiated Iodine Spike	<0.1	2.5
30-day Integrated Control Room Occupancy Dose		5
- Pre-incident iodine Spike	<0.1	
- Accident-Initiated Iodine Spike	<0.1	
~~Expected case~~	~~5.2×10^{-6}~~	~~8.7×10^{-7}~~

~~Whole Body Doses, rem~~

	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~25~~	~~25~~
~~Design basis case~~	~~2.3×10^{-3}~~	~~2.3×10^{-4}~~
~~Expected case~~	~~7.2×10^{-7}~~	~~6.9×10^{-8}~~

~~Population Doses, man-rem~~

~~Design basis case~~	~~0.15~~
~~Expected case~~	~~3.8×10^{-5}~~

Note:

1. The maximum 2-hour EAB dose occurs during the following time period :

- Pre-incident iodine Spike	0 - 2 hours
- Accident-Initiated Iodine Spike	8.73 – 10.73 hours

TABLE 15.5-9A
LOSS OF ELECTRICAL LOAD
ANALYSIS ASSUMPTIONS & KEY PARAMETER VALUES

Parameter	Value
Power Level	3580 MWt
Reactor Coolant Mass	446,486 lbm
Primary to Secondary SG tube leakage	0.75 gpm (total for all 4 SGs); leakage density 62.4 lbm/ft^3
Failed/Melted Fuel Percentage	0%
RCS Technical Specification Iodine Levels	Table 15.5-78 (1 μCi/gm DE I-131)
RCS Technical Specification Noble Gas Levels	Table 15.5-78 (270 μCi/gm DE Xe-133)
RCS Equilibrium Iodine Appearance Rates	Table 15.5-79 (1 μCi/gm DE I-131)
Pre-Accident Iodine Spike Concentration	Table 15.5-79 (60 μCi/gm DE I-131)
Accident-Initiated Iodine Spike Appearance Rate	500 times TS equilibrium appearance rate
Duration of Accident-Initiated Iodine Spike	8 hours
Initial Secondary Coolant Iodine Concentrations	0.1 μCi/gm DE I-131 (Table 4.2-1)
Initial and Minimum SG Liquid Mass	92,301 lbm/SG
Time period of tubes uncovered	insignificant
Steam Releases	0-2 hrs: 651,000 lbm 2-8 hrs: 1,023,000 lbm 8-10.73 hrs: same release rate as that for 2-8 hrs
Iodine Partition Coefficient in SGs	100
Iodine Species Released to Environment	97% elemental; 3% organic
Fraction of Noble Gas Released	1.0 (Released without holdup)
Termination of releases from SGs	10.73 hours
Environmental Release Point	MSSVs/10% ADVs
CR emergency Ventilation : Initiation Signal/Timing	Control Room is assumed to remain on normal ventilation for duration of the accident.
Control Room Atmospheric Dispersion Factors	Table 15.5-9B

TABLE 15.5-9B
LOSS OF ELECTRICAL LOAD
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release point and receptor	0-2hr	2-8 hr	8-10.73 hr
MSSVs/10% ADVs to CR NOP Intake (Note 1)	8.60E-04	5.58E-04	5.58E-04
MSSVs/10% ADVs to CR Inleakage (CR Centerline)	2.78E-03	1.63E-03	1.63E-03

Note 1: Due to the proximity of the release from the MSSVs/10% ADVs, to the normal operation CR intake of the affected unit, and due to the high vertical velocity of the steam discharge from the MSSVs/10% ADVs, the resultant plume from the MSSVs/10% ADVs will not contaminate the normal operation CR intake of the affected unit. Thus the χ/Q s presented reflect those applicable to the CR intake of the unaffected unit.

~~TABLE 15.5-10~~ ~~Sheet 1 of 1~~

~~SUMMARY OF OFFSITE DOSES FROM A SMALL LOSS-OF-COOLANT ACCIDENT~~

~~NO FUEL DAMAGE~~

~~Thyroid Doses, rem~~

	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~300~~	~~300~~
~~Design basis case~~	~~2.0×10^{-4}~~	~~2.7×10^{-5}~~
~~Expected case~~	~~9.0×10^{-7}~~	~~1.2×10^{-7}~~

~~Whole Body Doses, rem~~

	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~25~~	~~25~~
~~Design basis case~~	~~1.8×10^{-4}~~	~~5.4×10^{-5}~~
~~Expected case~~	~~4.4×10^{-6}~~	~~1.4×10^{-6}~~

~~Population Doses, man-rem~~

~~Design basis case~~	~~0.36~~
~~Expected case~~	~~0.013~~

~~TABLE 15.5-11~~

~~SUMMARY OF OFFSITE DOSES FROM AN UNDERFREQUENCY ACCIDENT~~

~~Thyroid Doses, rem~~

	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~300~~	~~300~~
~~Design basis case~~	~~0.021~~	~~0.0066~~
~~Expected case~~	~~4.0×10^{-6}~~	~~1.2×10^{-6}~~

~~Whole Body Doses, rem~~

	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~25~~	~~25~~
~~Design basis case~~	~~0.0018~~	~~2.2×10^{-4}~~
~~Expected case~~	~~5.3×10^{-7}~~	~~6.6×10^{-8}~~

~~Population Doses, man-rem~~

~~Design basis case~~	~~0.15~~
~~Expected case~~	~~4.3×10^{-5}~~

~~TABLE 15.5-12~~

~~SUMMARY OF OFFSITE DOSES FROM A SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL~~

	~~Thyroid Doses, rem~~	
	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~300~~	~~300~~
~~Design basis case~~	~~0.12~~	~~0.043~~
~~Expected case~~	~~9.5×10^{-5}~~	~~3.4×10^{-5}~~
	~~Whole Body Doses, rem~~	
	~~Site Boundary - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~25~~	~~25~~
~~Design basis case~~	~~6.1×10^{-3}~~	~~6.7×10^{-4}~~
~~Expected case~~	~~6.5×10^{-6}~~	~~6.9×10^{-7}~~
	~~Population Doses, man-rem~~	
~~Design basis case~~	~~0.42~~	
~~Expected case~~	~~4.3×10^{-4}~~	

~~TABLE 15.5-13~~

~~CALCULATED ACTIVITY RELEASES FROM LOCA - EXPECTED CASE (CURIES)~~

~~Nuclide~~	~~0-2 hr~~	~~2-8 hr~~	~~8-24 hr~~	~~24-96 hr~~	~~4-30 Days~~
~~I-131~~	~~0.4105E-01~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-132~~	~~0.6675E-02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-133~~	~~0.3110E-01~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-134~~	~~0.7392E-02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-135~~	~~0.1629E-01~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-131ORG~~	~~0.1424E-01~~	~~0.3356E-01~~	~~0.4704E-01~~	~~0.1384E-01~~	~~0.1660E-03~~
~~I-132ORG~~	~~0.1780E-02~~	~~0.1556E-02~~	~~0.2210E-03~~	~~0.4311E-06~~	~~0.6136E-17~~
~~I-133ORG~~	~~0.1049E-01~~	~~0.2211E-01~~	~~0.2334E-01~~	~~0.3542E-02~~	~~0.5058E-05~~
~~I-134ORG~~	~~0.1314E-02~~	~~0.2862E-03~~	~~0.1670E-05~~	~~0.9077E-12~~	~~0.0~~
~~I-135ORG~~	~~0.5139E-02~~	~~0.8365E-02~~	~~0.4731E-02~~	~~0.1933E-03~~	~~0.1730E-08~~
~~I-131PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-132PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-133PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-134PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-135PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~Kr-83M~~	~~0.3823E 00~~	~~0.3085E 00~~	~~0.3684E-01~~	~~0.4757E-04~~	~~0.1063E-15~~
~~Kr-65~~	~~0.5356E 01~~	~~0.1598E 02~~	~~0.4257E 02~~	~~0.9571E 02~~	~~0.8243E 03~~
~~Kr-85M~~	~~0.1750E 01~~	~~0.2889E 01~~	~~0.1689E 01~~	~~0.7387E-01~~	~~0.8775E-06~~
~~Kr-87~~	~~0.1285E 01~~	~~0.6227E 00~~	~~0.2430E-01~~	~~0.1922E-05~~	~~0.1515E-22~~
~~Kr-88~~	~~0.3503E 01~~	~~0.4192E 01~~	~~0.1180E 01~~	~~0.1097E-01~~	~~0.1648E-09~~
~~Xe-133~~	~~0.5617E 02~~	~~0.1648E 03~~	~~0.4139E 03~~	~~0.7359E 03~~	~~0.1469E 04~~
~~Xe-133M~~	~~0.9285E 00~~	~~0.2650E 01~~	~~0.6162E 01~~	~~0.8236E 01~~	~~0.5596E 01~~
~~Xe-135~~	~~0.6662E 01~~	~~0.1490E 02~~	~~0.1826E 02~~	~~0.3887E 01~~	~~0.1721E-01~~
~~Xe-135M~~	~~0.1289E 00~~	~~0.6268E-03~~	~~0.7107E-10~~	~~0.1070E-28~~	~~0.0~~
~~Xe-138~~	~~0.3957E 00~~	~~0.1035E-02~~	~~0.1840E-10~~	~~0.1979E-31~~	~~0.0~~

