55. The Steam Generator Blowdown System does not communicate directly with the containment atmosphere or reactor coolant pressure boundary. This valve is required to close when the auxiliary feedwater pumps are started in order to maintain steam generator inventory.

56. Administratively controlled valve per Technical Specifications. Valve is sealed closed in accordance with SRP Section 6.2.4, Revision 3 Criteria II.6 and II.14.

must also be capable of maintaining the environmental qualification (EQ) parameters to within acceptable limits.

The DCPP LOCA containment response analysis considers a spectrum of cases that address differences between the individual DCPP Units, LOCA break locations, and postulated single failures (minimum and maximum safeguards). The limiting cases that address the containment peak pressure case and limiting long-term EQ temperature are presented in this section.

Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC version 7.2. The GOTHIC Technical Manual (Reference 13) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualifications Report (Reference 14) provides a comparison of the solver results with both analytical solutions and experimental data.

The GOTHIC containment modeling for DCPP is consistent with the NRC approved Kewaunee evaluation model (Reference 15). Kewaunee and DCPP both have large dry containment designs with similar active heat removal capabilities. The latest code version is used to take advantage of the diffusion layer model heat transfer option. This heat transfer option was approved by the NRC (Reference 15) for use in Kewaunee containment analyses with the condition that the effect of mist be excluded from what was earlier termed as the mist diffusion layer model. The GOTHIC containment modeling for DCPP has followed the conditions of acceptance placed on Kewaunee. The differences in GOTHIC code versions are documented in Appendix A of the GOTHIC User Manual Release Notes (Reference 16). Version 7.2 is used consistently with the restrictions identified in Reference 15; none of the user-controlled enhancements added to version 7.2 were implemented in the DCPP containment model. A description of the DCPP GOTHIC model is provided later in this section.

6.2D.3.2.3 Input Parameters and Assumptions

The major modeling input parameters and assumptions used in the DCPP LOCA containment evaluation model are identified in this section. The assumed initial conditions and input assumptions associated with the fan coolers and containment sprays are listed in Table 6.2D-17. The containment spray flow data used in the analysis are presented in Table 6.2D-18. The function of the residual heat removal system (RHR) during a LOCA is to remove heat from the core by way of the ECCS. The ECCS recirculation and CCW system parameters are outlined in Table 6.2D-17. The containment structural heat sink input is provided in Table 6.2D-19, and the corresponding material properties are listed in Table 6.2D-20.

As stated in Section 6.2.2, and further discussed in Section 15.5, to address the delayed radioactivity release associated with the core damage sequence of Alternate Source Terms (AST), credit is taken for fission product removal due to CSS operation in the injection and recirculation mode following a LOCA. Utilization of CSS operation in

the recirculation mode will reduce containment pressure and temperature in the long term due to enhanced heat removal by sprays. However, for conservatism the effect of CSS during the recirculation mode is not considered in this analysis.

The LOCA containment analysis described here uses revised input and assumptions in support of the current design, while addressing analytical conservatisms. The assumptions used in the M&E release input model and the containment pressure input model are discussed in WCAP-10325-P-A (Reference 1) and WCAP-8264-P-A Revision 1 (Reference 5). Significant assumptions contained in the LOCA containment integrity analysis include:

(1) For all the long term cases and the base hot leg cases, there is a loss of offsite power coincident with the LOCA. For the hot leg break case with safety injection, off site power is available to allow the safety injection to begin during the blowdown.

(2) In all cases, two containment fan cooler units (CFCUs), each from separate trains, are assumed to be unavailable due to maintenance

(3) The long term decay heat steaming M&E release calculation assumes that:

(a) All the decay heat is released to the containment as steam to maximize the pressure and temperature of the containment vapor region,

(b) 102% reactor power and ANS 1979 +2 sigma decay heat,

(4) The bounding auxiliary saltwater (ASW) temperature is 64 °F

(a) It should be noted that a separate set of analyses assuming a 70 °F ocean water temperature addresses operation with an elevated ultimate heat sink temperature (refer to Section 9.2.2.3.13).

The following are notable features of the current containment integrity analysis.

Decay heat steaming M&E release rates, after the end of the sensible heat release from the RCS and steam generators, are calculated each time step by GOTHIC using the transient containment pressure and recirculation safety injection water temperature.

Non-condensable accumulator gas release is modeled in the GOTHIC model (refer to Section 6.2D.3.2.5); no accumulator nitrogen gas addition due to refill is considered in the analysis.

A recirculation system model that couples the RHR, CCW, CFCUs and auxiliary saltwater systems was developed. Detailed accounting of CCW flow rates through the containment heat removal systems was used for the CFCUs, RHR heat exchangers, and miscellaneous CCW heat loads.

termination assumed in the LOCA containment analysis are presented in Table 6.2D-17.

Accumulator Nitrogen Gas Modeling

The accumulator nitrogen gas release is modeled with a flow boundary condition in the LOCA containment model. The nitrogen release rate was conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas was released to the containment at a rate of 327.4 lbm/s. The release begins at 51.9 seconds, the minimum accumulator tank water depletion time.

6.2D.3.2.6 LOCA Containment Integrity Analysis Results

The containment pressure, steam temperature, and water (sump) temperature profiles of the DEHL peak pressure case are shown in Figures 6.2D-3 through 6.2D-5. Table 6.2D-13 provides the transient sequence of events for the DEHL transient.

The containment pressure, steam temperature, and water (sump) temperature profiles of the DEPS long-term EQ temperature transient are shown in Figures 6.2D-6 through 6.2D-8(The peak DEPS values are from Unit 2). The sequence of events for the Unit 1 and Unit 2 DEPS transients are presented in Tables 6.2D-14 and 6.2D-15, respectively. The peak pressure (Figure 6.2D-6) for the DEPS case occurs at 24.1 seconds after the end of the blowdown. The fans begin to cool the containment at 48.7 seconds. Containment sprays begin injecting at 88.0 seconds. The pressure comes down as the steam generators reach equilibrium with the containment environment, but spikes up again at recirculation when the CCW temperature increases and the CCW flow rate to the CFCUs decreases. The sensible heat release from the steam generator secondary system and RCS metal is completed at 3600 seconds, but at 3798 seconds, the RWST reaches a low level alarm and spray flow is terminated. The containment pressure increases for a time and then begins to decrease over the long term as the RHR heat exchangers and CFCUs remove the heat from the containment.

Table 6.2D-21 summarizes the containment peak pressure and temperature results and pressure and temperature at 24 hours for EQ support and the acceptance limits for these parameters.

A review of the results presented in Table 6.2D-21 shows that the analysis margin (analysis margin is the difference between the calculated peak pressure and temperature and the acceptance limits) is maintained for DCPP. From the GOTHIC analysis the containment peak pressure is 41.4 psig. At 24 hours, the maximum containment pressure is 8.9 psig and the maximum temperature is 167.5°F.

