~~dose limits of 10 CFR Part 20.~~

15.5.17.2.~~10~~ 4 Post LOCA~~-accident~~ Control Room Operator Exposures~~s~~

The design basis for control room ventilation, shielding, and administration is to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE~~whole body, or its equivalent to any part of the body, for the duration of the most severe design basis accident. This basis is consistent with GDC 19, 1971~~.

The control room shielding, described in Section 12.1 is designed to attenuate gamma radiation from post-accident sources to levels consistent with the requirements of GDC 19,~~1971~~1999 and 10 CFR 50.67.

The control room ventilation system is described in Section 9.4.~~1.~~ It is designed to limit the concentration of post-accident activity in the control room air to levels consistent with requirements of GDC 19,~~1971~~1999 and 10 CFR 50.67.

The control room post-accident administration is described in the DCPP Manual. It is to limit post-accident control room personnel exposures to levels consistent with requirements of GDC 19, ~~1971~~1999 and 10 CFR 50.67.

Exposures to control room personnel during post-LOCA occupancy have been estimated for a design basis LOCA to evaluate the adequacy of the control room shielding, the adequacy of the control room ventilation system, and the adequacy of the control room administration in limiting exposures to the specified limits. ~~Exposures have also been calculated for the expected case LBLOCA to obtain a more realistic estimate of exposure to control room personnel.~~

Radiation exposures to personnel in the control room could result from the following sources:

(1) Airborne activity, which infiltrates into the control room

~~(2)~~ Direct gamma radiation~~to the control room from activity in the containment structure~~

~~(3)~~

~~(4)~~(2) ~~Direct gamma radiation to the control room from activity in the containment leakage plume~~ from the external cloud and contained sources.

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 LOCA-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

i. An SIS will be generated at t = 6 sec following a LOCA.

ii. 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=44.2 secs (i.e., 6 + 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.

iii. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=18 secs (i.e., 6 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors:

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a LOCA at either unit are provided in Table 15.5-23B. The χ/Q values presented in Table 15.5-23B take into consideration the various release points-receptors applicable to the LOCA to identify the bounding χ/Q values applicable to a LOCA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2 and summarized in the notes of Table 2.3-147 and Table 2.3-148 for Unit 1 and Unit 2, respectively.

Direct Shine from External and Contained Sources

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

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

1. Direct shine from containment – shine from the airborne source in the containment structure via the bulk shielding (3'-8" thick concrete walls below the bendline, 2'-6" thick concrete dome), including shine through one of the main steam line penetrations and the Personnel Hatch facing the control room.
2. Direct shine from the contaminated cloud outside the control room pressure boundary resulting from containment leakage, ESF system leakage, RWST back-leakage, MEDT leakage - shine occurs through the control room walls, via wall penetrations such as control room doors to the outside, and from the airborne activity in cable spreading room below via control room floor penetrations.

3. Dose due to scattered gamma radiation through wall penetrations from the CRVS filters located in the adjacent mechanical equipment room.
4. Direct shine from the sump fluid that is postulated to collect in the RWST.

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

The direct shine dose estimate in the control room takes into consideration the function of Room 506 (which serves as a control room foyer adjacent to the Shift Supervisor's office), where occupancy is deemed to be minimal; i.e., conservatively estimated at less than 5% of the total time spent daily in the control room. The above "occupancy adjustment" is utilized to determine the maximum 30-day integrated dose in control room (i.e., the total direct shine dose in the control room includes the 30-day dose in Room 506 adjusted by the referenced occupancy factor).
~~The control room ventilation system is designed to minimize infiltration of post-accident airborne activity into the control room complex. Mode 4 operation of the ventilation system provides zone isolation with filtered positive pressurization and filtered recirculation. Mode 4 operation of the ventilation system is initiated automatically and the least contaminated positive pressurization inlet is selected manually as described in Chapter 9.4.1. Both the pressurization and partial recirculation air flow pass through high-efficiency particulate air (HEPA) and charcoal filters.~~

~~In addition to positive pressurization, there are vestibules on control room doors that will minimize infiltration. Table 15.5-31 identifies infiltration pathways and flowrates that have been used in the calculation of post-accident control room radiological exposures.~~

~~Airborne radiation doses inside the control room were evaluated for a DBA LOCA. Regulatory Guide 1.4, Revision 1 was used to determine activity levels in the containment. Activity releases are based on a containment leakage of 0.1 percent/day for the first day and 0.05 percent/day thereafter.~~

~~The containment leakage was assumed to be released unfiltered from the containment building to the atmosphere. Recirculation loop leakages, assumed to be from an RHR pump seal, will pass through charcoal filters and be released to the atmosphere through the main vent at the top of the containment.~~

~~Radioactivity from the atmosphere would enter the control room through two pathways:~~

~~via the pressurization air intakes through charcoal filters~~

~~via infiltration of air inleakage~~

~~The flow rate of pressurization air into the control room is 2100 cfm. The flow rate of recirculated control room air through the charcoal filters is 2100 cfm. Previous analyses had not taken credit for recirculation of control room air. This was an unnecessary conservatism in that a passive failure had already been assumed to occur (RHR pump seal leak) and a second failure is not required.~~

~~A 10 CFM inleakage rate per Standard Review Plan, Section 6.4, was conservatively assumed in the analysis due to the possible pathway through the single doors from the equipment condensing unit areas to the HVAC equipment room. Additionally, an assumed 10-second delay in closure of the CRVS outside air isolation dampers results in 2110 cfm of control room infiltration for the first 10 seconds following the design basis LOCA.~~

~~Table 15.5-32 presents a summary of the parameters used in the analysis.~~

~~The control room shielding is designed to minimize direct gamma radiation (containment shine). Control room exposures resulting from containment shine were estimated using ISOSHLD II. The control room receptor point is 27 feet from the containment structure and protected by an additional 2.5-foot-thick concrete shield. A further contribution to control room direct gamma radiation results from the atmospheric activity cloud external to the control room. Control room exposures resulting from plume shine were estimated using ISOSHLD II. The shine exposure model assumes a parallelepiped radiation source located directly above the control room. The control room receptor point is protected by a 1.5-foot-thick concrete shield.~~

Control Room Operator Dose during Access

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

This approach is consistent with the approach used by other licensees, and is reasonable since a) transit to and from the control room is only expected after the first 24 hours following the accident by which time the airborne levels inside containment has reduced significantly due to the use of active fission product removal mechanisms such as containment sprays, and radioactive decay, and b) the operator is protected from radioactive ESF fluids by the shielding provided by the buildings that house such equipment. In addition, it is expected that during a postulated event, access to the control room will be controlled by Health Physics and the Emergency Plan based on real time data, with the purpose of minimizing personnel dose.

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

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

(1) Airborne activity in the containment leakage plume ~~and~~

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

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

Egress-ingress thyroid and whole body exposures from airborne activity are functions of containment activity, containment leakage, atmospheric dispersion, and excursion time. The EMERALD computer code was used to calculate the airborne activity concentrations, and then conventional exposure equations from Regulatory Guide 1.4, Revision 1, were used to calculate gamma, beta, and thyroid exposures (Reference 6). The exposure from betas ~~is~~ was calculated on the basis of an infinite uniform cloud, and exposure from gammas ~~is~~ was calculated on the basis of a semi-infinite cloud.