~~TABLE 15.5-14~~

~~CALCULATED ACTIVITY RELEASES FROM LOCA - DESIGN BASIS CASE (CURIES)~~

~~Nuclide~~	~~0-2 Hr~~	~~2-8 Hr~~	~~2-24 Hr~~	~~24-96 Hr~~	~~4-30 Days~~
~~I-131~~	~~0.2703E 02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-132~~	~~0.3985E 02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-133~~	~~0.6207E 02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-134~~	~~0.7063E 02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-135~~	~~0.5712E 02~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-131ORG~~	~~0.7340E 02~~	~~0.2170E 03~~	~~0.5561E 03~~	~~0.1070E 04~~	~~0.3227E 04~~
~~I-132ORG~~	~~0.8325E 02~~	~~0.8763E 02~~	~~0.1862E 02~~	~~0.9240E-01~~	~~0.8557E-10~~
~~I-133ORG~~	~~0.1639E 03~~	~~0.4314E 03~~	~~0.8078E 03~~	~~0.5263E 03~~	~~0.5383E 02~~
~~I-134ORG~~	~~0.9847E 02~~	~~0.2469E 02~~	~~0.2045E 00~~	~~0.2811E-06~~	~~0.2665E-31~~
~~I-135ORG~~	~~0.1411E 03~~	~~0.2838E 03~~	~~0.2668E 03~~	~~0.3148E 02~~	~~0.1834E 01~~
~~I-131PAR~~	~~0.9175E 02~~	~~0.2713E 03~~	~~0.6951E 03~~	~~0.1338E 04~~	~~0.4033E 04~~
~~I-132PAR~~	~~0.1041E 03~~	~~0.1095E 03~~	~~0.2327E 02~~	~~0.1155E 00~~	~~0.1070E-09~~
~~I-P33PAR~~	~~0.2048E 03~~	~~0.5392E 03~~	~~0.1010E 04~~	~~0.6579E 03~~	~~0.6728E 02~~
~~I-134PAR~~	~~0.1231E 03~~	~~0.3086E 02~~	~~0.2557E 00~~	~~0.3514E-06~~	~~0.3331E-31~~
~~I-135PAR~~	~~0.1764E 03~~	~~0.3548E 03~~	~~0.3335E 03~~	~~0.3935E 02~~	~~0.2293E 01~~
~~Kr-83M~~	~~0.9280E 03~~	~~0.7487E 03~~	~~0.8940E 02~~	~~0.1154E 00~~	~~0.2578E-12~~
~~Kr-85~~	~~0.6379E 02~~	~~0.1913E 03~~	~~0.5097E 03~~	~~0.1145E 04~~	~~0.9827E 04~~
~~Kr-85M~~	~~0.2823E 04~~	~~0.4660E 04~~	~~0.2723E 04~~	~~0.1191E 03~~	~~0.1413E-02~~
~~Kr-87~~	~~0.3847E 04~~	~~0.1864E 04~~	~~0.7273E 02~~	~~0.5752E-02~~	~~0.4530E-19~~
~~Kr-88~~	~~0.7090E 04~~	~~0.8484E 04~~	~~0.2388E 04~~	~~0.2220E 02~~	~~0.3333E-06~~
~~Xe-133~~	~~0.1684E 05~~	~~0.4942E 05~~	~~0.1241E 06~~	~~0.2205E 06~~	~~0.4392E 06~~
~~Xe-133M~~	~~0.4250E 03~~	~~0.1212E 04~~	~~0.2819E 04~~	~~0.3766E 04~~	~~0.2556E 04~~
~~Xe-135~~	~~0.7402E 04~~	~~0.1655E 05~~	~~0.2028E 05~~	~~0.4316E 04~~	~~0.1910E 02~~
~~Xe-135M~~	~~0.8506E 03~~	~~0.4137E 01~~	~~0.4690E-06~~	~~0.7061E-25~~	~~0.0~~
~~Xe-138~~	~~0.2504E 04~~	~~0.6552E 01~~	~~0.1164E-06~~	~~0.1252E-27~~	~~0.0~~

~~TABLE 15.5-15~~

~~THYROID DOSE - 2 HOUR - CONTAINMENT LEAKAGE - EXPECTED CASE (REM)~~

~~Distance From Release Point, meters~~

~~Nuclide~~	~~800~~	~~1200~~	~~2000~~	~~4000~~	~~7000~~	~~10000~~	~~20000~~
~~I-131~~	~~0.7456E-03~~	~~0.4792E-03~~	~~0.2636E-03~~	~~0.1097E-03~~	~~0.5060E-04~~	~~0.3101E-04~~	~~0.1247E-04~~
~~I-132~~	~~0.4383E-05~~	~~0.2817E-05~~	~~0.1549E-05~~	~~0.6445E-06~~	~~0.2974E-06~~	~~0.1823E-06~~	~~0.7332E-07~~
~~I-133~~	~~0.1527E-03~~	~~0.9814E-04~~	~~0.5398E-04~~	~~0.2246E-04~~	~~0.1036E-04~~	~~0.6350E-05~~	~~0.2554E-05~~
~~I-134~~	~~0.2268E-05~~	~~0.1458E-05~~	~~0.8017E-06~~	~~0.3336E-06~~	~~0.1539E-06~~	~~0.9432E-07~~	~~0.3794E-07~~
~~I-135~~	~~0.2479E-04~~	~~0.1593E-04~~	~~0.8763E-05~~	~~0.3646E-05~~	~~0.1682E-05~~	~~0.1031E-05~~	~~0.4147E-06~~
~~I-131ORG~~	~~0.2587E-03~~	~~0.1663E-03~~	~~0.9145E-04~~	~~0.3805E-04~~	~~0.1756E-04~~	~~0.1076E-04~~	~~0.4328E-05~~
~~I-132ORG~~	~~0.1169E-05~~	~~0.7514E-06~~	~~0.4132E-06~~	~~0.1719E-06~~	~~0.7933E-07~~	~~0.4862E-07~~	~~0.1956E-07~~
~~I-133ORG~~	~~0.5149E-04~~	~~0.3310E-04~~	~~0.1820E-04~~	~~0.7573E-05~~	~~0.3495E-05~~	~~0.2142E-05~~	~~0.8615E-06~~
~~I-134ORG~~	~~0.4031E-06~~	~~0.2591E-06~~	~~0.1425E-06~~	~~0.5929E-07~~	~~0.2736E-07~~	~~0.1677E-07~~	~~0.6744E-08~~
~~I-135ORG~~	~~0.7820E-05~~	~~0.5026E-05~~	~~0.2764E-05~~	~~0.1150E-05~~	~~0.5307E-06~~	~~0.3252E-06~~	~~0.1308E-06~~
~~I-131PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-132PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-133PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-134PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-135PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~TOTAL~~	~~0.1249E-02~~	~~0.8030E-03~~	~~0.4416E-03~~	~~0.1837E-03~~	~~0.8479E-04~~	~~0.5196E-04~~	~~0.2090E-04~~