An assessment was performed to evaluate the effect of operation of containment spray in the recirculation mode following a LOCA on the sump water temperature transient. As shown in Figure 6.2D-8A, the sump water temperature transient is effected only in

the long-term (i.e., after initiation of recirculation spray) at which time it becomes more limiting due to increased heat addition from the containment atmosphere to the sump fluid by the containment recirculation sprays which is credited to operate for approximately 6.25 hours. The effect of this increase in temperature on the performance of the CCW system is discussed in Section 9.2.2.3.13.

6.2D.3.2.7 Conclusion

The DCPP containment can adequately account for the M&E releases that would result from a LOCA. The DCPP containment systems will provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained.

- The peak calculated pressure is less than the containment design pressure of 47 psig
- The peak calculated containment average air temperature is less than the containment design temperature of 271 °F
- The calculated pressure at 24 hours is less than 50 percent of the peak calculated value

6.2D.4 LONG-TERM MAIN STEAMLINE BREAK INSIDE CONTAINMENT

6.2D.4.1 MSLB Mass and Energy Release Analysis

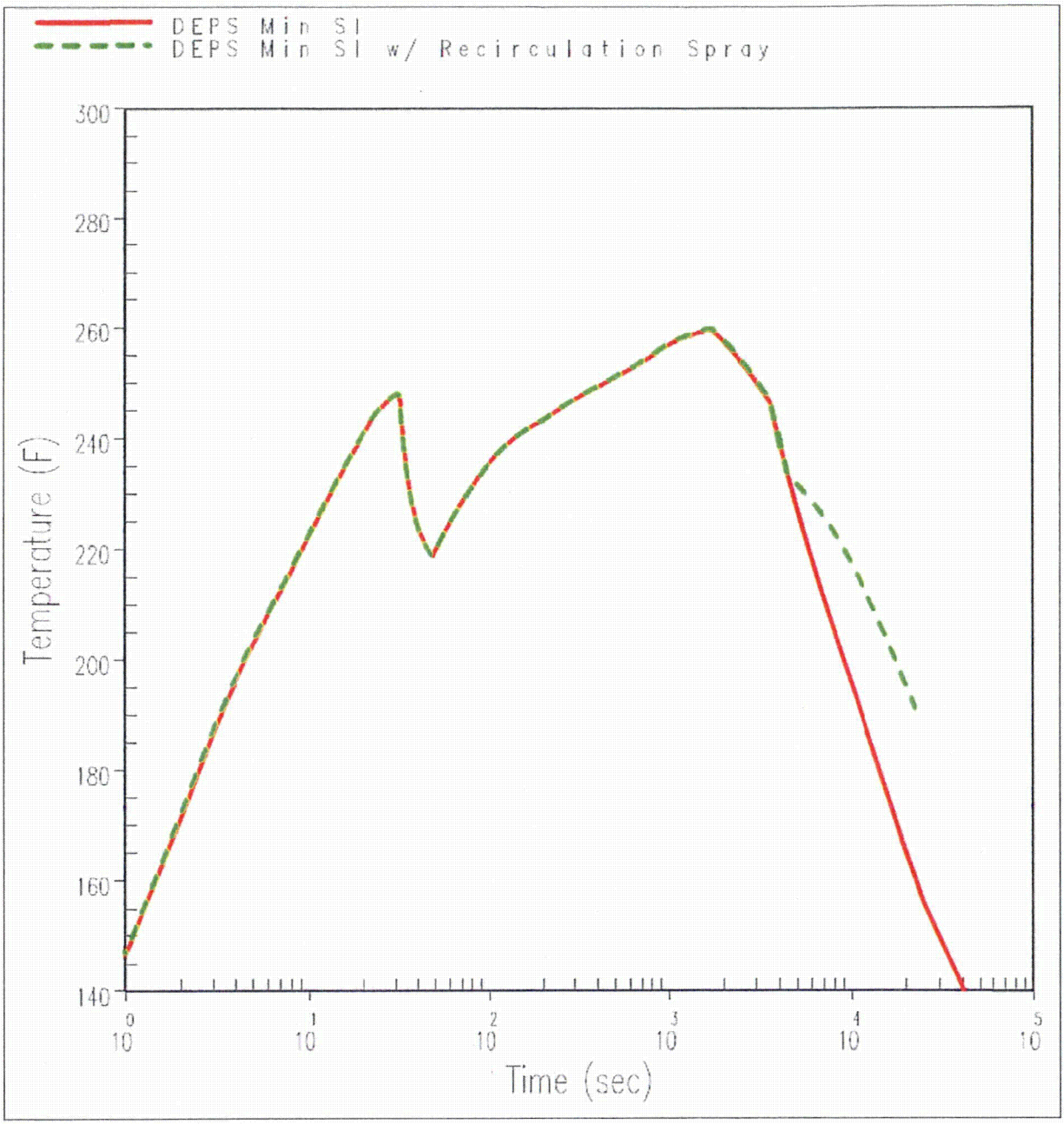
6.2D.4.1.1 Acceptance Criteria

There are no direct acceptance criteria for MSLB M&E releases. The analysis methods follow the guidelines provided by the USNRC with respect to the sources of M&E during the various phases of a MSLB transient (refer to Section 6.2D.4.1.4).

The specific acceptance criteria for the containment response to a MSLB are discussed in Section 6.2D.4.2.1.

6.2D.4.1.2 Introduction and Background

The MSLB is classified as an American Nuclear Society (ANS) Condition IV event, an infrequent fault. A MSLB occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment that could produce high pressure conditions for extended periods of time. The magnitude of the releases following a MSLB is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. There are competing effects and credible single failures in the postulated accident scenario used to determine the worst cases for containment pressure and the associated containment temperature following a MSLB.



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UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 6.2D-8A DEPS Minimum SI Sump Temperature Comparison

restoring power to the RWST supply valves to the RHR and the SI pumps). The same sequence (as delineated in Table 6.3-5) is followed regardless of which power supply is available (Offsite Power System or Standby Power Supply). Controls for ECCS components are grouped together on the main control board. The component position lights verify when the function of a given switch has been completed. The total required switchover time for the changeover from injection to recirculation is approximately 10 minutes, as shown in Table 6.3-5. The postulated single failure during the changeover sequence is the failure of an RHR pump to trip on low RWST level. The operator action requires approximately 5 minutes to locally open the breaker for an RHR pump motor. The operator action is performed concurrently with the changeover sequence and there is no increase in the total time for the changeover. The changeover sequence can be completed, with the single failure, and the remaining useable RWST volume exceeds the licensing basis of 32,500 gallons.

6.3.3.6.1.1.3 Recirculation Mode After Loss of Primary Coolant

After the injection operation, water collected in the containment recirculation sump is cooled and returned to the RCS by the low-head/high-head recirculation flowpath. The RCS can be supplied simultaneously from the RHR pumps and from a portion of the discharge from the RHR heat exchanger that is directed to CCP1 and CCP2 and SI pumps that return the water to the RCS. The latter mode of operation ensures flow in the event of a small rupture where the depressurization proceeds more slowly, so that the RCS pressure is still in excess of the shutoff head of the RHR pumps at the onset of recirculation. Approximately 7.0 hours after LOCA inception, the operators will manually initiate hot leg recirculation and complete the switchover process within 15 minutes. Hot leg recirculation is implemented to ensure termination of boiling and prevent boric acid crystallization. Some cold leg recirculation would be maintained with the CCPs (CCP1 and CCP2) after hot leg recirculation is initiated.