Because of the containment shielding and short excursion time, egress-ingress containment shine exposures ~~are~~ were estimated to be small. Egress-ingress containment shine exposures were calculated using ISOSHLD-II. The shine model ~~assumes~~ assumed a cylindrical radiation source having the same radius and height as the containment structure with a 3.5-foot-thick concrete shield surrounding it. The receptor point ~~is~~ was assumed to be a distance of 10 meters from the outer surface of the containment wall.

The e~~E~~stimate~~s~~d ~~of post-accident control room exposures and~~ egress-ingress exposures developed in support of DCPP original licensing basis are listed in Table 15.5-33 and summarized below. ~~The sum of the DBA case exposures are within the specified criteria, and the expected case exposures demonstrate the conservatism of the DBA case exposures.~~

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

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

Subsequent to the original licensing basis assessment described above, DCPP has identified additional post-LOCA fission product release pathways, as discussed in Section 15.5.17.2.1. The postulated effect of these additional radioactivity release paths, as well as the implementation of AST, on the estimated dose to control room personnel during routine egress ingress takes into consideration the following:

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

b. Routine ingress / egress to the control room during the 30 day period following a LOCA falls into the mission dose category as discussed in NUREG 0737, November 1980, Item II.B.2.

c. In accordance with NUREG 0737, November 1980, Item II.B.2 leakage of systems outside containment need not be considered as potential sources.

Based on the above considerations, the dose consequences of the additional activity release paths addressed in Section 15.5.17.2.1 (and listed below), in addition to Regulatory Guide 1.183 is addressed as follows:

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

ii. Containment leakage:

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

    Note: The iodine source term at t=24 hrs is essentially the organic iodines released to the containment which are not effected by sprays, and which per Regulatory Guide 1.183, represent 0.06% of the core iodines (i.e., 0.15% of the 40% core iodines released to containment atmosphere at accident initiation). Also, the essentially particulate nature of the radioactivity release associated with an AST source term, and the effectiveness of particulate removal by sprays / settling makes the dose contribution from the particulate source minimal after t=24 hours.

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

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

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

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

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

iii. The RHR Pump Seal Failure, ESF System Leakage, RWST back leakage and MEDT leakage – All of these releases are based on leakage of systems outside containment. In accordance with NUREG 0737, November 1980, Item II.B.2, the dose contribution due to these sources need not be considered for access calculations.

To address the TEDE dose acceptance criteria applicable to use of AST, the original licensing basis egress-ingress exposures have been updated as noted below in accordance with 10 CFR 20.1003.

10 CFR 20.1003 defines TEDE as the sum of the deep dose equivalent for external exposures (i.e., external whole body exposure) and the committed effective dose equivalent for internal exposures (i.e., sum of the product of the weighting factor applicable to each organ irradiated and the dose to that organ). Per 10 CFR 20.1003, the weighting factor for the whole body is 1.0 and for the thyroid is 0.03. While the weighting factor for beta radiation is undefined, the contribution of the beta dose to the total effective dose equivalent is expected to be insignificant. Therefore,

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

b. Direct radiation from the fission products in the containment structure to control room personnel during egress ingress is 0.022 rem TEDE.

Thus the total dose to the control room operator during access is estimated to be 0.037 rem TEDE. This value is 1% of the estimated operator dose due to control room occupancy following a LOCA (Refer to Table 15.5-23) and is therefore considered to be minimal.

15.5.17.2.5 Post-LOCA Technical Support Center Operator Exposure

In accordance with NUREG-0737, Supplement 1, January 1983, Section 8.2.1(f) the TSC design has been evaluated for the LOCA.

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

The TSC serves both units and is located at El 104' on the south-west side of the Unit 2 turbine building and is shared between Unit 1 and Unit 2.

The nominal TSC air intake flowrate during normal operations is 500 cfm. The air inflow is filtered through a HEPA filter and drawn into the TSC envelope which has a free volume. The TSC normal intake is isolated and the TSC ventilation placed into filtered / pressurized (CRVS Mode 4) operation by manual operator action within 2 hours of the LOCA.

The post-accident pressurization flow to the TSC is provided via the CRVS Mode 4 pressurization intakes (i.e., 1 per unit, each located on either side of the Turbine Building). As noted in Section 15.5.9, the control room pressurization air intakes have dual ventilation outside air intake design. The nominal air intake flowrate during the TSC pressurization mode is 500 cfm.

As discussed in Section 15.5.9, CRVS Mode 4 operation utilizes redundant PG&E Design Class I radiation monitors located at each pressurization air intake and has the provisions of acceptable control logic to automatically select the least contaminated inlet at the beginning of the accident, and manually select the least contaminated inlet during the course of the accident. Thus, during Mode 4 operation the TSC dose consequence analysis can utilize the χ/Q values for the more favorable pressurization air intake reduced by a factor of 4 to credit the "dual intake" design (refer to Section 2.3.5.2.2 for additional details).

The allowable methyl iodide penetration and filter bypass for the TSC Mode 4 Charcoal

Filter is <2.5% and <1%, respectively. Thus in accordance with Generic Letter 99-02, June 1999, the TSC charcoal filter efficiency for elemental and organic iodine used in the TSC dose analysis is 93%. The acceptance criteria for the TSC normal operation and Mode 4 HEPA filters is "penetration plus system bypass" < 1.0%. Thus, using methodology similar to the charcoal filters, the HEPA filter efficiency for particulates used in the TSC dose analysis is 98%.

During TSC Mode 4 operation, the TSC air is also recirculated through the same filtration unit as the pressurization flow (refer to Section 9.4.11). The air flow allowable through the pressurization charcoal / HEPA filter and minimum filtered recirculation flow for the TSC is provided in Table 15.5-82.

Unfiltered inleakage into the TSC during normal operation and Mode 4 is assumed to be 60 cfm (includes 10 cfm for ingress/egress based on the guidance provided in NUREG 0800, SRP 6.4.

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

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

The bounding atmospheric dispersion factors applicable to the radioactivity release points / TSC receptors applicable to a LOCA at either unit are provided in Table 15.5-23E. The χ/Q values presented take into consideration the various release points-receptors applicable to the LOCA to identify the bounding χ/Q values applicable to a LOCA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Section 2.3.5.2.2.

The direct shine dose into the TSC due to the external cloud and contained sources is calculated in a manner similar to that described for the control room in Section 15.5.17.2.4. The LOCA sources that could potentially impact the TSC operator dose due to direct shine are identified below.

1. Direct shine from containment – shine from the airborne source in the containment structure via the bulk shielding (3'-8" thick concrete walls below the bendline, 2'-6" thick concrete dome), including shine through the Personnel Hatch facing the TSC

2. Direct shine from the contaminated cloud outside the TSC pressure boundary resulting from containment leakage, ESF system leakage, RWST back-leakage,

MEDT leakage - shine occurs through the TSC walls and via wall penetrations such as TSC doors to the outside.

3. Dose due to scattered gamma radiation through wall penetrations from the TSC filters located in the adjacent mechanical equipment room and scatter past labyrinths provided for selected doors.

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

Table 15.5-82 lists key assumptions / parameters associated with DCPP TSC design.

The bounding TSC operator dose following a LOCA at either unit is presented in Table 15.5-23.