~~TABLE 15.5-17~~

~~THYROID DOSE - 30-DAY - CONTAINMENT LEAKAGE - EXPECTED CASE (REM)~~

	Distance From Release Point, meters						
Nuclide	800	1200	2000	4000	7000	10000	20000
I-131	0.7456E-03	0.4792E-03	0.2636E-03	0.1097E-03	0.5060E-04	0.3101E-04	0.1247E-04
I-132	0.4383E-05	0.2817E-05	0.1549E-05	0.6445E-06	0.2974E-06	0.1823E-06	0.7332E-07
I-133	0.1527E-03	0.9814E-04	0.5398E-04	0.2246E-04	0.1036E-04	0.6350E-05	0.2554E-05
I-134	0.2268E-05	0.1458E-05	0.8017E-06	0.3336E-06	0.1539E-06	0.9432E-07	0.3794E-07
I-135	0.2479E-04	0.1593E-04	0.8763E-05	0.3646E-05	0.1682E-05	0.1031E-05	0.4147E-06
I-131ORG	0.1252E-02	0.7543E-03	0.3960E-03	0.1587E-03	0.7223E-04	0.4452E-04	0.1762E-04
I-132ORG	0.2250E-05	0.1438E-05	0.7882E-06	0.3270E-06	0.1507E-06	0.9242E-07	0.3713E-07
I-133ORG	0.2091E-03	0.1280E-03	0.6797E-04	0.2751E-04	0.1257E-04	0.7734E-05	0.3074E-05
I-134ORG	0.4911E-06	0.3156E-06	0.1736E-06	0.7222E-07	0.3332E-07	0.2042E-07	0.8215E-08
I-135ORG	0.2352E-04	0.1473E-04	0.7955E-05	0.3264E-05	0.1498E-05	0.9202E-06	0.3679E-06
I-131PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-132PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-133PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-134PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-135PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	0.2417E-02	0.1496E-02	0.8016E-02	0.3266E-03	0.1466E-03	0.9195E-04	0.3666E-04

TABLE 15.5-19

WHOLE BODY DOSE - 2 HOUR - CONTAINMENT LEAKAGE - EXPECTED CASE (REM)

	Distance From Release Point, meters						
Nuclide	800	1200	2000	4000	7000	10000	20000
I-131	0.3042E-06	0.1955E-06	0.1075E-06	0.4474E-07	0.2065E-07	0.1265E-07	0.5089E-08
I-132	0.2274E-06	0.1462E-06	0.8039E-07	0.3344E-07	0.1543E-07	0.9457E-08	0.3804E-08
I-133	0.4190E-06	0.2693E-06	0.1481E-06	0.6162E-07	0.2843E-07	0.1742E-07	0.7009E-08
I-134	0.1843E-06	0.1185E-06	0.6517E-07	0.2711E-07	0.1251E-07	0.7667E-08	0.3084E-08
I-135	0.4245E-06	0.2729E-06	0.1501E-06	0.6244E-07	0.2881E-07	0.1766E-07	0.7102E-08
I-131ORG	0.1055E-06	0.6784E-07	0.3731E-07	0.1552E-07	0.7163E-08	0.4389E-08	0.1766E-08
I-132ORG	0.6066E-07	0.3899E-07	0.2144E-07	0.8921E-08	0.4117E-08	0.2523E-08	0.1015E-08
I-133ORG	0.1413E-06	0.9082E-07	0.4995E-07	0.2078E-07	0.9589E-08	0.5876E-08	0.2364E-08
I-134ORG	0.3277E-07	0.2106E-07	0.1158E-07	0.4819E-08	0.2224E-08	0.1363E-08	0.5482E-09
I-135ORG	0.1339E-06	0.8608E-07	0.4734E-07	0.1970E-07	0.9089E-08	0.5570E-08	0.2241E-08
I-131PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-132PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-133PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-134PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I-135PAR	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Kr-83M	0.1915E-06	0.1231E-06	0.6771E-07	0.2817E-07	0.1300E-07	0.7965E-08	0.3204E-08
Kr-85	0.1469E-04	0.9439E-05	0.5192E-05	0.2160E-05	0.9967E-06	0.6108E-06	0.2457E-06
Kr-85M	0.9000E-05	0.5784E-05	0.3181E-05	0.1324E-05	0.6108E-06	0.3743E-06	0.1506E-06
Kr-87	0.4797E-04	0.3083E-04	0.1696E-04	0.7056E-05	0.3256E-05	0.1995E-05	0.8026E-06
Kr-88	0.1048E-03	0.6735E-04	0.3704E-04	0.1541E-04	0.7112E-05	0.4358E-05	0.1753E-05
Xe-133	0.1260E-03	0.8097E-04	0.4453E-04	0.1853E-04	0.8550E-05	0.5239E-05	0.2108E-05
Xe-133M	0.2658E-05	0.1708E-05	0.9396E-06	0.3909E-06	0.1804E-06	0.1105E-06	0.4447E-07
Xe-135	0.4764E-04	0.3062E-04	0.1684E-04	0.7006E-05	0.3233E-05	0.1981E-05	0.7969E-06
Xe-135M	0.8720E-06	0.5605E-06	0.3083E-06	0.1282E-06	0.5918E-07	0.3627E-07	0.1459E-07
Xe-138	0.9485E-05	0.6096E-05	0.3353E-05	0.1395E-05	0.6437E-06	0.3945E-06	0.1587E-06
TOTAL	0.3653E-03	0.2348E-03	0.1291E-03	0.5373E-04	0.2479E-04	0.1519E-04	0.6112E-05