As discussed in Section 6.2.2, the CSS is operated during the injection and recirculation mode (up to 6.25 hours) to support fission product removal following a LOCA. During the recirculation mode, the CS pumps are isolated (refer to Table 6.3-5) and water, recirculated from the containment sump, is supplied by the RHR pumps for recirculation spray to the spray headers. Single failures that result in the loss of one RHR train are taken into consideration in system alignment to ensure appropriate division of the recirculation flow between containment spray and core injection to ensure fission product removal from the containment atmosphere and adequate core cooling.

The RWST is protected from back flow of reactor coolant from the RCS. All connections to the RWST except those that are designed to return flow to the RWST are provided with check valves to prevent back flow.

Redundancy in the external recirculation loop is provided by duplicate CCPs (CCP1 and CCP2), SI pumps, and RHR pumps and heat exchangers. Inside the containment, the high-pressure injection system is divided into two separate flow trains. For cold leg recirculation, CCP1 and CCP2 deliver to all four cold legs and the SI and RHR pumps

radiation protection to permit access without radiation exposures to personnel in excess of 10 CFR Part 20 limits under normal conditions.

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

Instrumentation and controls related to control room habitability are provided to monitor and maintain applicable variables within prescribed operating ranges.

6.4.1.1.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

Radiation monitoring instrumentation is provided to monitor radioactive releases entering each control room normal air intake and pressurization system intake. Area radiation monitoring is provided in the control room.

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

The control room is designed 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 a design basis accident.

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

The control room habitability systems are designed to provide backup to the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems.

6.4.1.1.9 10 CFR Part 50, Appendix R (Sections III.G, III.J, and III.L) - Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

Section III.G – Fire Protection of Safe Shutdown Capability: Fire protection of the control room habitability systems is provided by a combination of physical separation, fire-rated barriers, and automatic suppression (except in the control room) and detection.

Section III.J – Emergency Lighting: Emergency lighting or Battery Operated Lights (BOLs) are provided in the control room and associated areas required to safely shut down a unit in the event of a fire.

Section III.L – Alternative and Dedicated Shutdown Capability: Safe shutdown capabilities are provided in the control room and at an alternate location via the hot shutdown panel or locally at the 480-V switchgear, for equipment powered by the 480-V system required for the safe shutdown of the plant, in the event of a fire.

The control room habitability systems are designed to the fire protection guidelines of Branch Technical Position APCSB 9.5-1 (refer to Appendix 9.5B, Table B-1). The adequacy of the control room fire protection system is evaluated in Sections 7.7.1 and 9.5.1.

6.4.1.3.3 General Design Criterion 4, 1967 - Sharing of Systems

The control room is common to DCPP Unit 1 and Unit 2 and therefore requires sharing of SSCs between units. The CRVS is shared between Unit 1 and Unit 2. In addition, CRPS pressurization is shared by the control room and the TSC. The sharing of these systems is addressed in Section 9.4.1.3.3.

6.4.1.3.4 General Design Criterion 11, 1967 - Control Room

Control room habitability is provided by shielding, the CRVS, and the fire protection system. The adequacy of control room shielding is evaluated for normal operating conditions in Chapter 11 and Section 12.1. The adequacy of the CRVS is evaluated for normal operating conditions in Chapter 11 and Sections 9.4.1 and 12.1; for hazardous chemical emergencies in Section 9.4.1; and for fire emergencies in Sections 9.4.1 and 9.5.1.

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

Controls for control room habitability components are provided for system operation. Instrumentation is provided for monitoring habitability system parameters during normal operations and accident conditions (refer to Sections 6.4.1.3.6 and 6.4.1.5).

6.4.1.3.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

Main control room normal air intake radiation monitors and CRPS intake radiation monitors are provided to detect radioactivity in the air flow into the control room ventilation and pressurization systems. A control room area radiation monitor is provided to monitor radiation in the control room environs.

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

The control room habitability systems permit access and occupancy for operating the plant without personnel receiving radiation exposures in excess of GDC 19, ~~1971~~1999, limits for the duration of a design basis accident. The adequacy of control room shielding is evaluated for post-accident conditions in Section 15.5. The adequacy of the CRVS is evaluated for radiological emergencies in Section 15.5. ~~Note that for the postulated fuel handling accident in the fuel handling building, an alternate source term is assumed per 10 CFR 50.67 (refer to Section 15.5.22).~~

During normal operation TSC personnel are protected from radiation sources such that ~~doses~~ concentration limits are maintained below ~~limits~~ those prescribed in 10 CFR Part 20.

6.4.2.1.3 10 CFR 50.47 - Emergency Plans

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

6.4.2.1.4 NUREG-0737 (Items II.B.2 and III.A.1.2), November 1980 - Clarification of TMI Action Plan Requirements

Item II.B.2 - Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operations: Adequate access to the TSC is provided by increased permanent or temporary shielding.

Item III.A.1.2 - Upgrade Emergency Support Facilities: NUREG-0737, Supplement 1, January 1983 provides the requirements for III.A.1.2 as follows:

Section 8.2.1(e) - The TSC is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment.

Section 8.2.1(f) - The TSC is provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem TEDE for the duration of the accident ~~5 rem whole body, or its equivalent to any part of the body,~~ for the duration of the accident.

6.4.2.2 System Description

The TSC is designated to be habitable throughout the course of a design basis accident. The outside walls, with steel bulkhead doors, form an airtight perimeter boundary. The TSC structure is designed to PG&E Design Class III. For seismic qualification, refer to the DCPP Q-List (Reference 8 of Section 3.2).

During normal operation the outside air intake flow into the TSC is filtered through a HEPA filter. The TSC has the manual capability to isolate the area from the outside and to recirculate air by the air conditioning system (refer to Section 9.4.11). The hazardous chemical release warning will have to be received from the control room to enable those in the TSC to manually isolate the area from the outside.