15.5.17.2.~~11~~ 6 Summary

In the preceding sections, the potential exposures from a major primary system pipe rupture have been calculated for various possible mechanisms:

(1) Containment Pressure / Vacuum Relief

(2) Containment leakage

~~(1)~~(3) ESF System Leakage

~~(2)~~(4) RHR ~~recirculation~~ pump seal Failure~~loop leakage~~

(5) ~~Controlled post-accident containment venting~~RWST Back-Leakage

~~(3)~~(6) MEDT Leakage

~~(4)~~(7) Shine from Contained and External Sources (e.g., Contained Containment shine, RWST Shine, external clouds due to the various leakage sources, etc)

The analyses have been carried out using the models and assumptions specified in ~~regulations 10 CFR Part 100,~~ in Regulatory Guide 1.183~~10 CFR Part 50,~~ and ~~the~~ other regulatory guidance identified ~~safety and regulatory guides~~above. In all analyses, the resulting potential exposures to plant personnel, to individual members of the public, and to the general population have been found to be lower than the applicable guidelines and limits specified in 10 CFR ~~Part 100~~50.67 and Regulatory Guide 1.183~~, 10 CFR Part 50, and 10 CFR Part 20~~.

15.5.17.3 Conclusions

Based on the results discussed, the occurrence of a major pipe rupture in the primary system of a DCPP unit would not constitute an undue risk to the health and safety of the public. In addition, the ESF provided for the mitigation of the consequences of a LBLOCA are adequately designed.

Additionally, 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 0.25 Sv (25 rem) TEDE as shown in Table 15.5-23.

(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 0.25 Sv (25 rem) TEDE as shown in Table 15.5-23.

(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 15.5-23.

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, and in accordance with DCPP current licensing basis, the dose contribution to the operator during routine access to control room for the duration of the accident (0.04 rem TEDE), is not included with the control room occupancy dose for the demonstration of control room habitability

The radiation dose to an individual in the TSC for the duration of the accident is within 0.05 Sv (5 rem) TEDE as shown in Table 15.5-23.

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

~~(1) The radiological consequences of a major rupture of primary coolant pipes shall take into consideration fission product releases due to leakage from the containment, post-LOCA recirculation loop leakage in the Auxiliary Building (inclusive of a RHR pump seal failure resulting in a 50 gpm leak for 30 minutes starting at T=24 hrs post-LOCA), and containment shine as shown in Section 15.5.17.2.11.~~

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

~~i. An individual located at any point on the boundary of the exclusion area for the two hours immediately following the onset of the postulated fission product release shall not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure as shown by the EAB whole body dose reported for containment shine in Section 15.5.17.2.6, and the remaining doses presented in Table 15.5-75.~~

~~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 as shown by the EAB whole body dose reported for containment shine in Section 15.5.17.2.6 (conservative when applied to the LPZ), and the remaining doses presented in Table 15.5-75.~~

~~(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 15.5-33.~~

~~(4) In the event controlled venting of the containment is implemented post-LOCA using the containment hydrogen purge system (serves as a back-up capability for hydrogen control to the hydrogen recombiners), an individual located at any point on the boundary of the exclusion area, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose to the whole body in excess of the annual dose limit of 10 CFR Part 20 as shown in Table 15.5-29.~~

15.5.18 RADIOLOGICAL CONSEQUENCES OF A MAJOR STEAM PIPE RUPTURE

15.5.18.1 Acceptance Criteria

The radiological consequences of a MSLB 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 the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

(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 the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case..

Control Room Dose Criteria

~~(1)~~ 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.~~The radiological consequences of a major steam pipe rupture shall not exceed the dose limits of 10 CFR 100.11 as outlined below:~~

~~(2)~~

~~(3) 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 in excess of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case.~~

~~(4)~~

~~(5) 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 the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case.~~

~~(6)~~

~~(7) 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 for both the pre-accident and the accident- initiated iodine spike cases.~~

15.5.18.2~~-~~ Identification of Causes and Accident Description

15.5.18.2.1 Activity Release Pathways

As reported in Section 15.4.2, a major steam line rupture is not expected to cause cladding damage, and thus no release of fission products to the coolant is expected following this accident. If significant radioactivity exists in the secondary system prior to the accident, however, some of this activity will be released to the environment with the

steam escaping from the pipe rupture. In addition, if an atmospheric steam dump from the unaffected steam generators is necessitated by unavailability of condenser capacity, additional activity will be released. ~~Section 15.5.18.2.1 discusses the main steam line break (MSLB) dose analysis of record which is based on the OSGs. The OSG MSLB dose analysis is bounding for the RSGs as discussed in the following section. (See Table 6.4.2-1 of Reference 49 for a summary of OSG and RSG MSLB steam releases.)~~

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

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

Regulatory requirements provided for the MSLB in pertinent sections of Regulatory Guide 1.183 including Appendix E is used to develop the dose consequence model. Table 15.5-34A lists the key assumptions / parameters utilized to develop the radiological consequences following a MSLB.

~~15.5.18.2.1 Radiological Assessment for Accident-Induced Leakage~~

~~Because tubes in the faulted steam generator encounter a higher differential pressure during steam line rupture conditions than normal operating conditions, there is a potential for primary-to-secondary leakage in degraded tubing to increase to a rate that is higher than that during normal operation. This leakage is referred to as accident-induced leakage. This section provides the updated licensing basis description and radiological consequence analysis for a major steam line rupture analysis using an accident-induced leak rate of 10.5 gpm (at room temperature conditions), which is higher than the operational leakage limit in the Technical Specifications. The NRC approved this analysis in a letter to PG&E dated February 20, 2003, "Issuance of Amendment: RE: Revision to Technical Specification 1.1, 'Definitions, Dose Equivalent I-131,' and Revised Steam Generator Tube Rupture and Main Steam Line Break Analyses." Application of this accident-induced leak rate is governed by SG Program accident-induced leakage performance criteria documented in the Technical Specifications.~~

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

15.5.18.2.2 Activity Release Transport Model

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

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

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

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

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

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

<u>Faulted Steam Generator</u>

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

To maximize the control room and offsite doses following a MSLB, the maximum allowable primary to secondary SG tube leakage for all SGs (0.75 gpm or 1080 gpd at

Standard Temperature and Pressure (STP) conditions), is conservatively assumed to occur in the faulted SG. All iodine and noble gas activities in the referenced tube leakage are released directly to the environment without hold-up or decontamination. The primary to secondary SG tube leakage is assumed to go on until the RCS reaches 212°F, which based on minimum heat transfer rates, is conservatively estimated to occur 30 hours after the event.

Intact Steam Generators

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

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

15.5.18.2.3 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.

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

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

The bounding EAB and LPZ dose following a MSLB at either unit for both scenarios are presented in Table 15.5-34.

15.5.18.2.4 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 MSLB-specific assumptions associated with control room response and activity transport.

Timing for Initiation of CRVS Mode 4:

i. An SIS will be generated at t = 0.6 sec following a MSLB.

ii. 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 within 10 seconds at t=38.8 secs (i.e., 0.6 + 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.

iii. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=12.6 secs (i.e., 0.6 + 2 secs signal processing time + 10 sec damper closure time).