~~TABLE 15.5-20 (DELETED)~~

~~TABLE 15.5-21~~

~~WHOLE BODY DOSE - 30-DAY - CONTAINMENT LEAKAGE - EXPECTED CASE (REM)~~

	~~Distance From Release Point, meters~~						
~~Nuclide~~	~~800~~	~~1200~~	~~2000~~	~~4000~~	~~7000~~	~~10000~~	~~20000~~
~~I-131~~	~~0.3042E-00~~	~~0.1955E-06~~	~~0.1075E-06~~	~~0.4474E-07~~	~~0.2065E-07~~	~~0.1265E-07~~	~~0.5089E-08~~
~~I-132~~	~~0.2274E-06~~	~~0.1462E-06~~	~~0.8039E-07~~	~~0.3344E-07~~	~~0.1543E-07~~	~~0.9457E-08~~	~~0.3804E-08~~
~~I-133~~	~~0.4190E-06~~	~~0.2693E-06~~	~~0.1481E-06~~	~~0.6162E-07~~	~~0.2843E-07~~	~~0.1742E-07~~	~~0.7009E-08~~
~~I-134~~	~~0.1843E-06~~	~~0.1185E-06~~	~~0.6517E-07~~	~~0.2711E-07~~	~~0.1251E-07~~	~~0.7667E-08~~	~~0.3084E-08~~
~~I-135~~	~~0.4245E-06~~	~~0.2729E-06~~	~~0.1501E-06~~	~~0.6244E-07~~	~~0.2881E-07~~	~~0.1766E-07~~	~~0.7102E-08~~
~~I-131ORG~~	~~0.5109E-06~~	~~0.3078E-06~~	~~0.1616E-06~~	~~0.6474E-07~~	~~0.2947E-07~~	~~0.1816E-07~~	~~0.7187E-08~~
~~I-132ORG~~	~~0.1167E-06~~	~~0.7463E-07~~	~~0.4090E-07~~	~~0.1697E-07~~	~~0.7822E-08~~	~~0.4796E-08~~	~~0.1927E-08~~
~~I-133ORG~~	~~0.5738E-06~~	~~0.3511E-06~~	~~0.1865E-06~~	~~0.7549E-07~~	~~0.3448E-07~~	~~0.2122E-07~~	~~0.8435E-08~~
~~I-134ORG~~	~~0.3992E-07~~	~~0.2566E-07~~	~~0.1411E-07~~	~~0.5870E-08~~	~~0.2709E-08~~	~~0.1660E-08~~	~~0.6677E-09~~
~~I-135ORG~~	~~0.4028E-06~~	~~0.2522E-06~~	~~0.1362E-06~~	~~0.5590E-07~~	~~0.2566E-07~~	~~0.1576E-07~~	~~0.6301E-08~~
~~I-131PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-132PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-133PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-134PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~I-135PAR~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~Kr-83M~~	~~0.3536E-06~~	~~0.2263E-06~~	~~0.1241E-06~~	~~0.5151E-07~~	~~0.2375E-07~~	~~0.1456E-07~~	~~0.5851E-08~~
~~Kr-85~~	~~0.2189E-03~~	~~0.1162E-03~~	~~0.5620E-04~~	~~0.2106E-04~~	~~0.9086E-05~~	~~0.5697E-05~~	~~0.2150E-05~~
~~Kr-85M~~	~~0.2744E-04~~	~~0.1717E-04~~	~~0.9267E-05~~	~~0.3800E-05~~	~~0.1744E-05~~	~~0.1071E-05~~	~~0.4283E-06~~
~~Kr-87~~	~~0.7159E-04~~	~~0.4596E-04~~	~~0.2526E-04~~	~~0.1050E-04~~	~~0,4846E-05~~	~~0.2970E-05~~	~~0.1195E-05~~
~~Kr-88~~	~~0.2446E-03~~	~~0.1553E-03~~	~~0.8472F-04~~	~~0.3503E-04~~	~~0.1612E-04~~	~~0.9891E-05~~	~~0.3968E-05~~
~~Xe-133~~	~~0.1222E-02~~	~~0.6844E-03~~	~~0.3406E-03~~	~~0.1298E-03~~	~~0.5784E-04~~	~~0.3588E-04~~	~~0.1383E-04~~
~~Xe-133M~~	~~0.2137E-04~~	~~0.1224E-04~~	~~0.6179E-05~~	~~0.2385E-05~~	~~0.1072E-05~~	~~0.6632E-06~~	~~0.2578E-06~~
~~Xe-135~~	~~0.2113E-03~~	~~0.1283E-03~~	~~0.6775E-04~~	~~0.2729E-04~~	~~0.1244E-04~~	~~0.7664E-05~~	~~0.3040E-05~~
~~Xe-135M~~	~~0.8763E-06~~	~~0.5632E-06~~	~~0.3098E-06~~	~~0.1289E-06~~	~~0.5947E-07~~	~~0.3644E-07~~	~~0.1466E-07~~
~~Xe-138~~	~~0.9510E-05~~	~~0.6112E-05~~	~~0.3362E-05~~	~~0.1399E-05~~	~~0.6454E-06~~	~~0.3955E-06~~	~~0.1591E-06~~
~~TOTAL~~	~~0.2031E-02~~	~~0.1169E-02~~	~~0.5949E-03~~	~~0.2319E-03~~	~~0.1041E-03~~	~~0.6441E-04~~	~~0.2510E-04~~

TABLE 15.5-23

SUMMARY OF OFFSITE, CONTROL ROOM & TECHNICAL SUPPORT CENTER DOSES LOSS OF COOLANT ACCIDENT[3] ~~SUMMARY OF EXPOSURE FROM CONTAINMENT LEAKAGE[(a)]~~

	Dose (TEDE, rem)	Regulatory Limit (TEDE, rem)
Maximum 2-hour Exclusion Area Boundary Dose[1]	5.6	25
30-day Integrated Low Population Zone Dose	1	25
30-day Integrated Control Room Occupancy Dose[2]	3.7 (0.7)	5
30-day Integrated TSC Occupancy Dose[2]	4.1 (1.3)	5

~~Thyroid Doses, rem~~

	~~EAB - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~300~~	~~300~~
~~Design basis case~~	~~95.9~~	~~17.7~~
~~Expected case~~	~~1.25×10^{-3}~~	~~9.20×10^{-5}~~

~~Whole Body Doses, rem~~

	~~EAB - 2 Hours~~	~~LPZ - 30 Days~~
~~10 CFR Part 100~~	~~25~~	~~25~~
~~Design basis case~~	~~5.61[(b)]~~	~~0.57[(c)]~~
~~Expected case~~	~~3.65×10^{-4}~~	~~6.44×10^{-5}~~

~~Population Doses, man-rem~~

~~Design basis case~~	~~932.1~~
~~Expected case~~	~~0.269~~

~~Note~~:

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

2. The dose presented represents the operator dose due to occupancy. Value shown in parenthesis represents that portion of the total dose reported that is the contribution of direct shine from contained sources/external cloud.

~~1.~~ The dose received by the operator during transit outside the control room is not a measure of the "habitability" of the control room which is defined by the radiation protection provided to the operator by the control room shielding and ventilation system design. Thus, the estimated dose to the operator during routine post-LOCA access to the control room is addressed separately from the control room occupancy dose and is not included with the control room occupancy dose for the demonstration of control room habitability. As demonstrated in Section 15.5.17.2.4, the dose contribution to the operator during routine access to control room for the duration of the LOCA is minimal. ~~(a) These values correspond to the original analysis. See Table 15.5-75 for current analysis~~

~~2.~~

~~3. (b) The EAB Whole Body dose of 5.61 rem is 3.69 rem gamma and 1.92 rem beta~~

~~4.~~

~~5.~~3. ~~(c) The LPZ Whole Body dose of 0.57 rem is 0.33 rem gamma and 0.24 rem beta~~

~~TABLE 15.5-24~~ ~~Sheet 1 of 5~~

~~ASSUMPTIONS USED TO CALCULATE OFFSITE EXPOSURES FROM POST-LOCA CIRCULATION LOOP LEAKAGE IN THE AUXILIARY BUILDING~~

	~~Expected Small Leakage~~	~~Expected Large Leakage~~	~~DBA Small Leakage~~	~~DBA Large Leakage~~
~~A. ECCS, Containment Fan Cooler, Containment Spray System Operation~~				
~~1. ECCS trains functioning~~	~~2~~	~~2~~	~~2~~	~~2~~
~~2. Containment fan coolers functioning~~	~~5~~	~~5~~	~~2~~	~~2~~
~~3. Containment spray system trains functioning~~	~~2~~	~~2~~	~~1~~	~~1~~
~~B. Activity Deposited in Containment Recirculation Sump Water~~				
~~1. Iodine~~				
~~1. Iodine (Core inventory base on both U-235 & PU-239 fissions)~~	~~100% of gap inventory per Table 11.1-7; (I-127, 129, rel. fract. of 0.015; I-131, 132, 133, 134, 135 rel. fract. Table 11.1-7)~~	~~100% of gap inventory per Table 11.1-7; (I-127, 129, rel. fract. of 0.015; I-131, 132, 133, 134, 135, rel. fract. Table 11.1-7)~~	~~100% of gap inventory per Regulatory Guide 1.25; (I-127, 129, rel. fract. of 0.30; I-131, 132, 133, 134, 135 rel. fract. of 0.10)~~	~~100% of gap inventory per Regulatory Guide 1.25; (I-127, 129, rel. fract. of I-131, 132, 133, 134, 135, rel. fract. of 0.10)~~
~~a. Elemental iodine inventory~~	~~100% of gap iodine inventory~~	~~100% of gap iodine inventory~~	~~99.75% of gap iodine inventory~~	~~99.75% of gap iodine inventory~~
~~(1) I-127~~	~~30.2g, 0 Ci~~	~~30.2g, 0 Ci~~	~~903g, 0 Ci~~	~~903g, 0 Ci~~
~~(2) I-129~~	~~148.5g, 0 Ci~~	~~148.5g, 0 Ci~~	~~4,445g, 0 Ci~~	~~4,445g, 0 Ci~~
~~(3) I-131, 132, 133, 134, 135~~	~~6.5g, 1.82×10^6 Ci~~	~~6.5g, 1.82×10^6 Ci~~	~~97g, 8.45×10^7 Ci~~	~~97g, 8.45×10^7 Ci~~