The TSC is provided with its own PG&E Design Class II heating, ventilation and air conditioning (HVAC) system. It is not seismically qualified and is fed from a non-Class 1E power source, although the air cleanup portion of the system has the capability to be supplied power from a Class 1E bus. The PG&E Design Class I CRPS system provides a redundant supply of pressurization air to the TSC ventilation system. The CRPS connecting ductwork is designed to PG&E Design Class 1 and the TSC

6.4.2.3.4 NUREG-0737 (Items II.B.2 and III.A.1.2), November 1980 - Clarification of TMI Action Plan Requirements

Item II.B.2, Design Review Of Plant Shielding And Environmental Qualification Of Equipment For Space/Systems Which May Be Used In Postaccident Operations (originally Recommendation 2.1.6.b of NUREG-0578 [Reference 1]) - The TSC is designed to meet the criteria for shielding provided in NUREG-0737, Item II.B.2. Adequate shielding is provided to permit access to vital areas, including the control room and TSC. ~~Utilizing the guidelines of GDC 19, 1971, and the occupancy factors contained in Standard Review Plan 6.4, the TSC shielding design radiation dose rate limits were established at 10 mrem/hr for direct radiation and 5 mrem/hr for airborne particulate and gaseous releases (internal to TSC). The total dose rate to any individual in the TSC is thus limited to 15 mrem/hr, from a time period beginning 1 hour after start of the design basis accident to 30 days later. The adequacy of shielding for the TSC has been evaluated for normal and post-accident conditions as described in Section 12.1.~~ The acceptance criteria for the TSC dose is based on 10 CFR 50.67, in which dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident. The adequacy of TSC shielding is evaluated for post-LOCA conditions in 12.1.2.7 and Section 15.5.

Item III.A.1.2, Upgrade Emergency Support Facilities - NUREG-0737, Supplement 1, January 1983 provides the requirements for III.A.1.2.

The TSC is designed to meet the criteria for habitability provided in NUREG-0737, Supplement 1, items 8.2.1(e) and 8.2.1(f) (Reference 4). The guidance of NUREG-0696, 1981 (Reference 2), cited by NUREG-0737, Supplement 1, is followed regarding ventilation, filtration, radiation monitoring, and radiation protection. The TSC HVAC system is designed to PG&E Design Class II (Section 9.4.11.3).

Section 8.2.1(e) - The adequacy of the TSC ventilation system has been evaluated for normal and post-accident operating conditions as described in Sections 9.4.11 and 12.2 and for fire emergencies as described in Sections 9.4.1 and 9.5.1.

The adequacy of TSC fire protection features is evaluated in Section 9.5.1.

Section 8.2.1(f) – The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, as amended by Regulatory Guide 1.183, July 2000, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident. TSC shielding and the ventilation system prevent post-accident doses inside the TSC from exceeding 5 rem ~~whole body, or its equivalent to any part of the body,~~TEDE for the duration of the accident. Refer to Section 6.4.2.2 for a description of the TSC ventilation system. The adequacy of TSC shielding and ventilation is evaluated for post-LOCA conditions in Section 15.5.

five functional modes of operation. A ten-path rotary transfer assembly is a transfer device that is used to route a detector into any one of up to ten selectable paths. A common path is provided to permit cross-calibration of the detectors.

The control room contains the necessary equipment for control, position indication, and flux recording for each detector. Panels are provided to indicate the position of the detectors and to plot the flux level. Additional panels are provided for such features as drive motor controls, core path selector switches, plotting, and gain controls.

A flux mapping consists, briefly, of selecting (by panel switches) flux thimbles in given fuel assemblies at various core locations. The detectors are driven to the top of the core and stopped automatically. An x-y plot (position versus flux level) is initiated with the slow withdrawal of the detectors through the core from the top to a point below the bottom. Other core locations are selected and plotted in a similar manner. Each detector provides axial flux distribution data along the center of a fuel assembly. Various radial positions of detectors may then be compared to obtain a flux map for a region of the core.

Operating plant experience has demonstrated the adequacy of the incore instrumentation system in meeting the design bases stated.

7.7.2.10 Control Locations

7.7.2.10.1 Control Room

A common control room for Unit 1 and Unit 2 contains the controls and instrumentation necessary for operating each unit's reactor and turbine-generator during normal and accident conditions. The control boards for Unit 2 are physically separated from the Unit 1 control boards. The control room is continuously occupied by licensed operating personnel during all operating conditions. It is also expected to be continuously occupied during all accident conditions. In the remote case where it is not possible to occupy the control room, alternative control locations are provided. The control room for each unit is designed to normally accommodate three to five people.

Sufficient shielding, distance, and containment integrity are provided to ensure that control room personnel are not subjected to doses under postulated accident conditions that would exceed ~~2.5 rem to the whole body or 30 rem to the thyroid~~5 rem TEDE.~~, including~~ The dose~~s~~ received by the operator during both entry and exit is expected to be minimal. (Refer to Section 15.5 for detail) Control room ventilation is provided by a system capable of having a large percentage of recirculated air. The fresh air intake can be closed to limit the intake of airborne activity if monitors indicate that such action is appropriate. (A complete discussion of control room ventilation and air conditioning is presented in Chapter 9.)

The limiting post-LOCA recirculation phase CCW temperature transient results from an SSPS Train A failure scenario that conservatively assumes that only three CFCUs are in operation during the injection phase. The highest peak CCW temperature following an MSLB results from a split rupture at 30 percent power with the failure of a main steamline isolation valve. All of the limiting CCW temperature analyses assume 64°F ocean water and a single CCW heat exchanger in service. A separate set of analyses assuming a 70°F ocean water temperature credit two CCW heat exchangers in service to address operation with an elevated UHS temperature (Reference 3). Technical Specifications require that the second CCW heat exchanger be placed in service when the UHS temperature is greater than 64°F.

The CCW system is ~~qualified for a~~ designed to provide maximum post-accident supply temperature of 140°F and a maximum 6-hour~~for a period of up to 6 hours, and a long-term~~ continuous supply temperature of greater than 120°F.

During the recirculation mode, the CCW system is design to remove the heat from the sump (via the RHR heat exchanger) to provide recirculation spray (refer to Section 6.2.2) and emergency core cooling (refer to Section 6.3). Figure 6.2D-8A provides the comparison between the containment sump temperature profile with minimum safeguards, and with and without recirculation containment spray. As shown in Figure 6.2D-8A the long-term sump water temperature with credit of recirculation spray is more limiting for the CCW system due to increased heat addition to the sump from the containment atmosphere by sprays. However, in both cases the predicted CCW temperatures during LOCA conditions are still within the limits of the CCW system temperature acceptance criteria; i.e., a maximum post-accident supply temperature of 140°F for a period of up to 6 hours, and a maximum 6-hourlong-term continuous supply temperature of greater than 120°F.

Therefore, predicted CCW temperatures during both normal and accident conditions are within the limits of the CCW system temperature ~~qualification~~acceptance criteria.

(2) *Single Failure*

The CCW system is designed to continue to perform its safety function following an accident assuming a single active failure during the short-term recovery period and either a single active or passive failure during the long-term recovery period. Refer to Section 3.1.1 for a description of DCPP single failure criteria and definition of terms. During normal operation and up to 24 hours after an accident (the short-term recovery

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

9.4.1.1.3 General Design Criterion 4, 1967 - Sharing of Systems

The CRVS components are not shared by the DCPP Units unless safety is shown not to be impaired by the sharing.

9.4.1.1.4 General Design Criterion 11, 1967 - Control Room

A control room is provided from which actions to maintain safe operational status of the plant can be controlled. The control room is designed to support safe operational status and to maintain safe operational status from the control room or from an alternate location if control room access is lost due to fire or other causes. The control room provides adequate radiation protection to permit access without radiation exposures of personnel in excess of 10 CFR Part 20 limits under normal conditions.