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

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

Secondary Coolant Discharge: The radioactivity release due to the almost immediate dry-out of the faulted SG following a MSLB is based on a) the radioactivity concentration of the iodine in a finite cloud created by the secondary coolant liquid flash at the break point; b) conservation of total iodine activity in the SG liquid. The activity concentration at the release point is conservatively based on saturated steam at a density of 5.98E-04 gm/cm^3, (i.e., at 1 atmosphere and 212^{o}F). The activity concentration entering the control room is assumed to be the same as the concentration at the break point until the control room normal ventilation is isolated and the CRVS re-aligned to Mode 4 Pressurization.

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

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

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

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a MSLB at either unit are provided in Table 15.5-34B. The χ/Q values presented in Table 15.5-34B take into consideration the various release points-receptors applicable to the MSLB to identify the bounding χ/Q values applicable to a MSLB at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 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 MSLB at either unit is presented in Table 15.5-34. ~~The methodology selected for performing the radiological assessment follows NRC SRP 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Revision 2, 1981. Using an accident-induced leak rate of 10.5 gpm (at room temperature conditions) in the faulted SG, calculations using the LOCADOSE computer program demonstrate that the offsite doses are within 10 percent of 10 CFR 100.11 limits and control room doses are within GDC 19, 1971 limits.~~

~~The resultant doses from the MSLB event using an accident-induced leak rate of 10.5 gpm are listed below. The limiting case is the accident initiated iodine spike as the thyroid dose at the EAB is at the 30 rem limit.~~

15.5.18.3 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 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-34.

(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 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-34.

(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 15.5-34.

~~(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 in excess of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case as shown in Section 15.5.18.2.1.~~

~~(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 the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case as shown in Section 15.5.18.2.1.~~

~~(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 for both the pre-existing and the accident-initiated iodine spike cases as shown in Section 15.5.18.2.1.~~

~~As noted in Section 15.5.18.2.1, the above dose estimates reflect the OSGs and an accident induced leak rate of 10.5 gpm. The limiting case is a thyroid dose at the EAB which corresponds to the dose limit of 30 rem for an accident-initiated iodine spike. These dose estimates bound the doses with the RSGs which cannot credit Alternate Repair Criteria (ARC) for the steam generator tubes as the OSGs do.~~

15.5.19 RADIOLOGICAL CONSEQUENCES OF A MAJOR RUPTURE OF A MAIN FEEDWATER PIPE

15.5.19.1 Acceptance Criteria

The radiological consequences of a major rupture of a main feedwater pipe (referred to herein as a feedwater line break (FWLB)) 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 the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

(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 the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

~~100.11 as outlined below:~~

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

~~An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure~~

15.5.19.2 Identification of Causes and Accident Description

As reported in Section 15.4.2, a major feedwater line rupture is not expected to cause cladding damage, and thus no release of fission products to the coolant is expected following this accident. If significant radioactivity exists in the secondary system prior to the accident, however, some of this activity will be released to the environment with the feedwater escaping from the pipe rupture. In addition, if an atmospheric steam dump from the unaffected steam generators is necessitated by unavailability of condenser capacity, additional activity will be released. ~~As discussed in Section 15.5.18, about 1.47E+06 lbm of secondary coolant is the limiting Condition IV event release expected for a full cooldown without any condenser availability.~~

~~The radiological consequences of about 1.47E+06 lbm of secondary coolant release have been discussed in Section 15.5.18.~~
Per Standard Review Plan 15.2.8, Section III, Item 6 (Reference 86), the evaluation of the radiological consequences of a design basis FWLB may be based on a qualitative comparison to the results of the design basis MSLB.

The dose consequences following a FWLB will be bounded by a MSLB since the airborne environmental release via the break point is expected to be less than the MSLB.

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

15.5.19.3 Conclusions

On the basis of this comparison approach, it is concluded that the dose consequences at the EAB and LPZ following a feedwater line break will remain within the acceptance criteria listed in Section 15.5.19.1.~~Based on the results discussed, it can be concluded that potential exposures from major feedwater line ruptures will be well below the guideline levels specified in 10 CFR 100.11, and that the occurrence of such ruptures would not result in undue risk to the public.~~

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

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

~~The radiation dose to the whole body and to the thyroid of an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), are insignificant as shown in Table 15.5-9.~~

15.5.20 RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE ~~(SGTR)~~

15.5.20.1 Acceptance Criteria

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

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

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case.

(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 the 10 CFR 50.67 limit of 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case..

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 steam generator tube rupture shall not exceed the dose guidelines of SRP, Section 15.6.3, Revision 2, 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 in excess of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case, and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case.~~

~~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 in excess of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case, and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case.~~

~~(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 for both the pre-accident and the accident- initiated iodine spike cases.~~

15.5.20.2 Identification of Causes and Accident Description

15.5.20.2.1 Activity Release Pathways

This event is caused by the instantaneous rupture of a SG tube with a resultant release of primary coolant into the lower pressure secondary system. No melt or clad breach is postulated for the SGTR event. The calculation assumes a stuck-open PORV of the ruptured steam generator for 30 minutes. Based on an assumption of a Loss of Offsite Power coincident with reactor trip, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs / 10% ADVs of the intact steam

generators are used to cool down the reactor until initiation of shutdown cooling. A portion of the primary coolant break flow in the ruptured SG flashes and is released a) to the condenser before reactor trip and b) directly to the environment after reactor trip, via the MSSVs and 10% ADVs. The remaining break flow mixes with the secondary side liquid, and is released to the environment via steam releases through MSSVs and 10% ADVs. The activity in the RCS also leaks into the intact steam generators via SG tube leakage and is released to the environment from the MSSVs / 10% ADVs.

Regulatory requirements provided for the SGTR in pertinent sections of Regulatory Guide 1.183 including Appendix F is used to develop the dose consequence model. Table 15.5-64A lists the key assumptions / parameters utilized to develop the radiological consequences following a SGTR. Table 15.5-64C provides the time dependent steam flow from the Ruptured and Intact SGs and the flashed and unflashed break flow in the Ruptured SG.

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

15.5.20.2.2 Activity Release Transport Model

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

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

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

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

DCPP Plant Technical Specification 3.4.13d limits primary to secondary SG tube leakage to 150 gpd per steam generator for a total of 600 gpd in all 4 SGs. To

accommodate any potential accident induced leakage, the SGTR dose consequence analysis addresses a limit of 0.75 gpm from all 4 SGs (or a total of 1080 gpd). To maximize the dose consequences, the analysis conservatively assumes that all of the 0.75 gpm SG tube leakage occurs in the intact SGs.

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

Ruptured Steam Generator

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

In accordance with the requirements provided in Regulatory Guide 1.183, the noble gases in the entire break flow and the iodine in the flashed portion of the break flow are assumed to be immediately available for release from the steam generator. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and the inverse of the allowable partition coefficient of 100. The iodine releases from the SGs are assumed to be 97% elemental and 3% organic.

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

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

Intact Steam Generators

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

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

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

Initial Secondary Coolant Activity Release

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

15.5.20.2.3 Offsite Dose Assessment

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

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

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

The bounding EAB and LPZ dose following a SGTR at either unit for both scenarios are presented in Table 15.5-64.