~~TABLE 15.5-24~~ ~~Sheet 1 of 5~~

~~ASSUMPTIONS USED TO CALCULATE OFFSITE EXPOSURES FROM POST-LOCA CIRCULATION LOOP LEAKAGE IN THE AUXILIARY BUILDING~~

	~~Expected Small Leakage~~	~~Expected Large Leakage~~	~~DBA Small Leakage~~	~~DBA Large Leakage~~
~~b. Organic iodine~~	~~0% of gap iodine inventory~~	~~0% of gap iodine inventory~~	~~0.25% of gap iodine inventory~~	~~0.25% of gap iodine inventory~~
~~(1) I-127~~	~~0.0~~	~~0.0~~	~~2g, 0 Ci~~	~~2g, 0 Ci~~
~~(2) I-129~~	~~0.0~~	~~0.0~~	~~11g, 0 Ci~~	~~11g, 0 Ci~~
~~(3) I-131, 132, 133, 134, 135~~	~~0.0~~	~~0.0~~	~~02g, 2.12x10^{5}Ci~~	~~02g, 2.12x10^{5}Ci~~
~~c. Total iodine~~	~~185.2g, 1.82x10^{6}Ci~~	~~185.2g, 182x10^{6}Ci~~	~~5.458g, 8.47x10^{7}Ci~~	~~5.458g, 8.47x10^{7}Ci~~
~~2. Noble Gases~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~Other fission products~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~C. Containment Recirculation Sump Decay and Cleanup~~				
~~1. Radiological decay credit~~	~~Yes~~	~~Yes~~	~~Yes~~	~~Yes~~
~~2. Cleanup credit~~	~~None~~	~~None~~	~~None~~	~~None~~
~~D. Volume of Water in Which Activity is Deposited (diluted)~~				
~~1. Reactor coolant water, gal.~~	~~93,960~~	~~93,960~~	~~93,960~~	~~93,960~~
~~2. Accumulator water, gal.~~	~~25,040~~	~~25,040~~	~~25,040~~	~~25,040~~

~~TABLE 15.5-24~~ ~~Sheet 1 of 5~~

~~ASSUMPTIONS USED TO CALCULATE OFFSITE EXPOSURES FROM POST-LOCA CIRCULATION LOOP LEAKAGE IN THE AUXILIARY BUILDING~~

	~~Expected Small Leakage~~	~~Expected Large Leakage~~	~~DBA Small Leakage~~	~~DBA Large Leakage~~
~~D. Volume of Water in Which Activity is Deposited (diluted) (Cont'd)~~				
~~3. Refueling water storage tank, gal. (Table 6.3-1)~~	~~350,000~~	~~262,030~~	~~350,000~~	~~254,220~~
~~4. Total, gal.~~	~~469,000~~	~~381,030~~	~~469,000~~	~~373,220~~
~~E. Conditions of Loop Leakage Water~~				
~~1. pH of leakage water (Figure 6.2-15)~~	~~8.8~~	~~8.4~~	~~8.5~~	~~7.85~~
~~2. Temperature of leakage water, °F~~	~~120~~	~~238~~	~~120~~	~~242~~
~~F. Loop Leakage Rate~~	~~1910 cc/hr.~~	~~50 gpm (Table 6.3-9)~~	~~1910 cc/hr~~	~~50 gpm (Table 6.3-9)~~
~~G. Duration of Loop Leakage~~				
~~1. Time after LOCA leakage begins, hr (Table 6.3-5)~~	~~0.337~~	~~0.337~~	~~0.395~~	~~24~~
~~2. Time after LOCA leakage ends, hr~~	~~720~~	~~0.837~~	~~720~~	~~24.5~~
~~3. Total duration of loop leakage, hr~~	~~719.7~~	~~0.5~~	~~719.6~~	~~0.5~~

~~TABLE 15.5-24~~ ~~Sheet 1 of 5~~

~~ASSUMPTIONS USED TO CALCULATE OFFSITE EXPOSURES FROM POST-LOCA CIRCULATION LOOP LEAKAGE IN THE AUXILIARY BUILDING~~

	~~Expected Small Leakage~~	~~Expected Large Leakage~~	~~DBA Small Leakage~~	~~DBA Large Leakage~~
~~H. Auxiliary Building Iodine Decontamination Factors~~				
~~1. Elemental iodine decontamination factor~~				
~~a. M vapor, lbm / M liquid, lbm~~	~~–~~	~~2.78×102^{-2}~~	~~–~~	~~3.22×10^{-2}~~
~~b. V vapor, ft^3/lbm / V liquid, ft^3/lbm~~	~~–~~	~~$1.60 \times 10^{+3}$~~	~~–~~	~~$1.60 \times 10^{+3}$~~
~~c. Partition coefficient, PC, (g/1) liquid / (g/1) gas~~	~~–~~	~~$7.22 \times 10^{+5}$~~	~~–~~	~~$6.77 \times 10^{+3}$~~
~~d. Partition factor, PF, (g) gas / (g) liquid~~	~~–~~	~~6.18×10^{-5}~~	~~–~~	~~7.62×10^{-3}~~
~~e. Decontamination factor, DF, (g) leak / (g) gas~~	~~1.0~~	~~$1.62 \times 10^{+4}$~~	~~1.0~~	~~$1.32 \times 10^{+2}$~~
~~2. Organic iodine decontamination factor, DF, (g) leak / (g) gas~~	~~1.0~~	~~1.0~~	~~1.0~~	~~1.0~~
~~I. Auxiliary Building Decay, Plateout, and Filter Removal~~				
~~1. Radiological decay credit~~	~~None~~	~~None~~	~~None~~	~~None~~
~~2. Plateout credit~~	~~None~~	~~None~~	~~None~~	~~None~~

~~TABLE 15.5-24~~ ~~Sheet 1 of 5~~

~~ASSUMPTIONS USED TO CALCULATE OFFSITE EXPOSURES FROM POST-LOCA CIRCULATION LOOP LEAKAGE IN THE AUXILIARY BUILDING~~

	~~Expected Small Leakage~~	~~Expected Large Leakage~~	~~DBA Small Leakage~~	~~DBA Large Leakage~~
~~I. Auxiliary Building Decay, Plateout, and Filter Removal (Cont'd)~~				
~~3. Auxiliary building filter credit~~	~~None~~	~~Yes~~	~~None~~	~~Yes~~
~~a. Iodine filter efficiency~~				
~~(1) Elemental iodine, %~~	~~0.0~~	~~99.0~~	~~0.0~~	~~90.0~~
~~(2) Organic iodine, %~~	~~0.0~~	~~85.0~~	~~0.0~~	~~70.0~~
~~(3) Particulate iodine %~~	~~0.0~~	~~99.0~~	~~0.0~~	~~90.0~~
~~b. Noble gases~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~J. Atmospheric Dispersion~~				
~~1. Down wind radiological decay credit~~	~~None~~	~~None~~	~~None~~	~~None~~
~~2. Atmospheric dilution factors~~	~~Table 15.5-4~~	~~Table 15.5-4~~	~~Table 15.5-4~~	~~Table 15.5-4~~
~~K. Breathing Rates~~	~~Table 15.5-7~~	~~Table 15.5-7~~	~~Table 15.5-7~~	~~Table 15.5-7~~