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

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

9.4.1.1.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and accident conditions.

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

The control room is designed to permit access and occupancy of the control room under accident conditions ~~for operating the plant~~ without personnel receiving radiation exposures in excess of 5 rem TEDE ~~GDC 19, 1971, limits~~ for the duration of a design basis accident.

9.4.1.1.8 General Design Criterion 21, 1967 Single Failure

The PG&E Design Class I control room ventilation system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event shall be treated as a single failure.

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

The CRVS is designed to provide backup to the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems.

(2) The two outside air activity area monitors per intake that monitor ~~sample~~ air entering the pressurization system duct are Geiger Muller tube-type general purpose monitors with a 10^{-2} to 10^{4} mR/hr range. Each monitor has a control room readout module with instrument failure alarm. A high radiation signal provides control room annunciation, shuts down the operating CRPS train, and automatically starts the opposite Unit's CRPS.

(3) One area monitor is mounted on the radiation monitoring racks in the control room. This monitor is a Geiger Muller tube-type general-purpose monitor with a 10^{-1} to 10^{4} mR/hr range. This monitor has a control room readout with instrument failure alarm, local alarm, and control room annunciation.

(4) The chlorine monitors at the pressurization outside air duct are abandoned in place as there is no bulk chlorine on site.

High smoke or airborne radioactivity is annunciated in the control room so that the operator can take appropriate action.

9.4.1.3.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

Monitoring devices for control room HVAC systems are capable of detecting low levels of airborne radioactivity. Two outside air activity monitors per intake monitor the~~sample~~ air entering the control room supply duct. Two outside air activity area monitors per intake monitor the ~~sample~~ air entering the pressurization system duct. (refer to Section 9.4.1.3.5) One area monitor is mounted on the radiation monitoring racks in the control room. (refer to Sections 9.4.1.3.5)

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

CRVS Mode 4 operation is automatically initiated on a containment phase A isolation (safety injection) or normal air intake radiation monitor signal. Initiation of Mode 4 operation on a CRVS train initiates Mode 3 operation on the opposite train. Intake closure is designed to occur within 10 seconds or less after initiation of closure signal. Infiltration of activity from outdoors and other areas of the auxiliary building is limited by positive pressure, minimum leakage dampers, zero leakage penetrations, weather-stripped doors, door vestibules, and the absence of outside windows. Administrative controls ensure that all control room entranceways are normally closed.

The CRVS, in conjunction with shielding and administrative controls, is designed to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem ~~whole body, or its equivalent to any part of the body,~~ TEDE for the duration of the most severe design basis accident. An evaluation of post-accident control room radiological exposures is presented in Section 15.5. ~~(Note that for the fuel handling accident in the fuel handling building, a source term is assumed as described in Section 15.5.22).~~

The post-accident sample room HVAC system supports DCPP sampling contingency plan requirements (refer to Reference 23), and continues to meet the above original design basis requirements.

9.4.10.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

The initial checks of the motors, controls, system balance, etc. were made at the time of installation. The verification of the calculated flowrates was also accomplished at this time.

The system is periodically inspected and tested to ensure that all equipment is functioning properly.

9.4.10.5 Instrumentation Applications

Instrumentation provides local indication of the system's operating parameters; i.e., filter differential pressure (indicating usage and cleanliness), and subsystem temperatures and pressures. An interlock is also provided to prevent climate control unit heater operation under a low supply air flow condition.

In addition, an area radiation monitor with local annunciation is installed in the post-accident sample room to warn personnel of high or increasing radiation. High radiation will alarm at the main annunciator (refer to Section 11.4).

9.4.11 TECHNICAL SUPPORT CENTER

The basic function of the TSC HVAC system is to provide protection for personnel working in the TSC ~~center~~ from radiological contaminants and to provide heating, ventilation, and air conditioning for working areas and equipment.

9.4.11.1 Design Bases

9.4.11.1.1 General Design Criterion 4, 1967 - Sharing of Systems

The TSC HVAC system or components are not shared by the DCPP Units unless safety is shown not to be impaired by the sharing.

9.4.11.1.2 10 CFR 50.47 - Emergency Plans

The TSC HVAC system is adequate to support the use of the TSC for emergency response.

9.4.11.1.3 NUREG-0737 (Items II.B.2 and III.A.1.2), November 1980 - Clarification of TMI Action Plan Requirements

II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post Accident Operations: Adequate access to the TSC is provided by design changes, increased permanent or temporary shielding, or post-accident procedural controls.

III.A.1.2 – Upgrade Emergency Support Facilities: NUREG-0737, Supplement 1 (January 1983) provides the requirements for III.A.1.2 as follows:

Section 8.2.1(e) - The TSC is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment.

Section 8.2.1(f) - The TSC is provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem ~~whole body, or its equivalent to any part of the body,~~TEDE for the duration of the accident.

9.4.11.2 System Description

The TSC is provided with its own PG&E Design Class II HVAC system that is schematically shown in Figure 9.4-10. The entire system is manually initiated and is designed to maintain the occupied areas of the TSC at a temperature below 85°F. During normal operation, all the makeup air and recirculated air passes through a roughing filter, a HEPA filter, and the air conditioning unit. Makeup air in the normal operation mode is supplied via the outside air intake by a single makeup air fan. The TSC HVAC system has the capability to manually isolate the area from outside air and to recirculate air via the air conditioning system.

In the radiological accident mode of operation, the TSC HVAC system makeup air is supplied by the control room pressurization system (CRPS) in order to maintain the TSC area at a minimum of +1/8 inch water gauge pressure. Penetrations into the TSC are equipped with penetration seals and floor drain traps with make-up water supplies to prevent exfiltration of the positive pressure from within the TSC through these paths. The pressurization air, and a portion of the recirculated air, is passed through a filter bank containing a HEPA and charcoal filter~~s~~ for cleanup purposes, and supplied to the general area rooms along with the majority of the recirculated air. Exhaust air leaves the TSC by exfiltration.

The TSC HVAC System is fed from a non-Class 1E power source, although it has the capability to be supplied power from a Class 1E bus. The system is not seismically qualified but the ducting, duct supports, and equipment supports are designed and analyzed to seismic requirements. The ducting and components associated with the TSC pressurization air supply are PG&E Design Class I up to and including the manual damper upstream of the redundant duct heaters associated with the TSC filter bank. (refer to Reference 8 of Section 3.2).

The duct heaters maintain the relative humidity of the pressurization air below 70 percent.

9.4.11.3 Safety Evaluation

9.4.11.3.1 General Design Criterion 4, 1967 - Sharing of Systems

The TSC HVAC system is common to Unit 1 and Unit 2 and therefore requires sharing of SSCs between Units. The TSC HVAC system serves no safety functions. The TSC HVAC system is designed to provide adequate heating, ventilation, and air conditioning for working areas and equipment within the TSC. In addition, the CRPS is shared between the control room and the TSC. Sharing of the CRPS by the control room and the TSC is addressed in Section 9.4.1.3.2.