15.5.20.2.4 Control Room Dose Assessment

~~The SGTR accident is reanalyzed for RSGs and is discussed in Section 15.4.3, and the thermal and hydraulic analysis presented in Section 15.4.3.3 provides the basis for the evaluation of radiological consequences discussed in this section.~~

~~15.5.20.2.1 Offsite Exposures~~

~~The evaluation of the radiological consequences of a steam generator tube rupture event assumes that the reactor has been operating at the maximum allowable Technical Specification (Reference 22) limits for primary coolant activity and 1 gpm primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator via the ruptured tube and primary to secondary leakage, and are released to the atmosphere through the steam generator PORVs (and safety valves) and via the condenser air ejector exhaust and/or the vacuum pump exhaust (if in operation).~~

~~The quantity of radioactivity released to the environment, due to an SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator, and liquid-vapor partitioning in the turbine condenser hot well.~~

~~(1) *Design Basis Analytical Assumptions*~~

~~The major assumptions and parameters used in the analysis are itemized in Table 15.5-64.~~

~~(2) *Source Term Calculations*~~

~~The radionuclide concentrations in the primary and secondary system, prior to and following the SGTR are determined as follows:~~

~~(a) The iodine concentrations in the reactor coolant will be based upon pre-accident and accident initiated iodine spikes.~~

~~(i) Accident Initiated Spike - The initial primary coolant iodine concentration is 1 μCi/gm of Dose Equivalent (DE) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system~~

~~which increases the iodine release rate from the fuel to the coolant to a value 335 times greater than the release rate corresponding to the initial primary system iodine concentration. The initial appearance rate can be written as follows:~~

$$\cancel{P_i = A_i \lambda_i} \qquad \cancel{(15.5\text{-}15)}$$

~~where:~~

$\cancel{P_i}$ ~~= Equilibrium appearance rate for iodine nuclide i~~
$\cancel{A_i}$ ~~= equilibrium RCS inventory of iodine nuclide i corresponding to 1 μCi/gm of DE I-131~~
$\cancel{\lambda_i}$ ~~= removal coefficient for iodine nuclide i~~

~~(j) Pre-accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 μCi/gram of DE I--131.~~

~~(b) The initial secondary coolant iodine concentration is 0.1 μCi/gram of DE I--131.~~

~~(c) The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.~~

~~(d) The initial noble gas concentrations in the reactor coolant are based upon 651 μCi/g of Xe-133 DEC for the noble gasses Kr-85m, Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135m, Xe-135, and Xe-138, using noble gas whole body dose conversion factors documented in FGR 12 (Reference 42) Table III.1, associated with 1 percent fuel defects. The calculation of Xe-133 DEC ignores the contribution from Kr-85 and Xe-131m due to low concentration and small dose conversion factor.~~

~~(3)~~ *~~Radioactivity Transport Analysis~~*

~~The iodine transport analysis considers break flow flashing, steaming, and partitioning. The analysis assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. The analysis conservatively took no credit for scrubbing of iodine contained in the atomized coolant droplets. The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 100~~

~~between the steam generator liquid and steam phases. Droplet removal by the dryers is conservatively assumed to be negligible.~~

~~The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following a SGTR.~~

~~(a) The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the ruptured and intact steam generators to the atmosphere are presented in Table 15.4-14.~~

~~(b) The mass of break flow that flashes to steam and is immediately released to the environment is contained in Table 15.4-14 and is presented in Figure 15.4.3-11. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow is from the hot leg side of the steam generator, whereas the break flow actually consists of flow from both the hot leg and cold leg sides of the steam generator.~~

~~(c) No iodine scrubbing is credited for the break flow that flashes in the analysis and the iodine scrubbing efficiency is assumed to be 0 percent. Thus the location of the tube rupture is not significant for the radiological consequences. However, as discussed in Section 15.4.3.3, in the thermal and hydraulic analysis the tube rupture break flow is calculated conservatively assuming that the break is at the top of the tube sheet.~~

~~(d) The rupture (or leakage) site is assumed to be always covered with secondary water based on Reference 33, which concluded the effect of tube uncovery is essentially negligible for the radiological consequences for the limiting SGTR transient.~~

~~(e) The total primary to secondary leak rate for the 3 intact steam generators is assumed to be 1.0 gpm. The leakage to the intact steam generators is assumed to persist for the duration of the accident.~~

~~(f) The iodine partition coefficient between the liquid and steam of the ruptured steam generator is assumed to be 100 for non-flashed flow and 1 for flashed flow. The iodine partition coefficient between the liquid and steam of the intact steam generator is assumed to be 100.~~

~~(g) The noble gases in the break flow and primary to secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere. The whole body gamma doses are calculated combining the dose from the released noble gases with the dose from the iodine releases.~~

~~(h) For the accident initiated iodine spike case, an iodine spiking factor of 335, obtained from Regulatory Guide 1.195, May 2003 (Reference 44) is assumed.~~

~~(4) *Offsite Dose Calculation*~~

~~In equations 15.5-17 and 15.5-18, no credit is taken for a cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low population zone. Offsite thyroid doses are calculated using the equation:~~

$$\cancel{D_{Th} = \sum_i \left[(DCF)_i \sum_j \left((IAR)_{ij} (BR)_j (\chi/Q)_j \right) \right]} \qquad \text{~~(15.5-17)~~}$$

~~where:~~

~~$(IAR)_{ij}$ = integrated activity of iodine nuclide i released during the time time interval j in Ci~~

~~$(BR)_j$ = breathing rate during time interval j in meter^3/second (Table 15.5-68)~~

~~$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/meter^3 (Table 15.5-68)~~

~~$(DCF)_i$ = thyroid dose conversion factor via inhalation for iodine nuclide i in rem/Ci (Table 15.5-69)~~

~~D_{Th} = thyroid dose via inhalation in rem~~

~~Offsite whole-body gamma doses are calculated using the equation:~~

$$\cancel{D_{\gamma} = 0.25 \sum_{i} \left[\bar{E}_{\gamma i} \sum_{j} \left((IAR)_{ij} (\chi / Q)_{j} \right) \right]} \qquad \text{~~(15.5-18)~~}$$

~~where:~~

~~$(IAR)_{ij}$ = integrated activity of noble gas nuclide i released during time interval j in Ci~~

~~$(\chi / Q)_{j}$ = atmospheric dispersion factor during time interval j in seconds/m^3~~

~~$\bar{E}_{\gamma i}$ = average gamma energy for noble gas nuclide i in MeV/dis (Table 15.5-70)~~

~~D_{γ} = whole body gamma dose due to immersion in rem~~

~~(5) *Offsite Dose Results*~~

~~Thyroid and whole-body gamma doses at the Exclusion Area Boundary and the outer boundary of the Low Population Zone are presented in Table 15.5-71. All of these RSG doses are within the allowable guidelines as specified by the SRP, Revision 2 (Section 15.6.3).~~

~~The SGTR dose analysis of record is based on the RSGs and all doses are within 10 CFR 100.11 limits. The limiting dose for the SGTR analysis accepted by the NRC based on the OSGs is the EAB zero to two hour thyroid dose of 30.5 rem for the accident initiated iodine spike analysis case. This dose exceeds the SRP 15.6.3 allowable guideline value of 30 rem by 0.5 rem. However, the NRC found the 30.5 rem value acceptable in a letter to PG&E, dated February 20,2003, "Issuance of Amendment: RE: Revision to Technical Specification 1.1, 'Definitions, Dose Equivalent 1-131,' and Revised Steam Generator Tube Rupture and Main Steam Line Break Analyses."~~