~~TABLE 15.5-26~~

~~PERCENTAGE OCCURRENCE OF WIND DIRECTION AND CALM WINDS EXPRESSED AS PERCENTAGE OF TOTAL HOURLY OBSERVATIONS WITHIN EACH SEASON AT THE SITE (250-FOOT LEVEL)~~

~~Wind Direction~~

~~Season~~ [(a)]	~~Offshore~~ [(b)]	~~Onshore~~ [(c)]	~~Calm~~ [(d)]
~~Annual~~	~~57%~~	~~38%~~	~~5%~~
~~Dry~~	~~55%~~	~~40%~~	~~5%~~
~~Wet~~	~~54%~~	~~42%~~	~~4%~~
~~Transitional~~	~~62%~~	~~34%~~	~~4%~~

~~(a) Dry Season - May through September~~
~~Wet Season - November through March~~
~~Transitional - April and October~~

~~(b) Offshore wind directions are defined as wind directions from northwest through east southeast measured clockwise.~~

~~(c) Onshore wind directions are defined as wind directions from southeast through west-northwest measured clockwise.~~

~~(d) Calm wind directions are defined as winds with speeds less than one 1 mph.~~

~~TABLE 15.5-27~~

~~DIABLO CANYON POWER PLANT SITE PROBABILITY OF PERSISTENCE OFFSHORE WIND DIRECTION SECTORS (250-FOOT LEVEL)~~

Conse-cutive Hours	NNE				NE				ENE				E				ESE			
	A[(a)]	D[(a)]	W[(a)]	T[(a)]	A	D	W	T	A	D	W	T	A	D	W	T	A	D	W	T
1	0.461	0.652	0.432	0.390	0.426	0.655	0.373	0.477	0.505	0.769	0.401	0.732	0.678	0.906	0.577	0.840	0.400	0.430	0.370	0.434
2	0.245	0.107	0.327	0.178	0.252	0.241	0.237	0.318	0.190	0.231	0.180	0.195	0.158	0.038	0.189	0.160	0.152	0.180	0.154	0.110
3	0.129	0.054	0.039	0.248	0.136	0.103	0.151	0.102	0.133	–	0.176	0.073	0.069	0.057	0.090	–	0.082	0.075	0.068	0.124
4	0.073	0.071	0.049	0.118	0.041	–	0.047	0.045	0.064	–	0.090	–	0.066	–	0.100	–	0.086	0.120	0.046	0.138
5	0.058	0.045	0.077	0.030	0.041	–	0.044	0.057	0.064	–	0.090	–	0.000	–	0.000	–	0.058	0.100	0.028	0.069
6	0.010	0.000	0.000	0.036	0.050	–	0.071	–	0.019	–	0.027	–	0.000	–	0.000	–	0.060	0.060	0.034	0.124
7	0.012	0.000	0.022	–	0.000	–	0.000	–	0.000	–	0.000	–	0.000	–	0.000	–	0.050	0.035	0.080	–
8	0.013	0.071	–	–	0.033	–	0.047	–	0.025	–	0.036	–	0.000	–	0.000	–	0.023	–	0.046	–
9	–	–	–	–	0.000	–	0.000	–	–	–	–	–	–	–	0.045	–	0.013	–	0.026	–
10	–	–	–	–	0.021	–	0.030	–	–	–	–	–	–	–	–	–	0.014	–	0.028	–
11	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	0.000	–	0.000	–
12	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	0.000	–	0.000	–
13	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	0.037	–	0.074	–
14	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	0.000	–	0.000	–
15	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	0.000	–	0.000	–
16	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	0.023	–	0.046	–
17	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
18	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
19	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
20	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–

~~TABLE 15.5-27~~

~~DIABLO CANYON POWER PLANT SITE PROBABILITY OF PERSISTENCE OFFSHORE WIND DIRECTION SECTORS (250-FOOT LEVEL)~~

Conse-cutive Hours	NW				NNW				N				Calm			
	A	D	W	T	A	D	W	T	A	D	W	T	A	D	W	T
1	0.105	0.086	0.157	0.105	0.364	0.453	0.318	0.348	0.504	0.701	0.461	0.442	0.364	0.329	0.393	0.426
2	0.091	0.073	0.142	0.090	0.194	0.194	0.201	0.178	0.200	0.124	0.217	0.231	0.237	0.214	0.239	0.313
3	0.078	0.064	0.137	0.056	0.125	0.115	0.135	0.118	0.088	0.051	0.078	0.144	0.152	0.166	0.151	0.101
4	0.067	0.058	0.111	0.049	0.085	0.089	0.103	0.042	0.104	0.090	0.122	0.077	0.103	0.081	0.164	0.054
5	0.058	0.040	0.086	0.085	0.067	0.051	0.071	0.078	0.018	0.000	0.022	0.048	0.055	0.074	0.016	0.068
6	0.062	0.048	0.098	0.068	0.036	0.036	0.046	0.016	0.049	0.034	0.052	0.058	0.018	0.022	0.000	0.041
7	0.050	0.045	0.046	0.068	0.034	0.028	0.045	0.018	0.016	–	0.030	–	0.014	0.026	0.000	–
8	0.047	0.039	0.046	0.072	0.039	0.032	0.021	0.084	0.009	–	0.017	–	0.000	0.000	0.000	–
9	0.045	0.044	0.029	0.059	0.016	–	0.023	0.024	0.000	–	–	–	0.027	0.050	0.000	–
10	0.038	0.044	0.008	0.049	0.006	–	0.000	0.026	0.012	–	–	–	0.030	0.037	0.031	–
11	0.046	0.060	0.009	0.054	0.007	–	0.000	0.029	–	–	–	–	–	–	–	–
12	0.035	0.028	0.049	0.039	0.007	–	0.016	0.000	–	–	–	–	–	–	–	–
13	0.038	0.054	0.011	0.011	0.000	–	–	0.000	–	–	–	–	–	–	–	–
14	0.038	0.043	0.011	0.045	0.000	–	–	0.000	–	–	–	–	–	–	–	–
15	0.019	0.027	0.000	0.012	0.009	–	–	0.039	–	–	–	–	–	–	–	–
16	0.020	0.025	0.000	0.026	0.010	–	–	–	–	–	–	–	–	–	–	–
17	0.022	0.031	0.000	0.014	–	–	–	–	–	–	–	–	–	–	–	–
18	0.023	0.033	0.015	0.015	–	–	–	–	–	–	–	–	–	–	–	–
19	0.030	0.034	0.016	0.046	–	–	–	–	–	–	–	–	–	–	–	–
20	0.013	0.021	–	0.000	–	–	–	–	–	–	–	–	–	–	–	–
21	0.003	0.005	–	0.000	–	–	–	–	–	–	–	–	–	–	–	–
22	0.007	0.006	–	0.018	–	–	–	–	–	–	–	–	–	–	–	–
23	0.004	0.006	–	0.000	–	–	–	–	–	–	–	–	–	–	–	–
24	0.012	0.012	–	0.020	–	–	–	–	–	–	–	–	–	–	–	–
25	0.012	0.019	–	–	–	–	–	–	–	–	–	–	–	–	–	–
26	0.012	0.020	–	–	–	–	–	–	–	–	–	–	–	–	–	–
27	0.004	0.007	–	–	–	–	–	–	–	–	–	–	–	–	–	–
28	0.004	0.007	–	–	–	–	–	–	–	–	–	–	–	–	–	–
29	0.000	0.000	–	–	–	–	–	–	–	–	–	–	–	–	–	–
30	0.000	0.000	–	–	–	–	–	–	–	–	–	–	–	–	–	–
31	0.005	0.008	–	–	–	–	–	–	–	–	–	–	–	–	–	–

~~(a) A = Annual~~
~~D = Dry season (May through September)~~
~~W = Wet season (November through March)~~
~~T = Transitional months (April and October)~~