9.4.11.3.2 10 CFR 50.47 - Emergency Plans

The TSC HVAC system meets applicable requirements and is maintained in support of emergency response (refer to Sections 9.4.11.1.3 and 9.4.11.3.3).

9.4.11.3.3 NUREG-0737 (Items II.B.2 and III.A.1.2), November 1980- Clarification of TMI Action Plan Requirements

II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post Accident Operations:

The acceptance criteria for the TSC dose is based on 10 CFR 50.67, in which dose to an operator in the TSC shall not exceed 5 rem TEDE for the duration of the accident. The adequacy of TSC shielding is evaluated for post-LOCA conditions in 12.1.2.7 and Section 15.5.~~Plant shielding has been evaluated with regard to radiation doses at equipment locations and requirements for vital area access/occupancy (including the TSC) for post-accident plant operations.~~

Item III.A.1.2 - Upgrade Emergency Support Facilities: NUREG-0737, Supplement 1 (January 1983)

Section 8.2.1(e) - During normal operation, all the TSC HVAC makeup air and recirculated air passes through the climate control units (CCUs). The CCUs are designed to maintain the occupied areas of the TSC below 85°F, and provide for humidity and cleanliness appropriate for personnel and equipment.

The TSC HVAC system is a PG&E Design Class II system. The pressurization air for the system is supplied by the designated PG&E Design Class I control room pressurization system . The radiological accident mode of operation maintains the TSC area at a positive pressure. The relative humidity of the pressurization air is maintained below 70 percent. The air cleanup portion of the system is equipped with redundant

fans and heaters and a power supply that may be manually switched over to a Class 1E bus source. Post-accident dose in the TSC is discussed in Section ~~6.4.2~~15.5.

Section 8.2.1.(f) - TSC area radiation monitoring instruments provide continuous indication of the general area ambient radiation levels and provide local alarm annunciation in various areas and work spaces. TSC ventilation air monitoring instruments provide continuous sampling of the TSC HVAC return air ducts for detection of airborne radiation and provide for alarm annunciation in the TSC computation center.

The acceptance criteria for the TSC dose is based on Section 8.2.1(f) of NUREG-0737, Supplement 1, as amended by Regulatory Guide 1.183, July 2000, Section 1.2.1, and 10CFR50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident. TSC shielding and the ventilation system prevent post-accident doses inside the TSC from exceeding 5 rem TEDE for the duration of the accident. Refer to Section 9.4.11.2 for a description of the TSC ventilation system. The adequacy of TSC shielding and ventilation is evaluated for post-LOCA conditions in Section 15.5.

9.4.11.4 Inspection and Testing Requirements

Initial checks of the motors, controls, system balance, etc. were made at the time of installation. The system is periodically inspected to ensure that all equipment is functioning properly.

9.4.11.5 Instrumentation Requirements

TSC area radiation monitoring instruments provide continuous indication of the general area ambient radiation levels and provide local alarm annunciation in various areas and work spaces.

TSC ventilation air monitoring instruments provide continuous sampling of the TSC HVAC return air ducts for detection of airborne radioactivity and provide for alarm annunciation in the TSC computation center.

Instrumentation related to post-accident operation includes indication and alarm on low pressurization flow and an alarm for high temperature in the charcoal filter section of the charcoal/HEPA filter bank.

9.4.12 CONTAINMENT PENETRATION AREA GE/GW

The containment penetration area GE/GW ventilation system has the function of maintaining the ambient temperature and pressure of the GE/GW area within acceptable limits during normal operations. The GE/GW area ventilation system is a PG&E Design Class II draw-through type ventilation system.

The containment penetration area GE/GW ventilation system exhausts into the plant vent. The portion from the GE/GW area up to the GE/GW ventilation system isolation valves is PG&E Design Class II.~~is not a safety-related system, and its complete failure has no safety implication~~ The portion of the GE/GW ventilation system from the tie-in to the plant vent (a PG&E Design Class I component) to the two in-series isolation valves is PG&E Design Class I. The PG&E Design Class I portion of the GE/GW ventilation system is designed to provide pressure boundary for the plant vent, to prevent backflow and leakage of radioactive effluents as a result of releases during a design basis accident from the plant vent as follows:

(1) Two In-series, fail-to-close isolation dampers

(2) Instrument Class IA pressure switches initiate closure of the isolation dampers if there is a loss of pressure in the Design Class II duct upstream of the isolation valves.

Two redundant 100 percent capacity fans are provided, so that failure of one fan will not result in loss of ventilation for these areas during the normal plant operation.

The enclosure over the annular space between containment wall and auxiliary building roof at Elevation 140 foot is provided with blow-out type bellows to provide steam relief flow path during HELBA.

9.4.12.3 Safety Evaluation

9.4.12.3.1 General Design Criterion 2, 1967 – Performance Standards

The portion from the GE/GW area up to the GE/GW ventilation system isolation valves is PG&E Design Class II and seismically designed. The portion of the GE/GW ventilation system from the tie-in to the plant vent (a PG&E Design Class I component) to the two in-series isolation valves is PG&E Design Class I.~~The GE/GW ventilation system ducting and supports from isolation dampers to the connection to the plant vent has been seismically qualified for the Hosgri and Double Design Earthquake to maintain structural integrity of the plant vent.~~ All other components are PG&E Design Class II.

9.4.12.3.2 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The ventilation air flows to the enclosed annular gap on Elevation 140 foot, into the duct connected to the exhaust fan and discharged into the plant vent, where it is monitored for radioactivity.

For a description of the radiological monitoring system refer to Section 11.4.2.2.

9.4.12.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Regulatory Position	Compliance	Reasons or Comments
Adsorber leak testing should be conducted whenever DOP testing is done.	No. An in place test of the CRVS, ABVS and FHBVS charcoal adsorbers shows a penetration and system bypass < 1.0% when tested at least once per 24 months.	The in-place testing criteria are established in the Technical Specifications.
6. LABORATORY TESTING CRITERIA FOR ACTIVATED CARBON		
a. Activated carbon adsorber section should be assigned the decontamination efficiencies given in Table 2.	No. The control room ventilation system decontamination efficiencies assumed in the accident analysis are ~~95~~93% / ~~95~~93% for elemental and organic iodine respectively. The control room HVAC system charcoal samples are tested at 30°C/95% RH per ASTM D3803-89 with a 2.5% acceptance criteria versus using RDT M16-1T (1972) at DBA conditions.	The laboratory test acceptance criteria are established in the Ventilation Filter Test Program, which is controlled by the Technical Specifications. Testing per RDT M16-1T (1972) has been superseded by the more conservative test requirements in ASTM D3803-1989. In accordance with Generic Letter 99-02, June 1999 a safety factor of 2 is used in determining the charcoal filter efficiency for use in safety analyses.
	The auxiliary building ventilation system decontamination efficiencies assumed in the accident analysis are ~~90~~88%/~~70~~88% for elemental and organic iodine, respectively, versus 95%/95% as assigned by Regulatory Guide 1.52, June 1973 Table 2. Additionally, the charcoal samples are tested at 30°C/95% RH per ASTM D3803-89 with a ~~15%~~5% acceptance criterion versus using RDT M16-1T (1972) at DBA conditions.	The decontamination efficiencies used in the accident analysis are more conservative than the values stated in Regulatory Guide 1.52, June 1973. The laboratory test acceptance criteria are established in the Technical Specifications. Testing per RDT M16-1T (1972) has been superseded by the more conservative test requirements in ASTM D3803-1989. In accordance with Generic Letter 99-02, June 1999 a safety factor of 2 is used in determining the charcoal filter efficiency for use in safety analyses.
	The fuel handling building ventilation system decontamination efficiencies are 95%/95% as assigned by Regulatory Guide 1.52, June 1973 Table 2. Refer to Table 15.5-45 for a discussion of	The decontamination efficiencies used in the accident analysis are more conservative than the values stated in Regulatory Guide 1.52, June 1973. The laboratory test acceptance criteria are established in the Technical Specifications. Testing per RDT M16-1T (1972) has been superseded by the more