~~15.5.20.2.2 Control Room Exposures~~

~~Additional analyses were performed to determine the airborne doses to the control room operators from an SGTR. These calculations used the atmospheric releases of radioactivity determined in the analysis discussed in Section 15.5.20.2.1 and Reference 46. The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake contained in Table 15.5-68 are used to determine the activity available at the control~~

~~room intake. The inflow (filtered and unfiltered) to the control room and the control room filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analysis are presented in Table 15.5-72. The control room occupancy factors assumed were taken from Table 15.5-32.~~

~~Thyroid, whole body gamma, and beta skin doses are calculated for 30 days in the control room. Although all releases are terminated when the RHR system is put in service, the calculation is continued to account for additional doses due to continued occupancy.~~

~~The total primary to secondary leak rate is assumed to be 1.0 gpm. The leakage to the intact steam generators is assumed to persist for the duration of the accident.~~

~~The calculations determine the thyroid doses based on a pre-accident iodine spike and based on an accident initiated iodine spike with a spiking factor of 335. Both spike assumptions consider 0.1 μCi/gm D.E. I--131 secondary activity. The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases.~~

~~Control room thyroid doses are calculated using the following equation:~~

$$\cancel{D_{Th} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j \right) \right]} \qquad \cancel{(15.5\text{-}19)}$$

~~where:~~

~~D_{Th} = thyroid dose via inhalation (Rem)~~
~~DCF_i = thyroid dose conversion factor via inhalation for isotope i (Rem/Ci) (Table 15.5-69)~~
~~$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)~~
~~$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 15.5-68)~~

~~Control room whole body doses are calculated using the following equation:~~

$$D_{WB} = 0.25 * \left(\frac{1}{GF}\right) * \sum_i E_{\gamma i}\left(\sum_j Conc_{ij}\right) \qquad (15.5\text{-}20)$$

~~where:~~

~~D_{WB} = whole body dose via cloud immersion (Rem)~~
~~GF = geometry factor, calculated based on Reference 17, using the equation~~
~~$GF = \frac{1173}{V^{0.338}}$ where V is the control room volume in ft^3~~
~~$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis) (Table 15.5-70)~~
~~$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)~~

~~Control room skin doses are calculated using the following equation:~~

$$D_{\beta} = 0.23 * \sum_i E_{\beta i}\left(\sum_j Conc_{ij}\right) \qquad (15.5\text{-}21)$$

~~where~~

~~D_{β} = whole body dose via cloud immersion (Rem)~~
~~$E_{\beta i}$ = average beta disintegration energy for isotope i (Mev/dis) (Table 15.5-70)~~
~~$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)~~

~~Table 15.5-74 presents the resulting airborne doses to the control room operators. The resultant doses are well below the guidelines of GDC 19, 1971, and are below the corresponding post-LOCA control room exposures presented in Table 15.5-33.~~
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 SGTR-specific assumptions associated with control room response and activity transport.

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

i. An SIS will be generated at t = 219 sec following a SGTR.

ii. 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=257.2

secs (i.e., 219 + 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.

iii. In accordance with DCPP licensing basis, the CRVS normal operation dampers of the non-accident unit are not affected by the LOOP and are isolated at t=231 secs (i.e., 219 + 2 secs signal processing time + 10 sec damper closure time).

Control Room Atmospheric Dispersion Factors

As noted in Section 2.3.5.2.2, because of the proximity of the MSSVs/10% ADVs to the control room normal intake of the affected unit, and because the releases from the MSSVs/10% ADVs have a vertically upward discharge, it is expected that the concentrations near the normal operation control room intake of the affected unit (closest to the release point) will be insignificant. Therefore, prior to switchover to CRVS Mode 4 pressurization, only the unaffected unit's control room normal intake is assumed to be contaminated by releases from the MSSVs/10% ADVs.

The bounding atmospheric dispersion factors applicable to the radioactivity release points / control room receptors applicable to a SGTR at either unit are provided in Table 15.5-64B. The χ/Q values presented in Table 15.5-64B take into consideration the various release points-receptors applicable to the SGTR to identify the bounding χ/Q values applicable to a SGTR at either unit, and reflect the allowable adjustments / reductions in the values as discussed in Chapter 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 SGTR at either unit is presented in Table 15.5-64.

15.5.20.3 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 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-64.

(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 0.25 Sv (25 rem) TEDE for a pre-existing accident iodine spike case and 10% of the 10 CFR 50.67 limit for the accident initiated iodine spike case as shown in Table 15.5-64.

(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 15.5-64.

~~(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 in excess of dose guidelines of SRP, Section 15.6.3, Revision 2 (i.e., the 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case) as shown in Table 15.5-71.~~

~~(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 the dose guidelines of SRP, Section 15.6.3, Revision 2 (i.e., 10 CFR 100.11 dose limits for the whole body and the thyroid for the pre-existing iodine spike case and 10 percent of the 10 CFR 100.11 dose limits for the whole body and the thyroid for the accident-initiated iodine spike case) as shown in Table 15.5-71.~~

~~(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 for both the pre-existing and the accident-initiated iodine spike cases as shown in Table 15.5-74.~~

~~As noted in Section 15.5.20.2, the above dose estimates reflect the RSGs and are within 10 CFR 100.11 limits. The SGTR analysis accepted by the NRC based on OSGs is the EAB zero to two hour thyroid dose of 30.5 rem for the accident-initiated iodine spike analysis case. This dose exceeds the SRP 15.6.3 allowable guideline value of 30 rem by 0.5 rem. However, the NRC found the 30.5 rem value acceptable in a letter to PG&E, dated February 20,2003, "Issuance of Amendment: RE: Revision to Technical Specification 1.1, 'Definitions, Dose Equivalent 1-131,' and Revised Steam Generator Tube Rupture and Main Steam Line Break Analyses."~~

15.5.21 RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

15.5.21.1 Acceptance Criteria

The radiological consequences of a LRA 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 0.025 Sv (2.5 rem) TEDE.

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose in excess of 0.025 Sv (2.5 rem) TEDE.

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.

~~The radiological consequences of a locked rotor accident shall not exceed the dose limits of 10 CFR 100.11 as outlined below:~~

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

~~An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), shall not receive a total radiation dose to the whole body in excess of 25 rem, or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.~~

~~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.~~

15.5.21.2~~-~~ Identification of Causes and Accident Description

15.5.21.2.1 Activity Release Pathways

This event is caused by an instantaneous seizure of a primary reactor coolant pump (RCP) rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage is predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed SG tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system via the 10% ADVs and MSSVs. Following reactor trip, and based on an assumption of a LOOP coincident with reactor

trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs and 10% ADVs until initiation of shutdown cooling. DCPP has established that the LOL event generates the maximum primary to secondary heat transfer and the LRA assumes these same conservatively bounding secondary steam releases.