~~TABLE 15.5-28~~

~~ASSUMPTIONS USED TO CALCULATE ONSHORE CONTROLLED CONTAINMENT VENTING~~

	~~Expected Case~~	~~DBA Case~~
~~A. Activity Released to Containment Atmosphere~~		
~~1. Iodine~~	~~25% of gap iodine inventory~~	~~25% of core iodine inventory~~
~~a. Elemental~~	~~24.95% of gap iodine inventory~~	~~22.75% of core iodine inventory~~
~~b. Organic~~	~~0.05% of gap iodine inventory~~	~~1.0% of core iodine inventory~~
~~c. Particulate~~	~~0% of gap iodine inventory~~	~~1.25% of core iodine inventory~~
~~2. Noble gases~~	~~100% of gap inventory~~	~~100% of core inventory~~
~~3. Other fission products~~	~~None~~	~~None~~
~~B. Decay, Cleanup, and Leakage in Containment Atmosphere~~		
~~1. Radiological decay credit~~	~~Yes~~	~~Yes~~
~~2. Iodine spray cleanup~~		
~~a. Elemental~~	$\cancel{92.0\ hr^{-1}}$	$\cancel{31.0\ hr^{-1}}$~~(a)~~
~~b. Organic~~	$\cancel{0.58\ hr^{-1}}$	$\cancel{0\ hr^{-1}}$
~~c. Particulate~~	~~0~~	~~0~~
~~3. Filter cleanup of containment atmosphere~~		
~~a. Iodines~~	~~None~~	~~None~~
~~b. Noble gases~~	~~None~~	~~None~~
~~4. Containment leak rate~~	~~0.05%/per day~~	~~0.05%/per day~~

~~TABLE 15.5-28~~

~~ASSUMPTIONS USED TO CALCULATE ONSHORE CONTROLLED CONTAINMENT VENTING~~

	~~Expected Case~~	~~DBA Case~~
~~C. Containment Atmosphere Volume~~	~~2.68 x 10^6 cubic feet~~	~~2.68 x 10^6 cubic feet~~
~~D. Purge Schedule~~		
~~1. Time after LOCA purging begins~~	~~1968 hours, Chapter 6~~	~~672 hours, Chapter 6~~
~~2. Time after LOCA purging ends~~	~~6792 hours, remainder of 1 yr.~~	~~8088 hours, remainder of 1 yr.~~
~~E. Purge Flowrate~~	~~10 cfm, Chapter 6~~	~~25 cfm, Chapter 6~~
~~F. Filter Efficiency~~		
~~1. Iodines~~		
~~a. Elemental~~	~~99%~~	~~90%~~
~~b. Organic~~	~~85%~~	~~70%~~
~~c. Particulate~~	~~99%~~	~~90%~~
~~2. Noble gases~~	~~None~~	~~None~~
~~G. Atmospheric Dispersion~~		
~~1. Radiological decay credit~~	~~None~~	~~None~~
~~2. χ/Qs~~	~~Table 15.5-30~~	~~Table 15.5-30~~
~~H. Breathing Rates~~	~~Table 15.5-7~~	~~Table 15.5-7~~

~~(a) Although a subsequent safety evaluation showed that the Design Case coefficient of 31^{-1} (for 2600 gpm spray header flow) should be reduced to approximately 29 hr^{-1} (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).~~

~~TABLE 15.5-29~~

~~ONSHORE CONTROLLED CONTAINMENT VENTING EXPOSURES~~

	~~DBA~~	~~Expected~~
~~Thyroid exposure at site boundary (800 meters), rem~~	~~2.21~~	~~9.83×10^{-25}~~
~~Whole body exposure at site boundary (800 meters), rem~~	~~0.0841~~	~~7.15×10^{-3}~~

~~TABLE 15.5-30~~

~~ATMOSHPERIC DISPERSION FACTORS FOR ONSHORE CONTROLLED CONTAINMENT VENTING (STABILITY CATEGORY D)~~

~~Distance, km~~	~~χ/Q, sec/m^3~~
~~0.8~~	~~1.437×10^{-6}~~
~~1.2~~	~~2.440×10^{-6}~~
~~2.0~~	~~1.968×10^{-6}~~
~~4.0~~	~~7.884×10^{-7}~~
~~7.0~~	~~3.135×10^{-7}~~
~~10.0~~	~~1.691×10^{-7}~~
~~20.0~~	~~6.099×10^{-8}~~

~~Meteorological Input Parameters:~~

~~Height of release = 70 meters~~
~~Mixing depth = 350 meters~~
~~Mean wind speed = 5.8 meters per second~~
~~Sigma theta = 10 degrees~~
~~Sigma phi = 3 degrees~~
~~Vertical expansion rate beta, β, = 0.9~~
~~Azimuth expansion rate alpha, α, = 0.9~~
~~$\sigma_y = \sigma_\theta x^\delta$ and $\sigma_z = \sigma_\phi x^\beta$~~

TABLE 15.5-23A
LOSS OF COOLANT ACCIDENT
Assumptions & Key Parameter Values

Parameter	Value
Core Power Level (105% of the rated power of 3411 MWth)	3580 MWt
Fuel Activity Release Fractions	Per Reg. Guide 1.183 (See Section 15.5.17.2.2.2)
Fuel Release Timing (gap)	Onset: 30 sec
	Duration: 0.5 hr
Fuel Release Timing (Early-In-Vessel)	Onset: 0.5 hr
	Duration: 1.3 hr
Core Activity	Table 15.5-77
Chemical Form of Iodine released from fuel to containment atmosphere	4.85% elemental 95% particulate 0.15% organic
Chemical Form of Iodine Released from RCS and sump water	97% elemental 3% organic
Containment Vacuum/Pressure Relief Parameters	
Minimum Containment Free Volume:	2.550E+06 ft^3
Primary Coolant Tech Spec Activity	Table 15.5-78
Chemical Form of Iodine Released	97% elemental; 3% organic
Maximum RCS flash fraction after LOCA Noble Gases Halogens	 100% 40%
Maximum containment pressure relief line air flow rate	218 actual cubic feet per second (acfs)
Maximum duration of release via containment pressure relief line	13 sec
Release Point	Plant Vent

Containment Leakage Parameters	
Containment Spray Coverage – Injection Spray and Recirculation Spray Modes: Sprayed Volume Unsprayed Volume	82.5% (sprayed fraction) 2.103E+06 ft^3 4.470E+05 ft^3
Minimum mixing flow rate from unsprayed to sprayed region: Before actuation of CFCUs After actuation of CFCUs	 2 unsprayed regions/hr 9.13 unsprayed regions/hr
Minimum duration of mixing via CFCUs	Start = 86 sec End = 30 days
Containment spray in injection mode Initiation time Termination time	 111 sec 3798 sec
Maximum delay between end of injection spray and initiation of recirculation spray	12 min (based on manual operator action)
Containment spray in recirculation mode Initiation time Termination time	 4518 sec 22,518 sec
Long-term Sump Water pH	≥ 7.5
Maximum allowable DF for fission product removal	Elemental Iodine: 200 Others: not applicable
Elemental iodine and particulate spray removal coefficients in sprayed region during both injection spray and recirculation spray modes	See Table 6.2-32
Elemental iodine removal coefficients due to wall deposition	See Table 6.2-32
Particulate removal coefficients in unsprayed region due to gravitational settling	See Table 6.2-32
Containment Leak rate (0-24 hr)	0.1% weight fraction per day
Containment Leak rate (1-30 day)	0.05% weight fraction per day