(2) To ensure that the direct radiation from plant structures is sufficiently low so that the total dose at the site boundary from both direct radiation and effluents is within the limits specified in 10 CFR ~~100~~ 50.67 for all postulated accident conditions

(3) To permit continued operation of the other unit on the site in the unlikely event that a design basis accident occurs at one unit

A postaccident radiation shielding design review for DCPP, as required by NUREG 0737 (Reference 7), was performed and is reported in Reference 1.

It is noted that DCPP has incorporated a full implementation of Alternative Source Terms (AST) as defined in Regulatory Guide 1.183, July 2000, Section 1.2.1. The adequacy of the shielding associated with the control room and the technical support center have been re-evaluated using AST (refer to Section 15.5). However, the estimated short-term operator mission doses while performing vital functions post-LOCA, continue to be based on TID-14844 assumptions as documented in Reference 1. This approach is acceptable based on the AST benchmarking study reported in SECY-98-154 (Reference 8) which concluded that results of analyses based on TID-14844 would be more limiting earlier on in the event, after which time the AST results would become more limiting. Post-LOCA access to vital areas usually occurs within the first one or two weeks when the original TID-14844 source term remains limiting.

12.1.2 DESIGN DESCRIPTION

This section discusses the specific design criteria for individual shielding systems required to achieve the overall objectives and describes the actual shielding design.

12.1.2.1 Shielding Locations and Basic Configurations

Figure 1.2-1 shows a plot plan of the site and indicates the location of roads, major plant buildings, and switchyards. It should be noted that the plant site is not served by railroad facilities. Figure 1.2-2 presents a detail of the plant layout and shows the location of outside tanks that could house potentially radioactive materials.

Figures 1.2-4 through 1.2-9 provide scaled plan views of Unit 1 buildings that contain process equipment for treatment of radioactive fluids, and indicate locations and basic configurations of the shielding provided. Figures 1.2-10 through 1.2-12 show similar views of Unit 2 structures. Corresponding sectional views of Unit 1 structures including shielding are shown in Figures 1.2-21 through 1.2-26. Comparable sectional views of Unit 2 structures are shown in Figures 1.2-28 through 1.2-30. Units 1 and 2 are similar with respect to shielding design.

12.1.2.2 General Shielding Design Criteria and Features

Normal radiation levels in the control room are less than 0.5 mR/hr. The limiting case for shielding design is post-DBA conditions. The control room shielding is designed to limit the integrated doses under post-accident conditions to ≤ ~~2.5 rem to the whole body, which is well below the value of~~ 5 rem Total Effect Dose Equivalent (TEDE) as specified in 10 CFR 50.67, ~~10 CFR 50, General Design Criterion (GDC)~~GDC 19, 1999 and NUREG-0737, November 1980, Item III.D.3.4. For a discussion on post-accident shielding adequacy refer to Section 15.5.

12.1.2.7 Technical Support Center Shielding Design

The Technical Support Center (TSC) is designed to be habitable throughout the course of a DBA. Concrete shielding in the walls, roof, and floor is designed to limit the integrated doses under post-accident conditions to ≤ ~~2.~~5 rem ~~to the whole body~~TEDE, consistent with the criterion for the ~~control room~~TSC in Section 8.2.1(f) of NUREG-0737, Supplement 1, as amended by Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. For a discussion on post-accident shielding adequacy refer to Section 15.5.

12.1.2.8 Postaccident Sampling Compartment

The sampling compartment is shielded from external sources by concrete walls and concrete support columns. Personnel should be able to perform necessary postaccident sampling operations without experiencing a radiation dose exceeding the limits specified in NUREG-0737.

12.1.2.9 Old Steam Generator Storage Facility

The old steam generators (OSGs) and old reactor vessel head assemblies (ORVHAs) were removed from DCPP Units 1 and 2 during the steam generator and reactor vessel head replacement projects. These ten large components are temporarily stored in the OSG Storage Facility (OSGSF) specifically constructed for this purpose. The OSGSF meets the radwaste storage requirements for temporary storage of the OSGs and ORVHAs until site decommissioning. The radiological design of the OSGSF meets the radiation shielding requirements of 40 CFR 190, 10 CFR 20, and the DCPP License. The building is designed to have a maximum contact dose rate of 0.2 mR/hr on the exterior wall surface. This value is less than and is bounded by the 0.5 mR/hr radiation dose rate limitation requirement stated in Table 12.1-1 for the Plant Occupancy Zone in which the OSGSF is located (Zone 0 – Unlimited Access). The building design also provides locking access control entrance doors and concrete labyrinths designed to provide shielding.

15.5 RADIOLOGICAL CONSEQUENCES OF PLANT ACCIDENTS

The purposes of this section are: (a) to identify accidental events that could cause radiological consequences, (b) to provide an assessment of the consequences of these accidents, and (c) to demonstrate that the potential consequences of these occurrences are within the limits, guidelines, and regulations established by the NRC.

An accident is an unexpected chain of events; that is, a process, rather than a single event. In the analyses reported in this section, the basic events involved in various possible plant accidents are identified and studied with regard to the performance of the engineered safety features (ESF). The full spectrum of plant conditions has been divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public. The four categories as defined above are as follows:

Condition I: Normal Operation and Operational Transients

Condition II: Faults of Moderate Frequency

Condition III: Infrequent Faults

Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of these conditions is that the most frequent occurrences must yield little or no radiological risk to the public; and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur.

These categories and principles were developed by the American Nuclear Society (Reference 1). Similar, though not identical, categories have been defined in the guide to the Preparation of Environmental Reports (Reference 3). While some differences exist in the manner of sorting the different accidents into categories in these documents, the basic principles are the same.