~~Under adverse circumstances, a locked rotor accident could cause small amounts of fuel cladding failure in the core. If this occurs, some fission products will enter the coolant and will mostly remain in the coolant until cleaned up by the primary coolant demineralizers, or in the case of noble gases, until stripped from the coolant. Following such an incident, there are several possible modes of release of some of this activity to the environment.~~

~~In the short-term, if the accident occurs at a time when significant primary-to-secondary leakage exists, some of the additional activity entering the coolant will leak into the secondary system. The noble gases will be discharged to the atmosphere via the air ejectors or by way of atmospheric steam dump. The iodines will remain mostly in the liquid form and be picked up by the blowdown treatment system. Some fraction of the iodines, however, will be released via the air ejectors or by way of atmospheric steam dump. In addition, if an atmospheric steam dump is necessary, some of the activity contained in the secondary system prior to the accident will be released.~~

~~The amounts of steam released depend on the time relief valves remain open and the availability of condenser bypass cooling capacity. The amounts of radioactive iodine released depend on the amounts of steam released, the amount of activity contained in the secondary system prior to the accident, and the amount contained in the primary coolant which leaks into the secondary system. As discussed in Section 15.5.10, the amount of steam released following the locked rotor accident, if no condenser cooling is available, would not exceed approximately 1.7E+06 lbm.. In the analysis of both the design basis case and the expected case, this amount of steam was assumed to be released.~~

Regulatory requirements provided for the LRA in pertinent sections of Regulatory Guide 1.183 including Appendix G is used to develop the dose consequence model.

~~For the design basis case, it was assumed that the plant had been operating continuously with 1 percent fuel cladding defects and 1 gpm primary-to-secondary leakage. For the expected case calculation, operation at 0.2 percent defects and 20 gallons per day to the secondary was assumed. In both cases, leakage of water from primary to secondary was assumed to continue during cooldown at 75 percent of the pre-accident rate during the first 2 hours and at 50 percent of the pre-accident rate during the next 6 hours. These values were derived from primary-to-secondary pressure differentials during cooldown. It was also conservatively assumed for both cases that the iodine Partition Factor in the steam generators releasing steam was 0.01 on a mass basis (Reference 15). In addition, to account for the effect of iodine spiking, fuel escape rate coefficients for iodines of 30 times the normal operation values given in~~

~~Table 11.1-9 were used for a period of 8 hours following the start of the accident. Other detailed and less significant modeling assumptions are presented in Reference 4.~~

~~The assumptions used for meteorology, breathing rates, population density and other common factors were also described earlier. Both the primary and secondary coolant activities prior to the accident are discussed in Section 15.5.2.~~

~~In order to determine the primary coolant activities immediately after the accident, it was assumed that less than 10 percent of the total activity contained in the fuel rod gaps would be immediately released to the coolant and mixed uniformly in the coolant system volume. The gap inventories used 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 15.5-41. The potential doses are given in Table 15.5-42. The exposures are also shown in Figures 15.5-14 and 15.5-15 as a function of the amount of fuel failure that occurs. On the left boundary of these graphs, in the region of negligible fuel failures, the exposures are just the component resulting from the activity already present in the secondary system, or which leaks through the steam generators at pre-accident primary coolant levels. These exposures correspond to those shown in Figures 15.5-2 through 15.5-5.~~

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

~~Another mode of release following a locked rotor accident, or any accident involving significant fuel failure, is the long-term release by way of cleanup and leakage from the primary coolant system. The activity going through these pathways, principally Kr-85, would result in some incremental long-term dose beyond the normal yearly releases. This pathway of release has been evaluated, and the results are presented in Figure 15.5-16. Since the activity released in this way would reach the environment over a long term, the annual average atmospheric dilution factors (Table 15.5-5) and breathing rates have been used. The amounts of activity released were determined by multiplying the activities released from the gaps following the accident by the release fractions listed in Table 15.5-40.~~

~~These long-term release fractions were determined from the normal radioactivity transport analysis carried out for Chapter 11, for the anticipated operational occurrences case. In essence, these fractions are the fractions of a curie reaching the environment per curie released to the coolant, for each isotope. The pathways included are primary cleanup, leakage to the containment, and leakage to the auxiliary building. As shown in Table 15.5-40, essentially all of the Kr-85 released to the coolant is eventually released to the environment, as would be physically expected, and lower fractions of the other isotopes are released, depending on their respective overall cleanup, leakage, and decay factors in the plant. It can be concluded by comparing these exposures to the short-term exposures in Figure 15.5-12 that the incremental long-term exposures are negligible additions to the radiological consequences of accidents of this kind.~~

~~In addition, it can be concluded that accidents of this kind would not result in significant additions to the annual doses expected from normal plant operation.~~

~~From these short-term and long-term analyses, it can also be concluded that all potential exposures from a locked rotor 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 was made in Section 15.5.17, for conditions following a LBLOCA. The containment shine contribution to control room dose would not be applicable following a locked rotor accident.~~

The LRA is postulated to result in 10% fuel failure resulting in the release of the associated gap activity. As discussed in Section 15.5.3.1.3, the core gap activity is assumed to be comprised of 12% of the core I-131 inventory, 30% of the core Kr-85 activity, 10% of the remaining noble gas and halogen isotopes, and 17% of the core alkali metals (Cesium and Rubidium). Table 15.5-42A lists the key assumptions / parameters utilized to develop the radiological consequences following a LRA.

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

15.5.21.2.2 Activity Release Transport Model

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

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

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

particulates are assumed to be carried over in accordance with the design basis SG moisture carryover fraction.

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

15.5.21.2.3 Offsite Dose Assessment

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

The bounding EAB and LPZ dose following a LRA at either unit is presented in Table 15.5-42.

15.5.21.2.4 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 LRA-specific assumptions associated with control room response and activity transport.

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

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

<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, 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 an LRA at either unit are provided in Table

15.5-42B. The χ/Q values presented in Table 15.5-42B take into consideration the various release points-receptors applicable to the LRA to identify the bounding χ/Q values applicable to a LRA at either unit, and reflect the allowable adjustments / reductions in the values as discussed in 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 LRA at either unit is presented in Table 15.5-42.

15.5.21.3 Conclusions

~~By comparing the activity releases following a locked rotor accident, given in Table 15.5-41, with the activity releases calculated for a LBLOCA, given in Tables 15.5-13 and 15.5-14, it can be concluded that any control room exposures following a locked rotor accident will be well below the GDC 19, 1971, criterion level.~~

~~Additionally, the analysis demonstrates that the acceptance criteria are met as follows:~~
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 0.025 Sv (2.5 rem) TEDE as shown in Table 15.5-42.

(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 0.025 Sv (2.5 rem) TEDE as shown in Table 15.5-42.

(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 15.5-42.

~~(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 15.5-42.~~

~~(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 15.5-42.~~

~~(3) Since the activity releases from the locked rotor accident given in Table 15.5-41 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 within the established criteria of GDC 19, 1971 and discussed in Section 15.5.17.~~

15.5.22 RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

The procedures used in handling fuel in the containment and fuel handling area are described in detail in Section 15.4.5. In addition, design and procedural measures provided to prevent fuel handling accidents are also described in that section, along with a discussion of past experience in fuel handling operations. The basic events that could be involved in a fuel handling accident are discussed in that section, and the following discussion evaluates the potential radiological consequences of such an accident.

The assumption of a LOOP related to a postulated design basis accident which leads to a reactor trip does not directly correlate to an FHA. Specifically, a FHA does not directly cause a reactor trip and a subsequent LOOP due to grid instability; nor can a LOOP be the initiator of a FHA. Thus the FHA dose consequence analyses are evaluated without the assumption of a LOOP.