Containment Leakage Release Point (Unfiltered)	From the worst case release point of the following: Diffuse source via the containment wall Via Plant Vent Via Containment Pen Area GE Via Containment Pen Areas GW & FW
ESF System Environmental Leakage Parameters	
Minimum post-LOCA containment water volume sources	480,015 gal.
Minimum time after LOCA when recirculation is initiated	829 sec
Duration of leakage	30 days
Maximum ECCS fluid temperature after initiation of recirculation	259.9 °F
Maximum ECCS leak rate (including safety factor of 2)	Unfiltered via plant vent = 240 cc/min Unfiltered via Containment Penetration Areas GE or GW & FW = 12 cc/min
RHR pump seal failure	Filtered(1) via plant vent 50 gpm starting at t = 24 hrs for 30 min
Iodine Airborne Release Fraction	10%
Auxiliary Building ESF Ventilation System filter efficiency	Elemental iodine: 88% Organic iodine: 88%
Refueling Water Storage Tank (RWST) Back-Leakage Parameters	
Earliest initiation time of RWST back-leakage	829 sec
Maximum ECCS / sump water back-leakage rate to RWST (includes safety factor of 2)	2 gpm
RWST back-leakage iodine release fractions	See Table 15.5-23C
RWST back-leakage noble gas, as iodine daughters, release rate from the RWST vent	See Table 15.5-23C

Miscellaneous Equipment Drain Tank (MEDT) Leakage Parameters	
MEDT inflow rate (includes safety factor of 2)	1900 cc/min
MEDT leakage Iodine release fractions	See Table 15.5-23D
MEDT leakage noble gas, as iodine daughters release rate from plant vent	See Table 15.5-23D
CR Emergency Ventilation: Initiation Signal/Timing	
Initiation time (signal)	SI signal generated: 6 sec Non-Affected Unit NOP Intake Isolated: 18 sec Affected Unit NOP Intake Isolated and CRVS Mode 4 in full Operation: 44.2 sec
Bounding Control Room Atmospheric Dispersion Factors for LOCA	Table 15.5-23B

Note:
Releases from the RHR Pump Seal failure are filtered for CR dose evaluation and Site Boundary Dose Evaluation.

15.5-23B
LOSS OF COOLANT ACCIDENT
Control Room Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
Control Room Normal Intakes					
Plant Vent Release					
- Affected Unit Intake	1.67E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	9.10E-04	-----	-----	-----	-----
Containment Penetration Areas					
- Affected Unit Intake	6.84E-03	-----	-----	-----	-----
- Non-Affected Unit Intake	2.24E-03	-----	-----	-----	-----
Control Room Infiltration					
Plant Vent	1.26E-03	8.96E-04	3.44E-04	3.44E-04	2.99E-04
Containment Penetration Areas	3.22E-03	1.85E-03	7.29E-04	7.15E-04	6.64E-04
RWST Vent	1.07E-03	5.80E-04	2.18E-04	2.19E-04	1.79E-04
Control Room Pressurization Intake					
Plant Vent	5.65E-05	3.70E-05	1.35E-05	1.37E-05	1.11E-05
Containment Penetration Areas	6.45E-05	4.05E-05	1.65E-05	1.38E-05	1.12E-05
RWST Vent	5.25E-05	3.03E-05	1.15E-05	1.10E-05	8.83E-06

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

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

TABLE 15.5-23C
LOSS OF COOLANT ACCIDENT
RWST Iodine Releases Fraction and Gas Venting Rate to Atmosphere

From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Sec	Sec	Fraction $I_{released}$ / $I_{entering}$	Fraction V_{rwst} / day
829	7200	9.451E-05	2.610E+00
7200	28,800	6.357E-05	7.291E-01
28,800	86,400	8.796E-06	7.375E-02
86,400	345,600	4.560E-07	9.955E-03
345,600	471,600	6.347E-07	1.311E-02
471,600	1,011,600	8.231E-07	1.489E-02
1,011,600	2,048,400	1.114E-06	1.547E-02
2,048,400	2592000	1.483E-06	1.702E-02

Where:

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

TABLE 15.5-23D
LOSS OF COOLANT ACCIDENT
MEDT Iodine Release Fraction and Gaseous Venting Rate to Atmosphere

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

Where:

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

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

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

TABLE 15.5-23E
LOSS OF COOLANT ACCIDENT
TSC Limiting Atmospheric Dispersion Factors (sec/m^3)

Release Location / Receptor	0-2 hr	2-8 hr	8-24 hr	24-96 hr	96-720 hr
TSC Normal Intakes					
Plant Vent Release	5.52E-04	-----	-----	-----	-----
Containment Penetration Areas	1.80E-03	-----	-----	-----	-----
RWST Vent	3.63E-04	-----	-----	-----	-----
TSC Infiltration					
Plant Vent	5.43E-04	2.16E-04	9.97E-05	8.11E-05	6.58E-05
Containment Penetration Areas	1.83E-03	7.49E-04	3.16E-04	2.92E-04	2.41E-04
RWST Vent	3.72E-04	1.68E-04	6.64E-05	6.17E-05	5.10E-05
CR/TSC Pressurization Intake					
Plant Vent	-----	3.70E-05	1.35E-05	1.37E-05	1.11E-05
Containment Penetration Areas	-----	4.05E-05	1.65E-05	1.38E-05	1.12E-05
RWST Vent	-----	3.03E-05	1.15E-05	1.10E-05	8.83E-06

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

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

~~TABLE 15.5-31~~

~~CONTROL ROOM INFILTRATION ASSUMED FOR RADIOLOGICAL EXPOSURE CALCULATIONS~~

~~Leakage Path~~	~~Leakage Equation~~	~~Leakage (cfm)~~
~~A. Windows~~	~~No leakage, no windows.~~	~~0.0~~
~~B. Doors~~		~~3~~
~~C. Penetrations~~		
~~1. Ducting (external seal)~~	~~No leakage: ducting penetrations caulked to full depth and exterior surfaces sealed with FLAMEMASTIC 71A and control room will be positively pressurized.~~	~~0.0~~
~~2. Piping (external seal)~~	~~No leakage: concrete walls and floor poured with piping in place and control room will be positively pressurized.~~	~~0.0~~
~~3. Conduits and trays~~		
~~a. External seal~~	~~No leakage: space between exposed conductors and trays is sealed with B&W KAOWOOL ceramic fiber 6 inches in depth, with two coats of FLAMEMASTIC 72A, and control room will be positively pressurized.~~	~~0.0~~
~~b. Internal seal~~	~~No leakage: conduits are sealed with THIXOTROPIC silicone rubber compound, with a minimum depth of one diameter, and control room will be positively pressurized.~~	~~0.0~~

~~TABLE 15.5-31~~

~~CONTROL ROOM INFILTRATION ASSUMED FOR RADIOLOGICAL EXPOSURE CALCULATIONS~~

~~Leakage Path~~	~~Leakage Equation~~	~~Leakage (cfm)~~
~~D. Dampers~~	~~$Q = A \times q \times \Delta p$~~ ~~where:~~ ~~Q = leakage, cfm~~ ~~A = damper area, square feet~~ ~~q = leakage per unit damper area per in. of water~~[(a)] ~~Δp = pressure difference across damper, in. of water~~[(b)]	
~~1. Mode damper #2~~	~~$A = 6.00\ ft^2$, $q = 0.001\ cfm/ft^2$ - in. and $\Delta p = 6.0$ in. W.G.~~	~~<0.05~~
~~2. Mode damper #3~~	~~$A = 1.84\ ft^2$, $q = 0.001\ cfm/ft^2$ - in. and $\Delta p = 6.0$ in. W.G.~~	~~<0.05~~
~~3. Mode damper #7~~	~~$A = 6.00\ ft^2$, $q = 0.001\ cfm/ft^2$ - in. and $\Delta p = 6.0$ in. W.G.~~	~~<0.05~~
~~4. Mode damper #8~~	~~$A = 1.78\ ft^2$, $q = 0.001\ cfm/ft^2$ - in. and $\Delta p = 6.0$ in. W.G.~~	~~<0.05~~
~~E. Total~~		~~≅3~~[(c)]

~~(a) From manufacturer's published data.~~
~~(b) Assume conservatively large value of 6 inches of water; dampers will never see a pressure differential this large.~~
~~(c) 10 cfm is conservatively assumed in the analysis.~~