15.5.22.1 ~~Fuel Handling Accident In The Fuel Handling Area~~

~~15.5.22.1.1~~ Acceptance Criteria

The radiological consequences of a FHA in the Fuel Handling Building (FHB) or in the Containment shall not exceed the dose limits of 10 CFR 50.67, as modified by Regulatory Guide 1.183, July 2000 and outlined below:

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

(1) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release shall not receive a radiation dose in excess of 0.063 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.

<u>Control Room Dose Criteria (</u>10 CFR 50.67)

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.

~~The radiological consequences of a fuel handling accident in the fuel handling area shall not exceed the dose limits of 10 CFR 50.67 as outlined below:~~

~~(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).~~

~~(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).~~

~~(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.~~

15.5.22.~~1~~.2 Identification of Causes and Accident Description

15.5.22.2.1 Activity Release Pathways

This event postulates that a spent fuel assembly is dropped during refueling in the Spent Fuel Pool (SFP) located in the FHB, or in the reactor cavity located in the Containment. All of the fuel rods (264 rods) in the dropped fuel assembly are assumed to be damaged; thus all of the activity in the fuel gap of the dropped assembly is assumed to be instantaneously released into the SFP or into the reactor cavity. As documented in the NRC SER for Amendments 8 and 6 to DCPP Facility Operating - License Nos. DPR-80 and DPR-82, respectively (Reference 87), the assumption that all fuel rods in one assembly rupture is conservative because the kinetic energy available for causing damage to a fuel assembly dropped through water is fixed by the drop distance. The kinetic energy associated with the maximum drop height for a fuel handling accident is not considered sufficient to rupture the equivalent number of fuel rods of one assembly in both the dropped assembly and the impacted assembly.

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.

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.

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.

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.

~~The radiological consequences of a fuel handling accident in the fuel handling area were analyzed using the LOCADOSE computer code.~~

~~The values assumed for individual fission product inventories are calculated for a source term assuming approximately 105 percent full power operation (3580 MW thermal) immediately preceding shutdown. The accident is assumed to occur 100 hours after shutdown. This latter interval represents approximately the minimum time required to prepare (cooldown, head and internals removal, cavity flooding, etc.) the core for refueling and is therefore somewhat conservative in that it would require that the accident occur during handling of the first few fuel assemblies.~~

~~The source term is conservatively assumed to be a composite of the highest fission product activity totals for various combinations of burnup and enrichment. The ORIGEN-2 computer code was used to calculate these worst-case fission product inventories. The DBA gap activity inventory is based on NRC Safety Guide 25, March 1972, assumptions: radial peaking factor of 1.65, gap fraction of 10 percent for noble gases other than Kr-85, gap fraction of 30 percent for Kr-85, and gap fraction of 10 percent for iodines.~~

~~The assumption is made for both cases that 100 percent of the activity (consisting principally of fission product isotopes of the elements xenon, krypton, and iodine) present in the gap between the fuel pellets and the cladding in the damaged rods is immediately released to the pool or cavity water. This assumption is conservative for~~

~~elemental iodine because the low cladding and gap temperatures would result in a large fraction of it being condensed and temporarily retained within the cladding.~~

~~The analysis assumes that the fission product release occurs at a water depth of 23 feet, which is the minimum water depth above the top of the fuel as required by Technical Specifications. The spent fuel pool, where handling operations are most likely to result in fuel damage, has a water depth of about 38 feet. Using a depth of 23 feet accounts for cases in which the release occurs from the top of an assembly that is resting vertically on the floor, and for releases that occur near the top of the storage racks. Finally, consistent with Safety Guide 25, March 1972 the analysis assumes that all activity that escapes from the pool to the fuel handling area air spaces is released from the area within a 2-hour time period.~~

~~Of the activity reaching the water, 100 percent of the noble gases, xenon and krypton, are assumed to be immediately released to the fuel handling area air spaces. However, the ability of the pool water to scrub iodine from the gas bubbles as they rise to the surface has been considered. The pool DFs for the inorganic and organic species are 500 and 1, respectively, giving an overall effective DF of 200 (i.e., 99.5 percent of the total released from the damaged rods is retained by the pool water). This difference in DFs for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75 percent inorganic and 25 percent organic species. These assumptions are consistent with those suggested in NRC Regulatory Guide 1.183, July 2000. Table 15.5-44 itemizes the gap activity available for release from the FHB atmosphere to the environment.~~

~~Table 15.5-45 itemizes the assumptions and numerical values used to calculate the fuel handling accident radiological exposures. The potential releases of activity to the atmosphere are listed in Table 15.5-44. The exposures resulting from the postulated fuel handling accident inside the fuel handling area are presented in Table 15.5-47. These exposures are well below the Regulatory Guide 1.183, July 2000 limits and demonstrate the adequacy of the fuel handling safety systems.~~

In the very unlikely event of a serious fuel handling accident and in combination with the conservative assumptions discussed above, containment building or fuel handling area activity concentrations may be quite high. High activity concentrations necessitate the evacuation of fuel handling areas in order to limit exposures to fuel handling personnel. Upon indication of a serious fuel handling accident, the fuel handling area will be evacuated until the extent of the fuel damage and activity levels in the area can be determined. Any serious fuel handling accident would be both visually and audibly detectable via radiation monitors in the fuel handling areas that locally alarm in the event of high activity levels and would alert personnel to evacuate.

~~Although conservatively neglected for this analysis, the~~The fuel handling area has the additional safety feature of ventilation air flow that sweeps the surface of the spent fuel pool carrying any activity away from fuel handling personnel. This sweeping of the

spent fuel pool is expected to considerably lower activity levels in the fuel handling area in the event of a serious fuel handling accident.

~~After charcoal filter cleanup (another design feature conservatively neglected in this analysis), fuel handling area post-accident ventilation air exhausts through the plant vent at a height of 70 meters. Site meteorology is such that it is very unlikely that any airborne activity will enter the control room ventilation system.~~

~~Spent fuel cask accidents in the fuel handling area causing fuel damage are precluded due to crane travel limits and design and operating features as described in Sections 9.1.4.3.9 and 9.1.4.2.6. Spent fuel handling accidents in the fuel handling area would not jeopardize the health and safety of the public.~~

Spent fuel cask accidents in the fuel handling area causing fuel damage are precluded due to crane travel limits and design and operating features as described in Sections 9.1.4.3.9 and 9.1.4.2.6. Spent fuel handling accidents in the fuel handling area would not jeopardize the health and safety of the public.

The FHA dose assessment follows the requirements provided for the FHA in pertinent sections of Regulatory Guide 1.183 including Appendix B. As discussed in Section 15.5.3.1.3, the core gap activity is assumed to be comprised of 12% of the core I-131 inventory, 30% of the core Kr-85 activity, 10% of the remaining noble gas and halogen isotopes, and 17% of the core alkali metals (Cesium and Rubidium). Table 15.5-47A lists the key assumptions / parameters utilized to develop the radiological consequences following an FHA at either location and at either unit.

DCPP procedures prohibit movement of recently irradiated fuel which is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. Table 15.5-47C provides the gap activity inventory of the noble gases, iodines and alkali metals in a single fuel assembly at 72 hrs post reactor shutdown.

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

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

15.5.22.2.2 Activity Release Transport Model

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