It should also be noted that the range of plant operating parameters included in the Condition I category, and some of those in the Condition II category, fall in the range of normal operation. For this reason, the radioactive releases and radiological exposures associated with these conditions are analyzed in Chapter 11 and are not discussed separately in this chapter. The analyses of the variations in system parameters associated with Condition I occurrences or operating modes are discussed in Chapter 7 since these states are not accident conditions. In addition, some of the events identified as potential accidents in Regulatory Guide 1.70, Revision 1 (Reference 2), have no significant radiological consequences, or result in minor releases within the range of normal releases, and are thus not analyzed separately in this chapter.

15.5.1 DESIGN BASES

The following regulatory requirements~~, including Code of Federal Regulations (CFR) 10 CFR Part 100, General Design Criteria (GDC), Safety Guides, and Regulatory Guides~~ are applicable to the DCPP radiological consequence analyses presented in this ~~Chapter~~. They form the bases of the acceptance criteria and methodologies as described in the following Sections:

(1) 10 CFR Part 100, "Reactor Site Criteria"

(2) 10 CFR 50.67, "Accident Source Term"

(3) General Design Criterion 19, ~~1971~~ 1999 "Control Room"

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

~~(5) Safety Guide 7, March 1971, "Control of Combustible Gas Concentrations in Containment"~~

~~(6)~~(5) Safety Guide 24, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure"

~~(7)~~(6) Safety Guide 25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"

~~(8)~~(7) Regulatory Guide 1.183, July 2000, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

~~(9) Regulatory Guide 1.195, May 2003, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors"~~

15.5.1.1 List of Analyzed Accidents

The following table summarizes the accident events that have been evaluated for radiological consequences. The table identifies the applicable UFSAR Section describing the analysis and results for each event, the offsite/onsite locations and applicable dose limits, and the radiological analysis and isotopic core inventory codes used.

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
CONDITION II					

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
Loss of Electrical Load (LOL)	15.5.10	EAB and LPZ Control Room~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	~~300 rem~~ ~~25 rem~~2.5 rem TEDE 5 rem TEDE	RADTRAD 3.03~~EMERALD~~	SAS2 / ORIGEN-S~~EMERALD~~
CONDITION III					
Small Break LOCA (SBLOCA)	15.5.11	EAB and LPZ Control Room~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE ~~300 rem~~ ~~25 rem~~ 5 rem TEDE	N/A Refer to Section 15.5.23~~EMERALD~~	N/A Refer to Section 15.5.23~~EMERALD~~

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
Minor Secondary System Pipe Breaks	15.5.12	EAB and LPZ ~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE ~~300 rem~~ ~~25 rem~~	N/A Refer to Section 15.5.18~~N/A~~ ~~Refer to Section 15.5.12~~	N/A Refer to Section 15.5.18~~N/A~~ ~~Refer to Section 15.5.12~~
Inadvertent Loading of a Fuel Assembly	15.5.13	EAB and LPZ ~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE ~~300 rem~~ ~~25 rem~~	N/A Refer to Section 15.5.13	N/A Refer to Section 15.5.13
Complete Loss of Forced Reactor Coolant Flow	15.5.14	EAB and LPZ ~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE ~~300 rem~~ ~~25 rem~~	N/A Refer to Section 15.5.~~14~~10	N/A Refer to Section 15.5.~~14~~10
Under-Frequency	15.5.15	EAB and LPZ ~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE ~~300 rem~~ ~~25 rem~~	N/A Refer to Section 15.5.10~~EMERALD~~	N/A Refer to Section 15.5.10~~EMERALD~~
Single Rod Cluster Control Assembly Withdrawal	15.5.16	EAB and LPZ ~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE ~~300 rem~~ ~~25 rem~~	N/A Refer to Section 15.5.23~~EMERALD~~	N/A Refer to Section 15.5.23~~EMERALD~~
CONDITION IV					
Large Break LOCA (LOCA)	15.5.17	EAB and LPZ Control Room TSC~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~ ~~Control Room~~ ~~Thyroid~~ ~~Whole Body~~	25 rem TEDE 5 rem TEDE 5 rem TEDE ~~300 rem~~ ~~25 rem~~ ~~30 rem~~ ~~5 rem~~	RADTRAD 3.03 PERC2~~EMERALD~~ ~~LOCADOSE~~	SAS2 / ORIGEN-S~~EMERALD~~ ~~ORIGEN-2~~

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
Main Steam Line Break (MSLB)	15.5.18	EAB and LPZ Pre-Accident Iodine Spike ~~Thyroid~~ ~~Whole Body~~ Accident-initiated Iodine Spike ~~Thyroid~~ ~~Whole Body~~ Control Room ~~Thyroid~~ ~~Whole Body~~	 ~~300~~ 25 rem TEDE ~~25 rem~~ ~~30 rem~~ 2.5 rem TEDE ~~30 rem~~ 5 rem TEDE	RADTRAD 3.03~~LOCADOSE~~	SAS2 / ORIGEN-~~SORIGEN-2~~
Main Feedwater Line Break (FWLB)	15.5.19	EAB and LPZ Pre-Accident Iodine Spike Accident-initiated Iodine Spike ~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~	 25 rem TEDE 2.5 rem TEDE ~~300 rem~~ ~~25 rem~~	N/A Refer to Section 15.5.18~~9~~	N/A Refer to Section 15.5.1~~9~~8

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
Steam Generator Tube Rupture (SGTR)	15.5.20	EAB and LPZ Pre-Accident Iodine Spike Accident-initiated Iodine Spike Control Room ~~EAB and LPZ~~ ~~Pre-Accident Iodine Spike~~ ~~Thyroid~~ ~~Whole Body~~ ~~Accident-initiated Iodine Spike~~ ~~Thyroid~~ ~~Whole Body~~ ~~Control Room~~ ~~Thyroid~~ ~~Whole Body~~	25 rem TEDE 2.5 rem TEDE 5 rem TEDE ~~300 rem~~ ~~25 rem~~ ~~30 rem~~ ~~2.5 rem~~ ~~30 rem~~ ~~5 rem~~	RADTRAD 3.03~~RADTRAD~~	SAS2 / ORIGEN-~~SEMERALD-NORMAL~~
Locked Rotor Accident (LRA)	15.5.21	EAB and LPZ Control Room~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~ ~~Control Room~~ ~~Thyroid~~ ~~Whole Body~~	2.5 rem TEDE 5 rem TEDE ~~300 rem~~ ~~25 rem~~ ~~30 rem~~ ~~5 rem~~	RADTRAD 3.03~~EMERALD~~	SAS2 / ORIGEN-~~SEMERALD~~
Fuel Handling-Accident (FHA) ~~Fuel Handling-Area~~	15.5.22~~.1~~	EAB and LPZ Control Room~~EAB and LPZ~~ ~~Control Room~~	6.3 rem TEDE 5 rem TEDE ~~0.063 Sv TEDE (6.3 rem)~~ ~~0.05 Sv TEDE (5 rem)~~	RADTRAD 3.03~~LOCADOSE~~	SAS2 / ORIGEN-~~SORIGEN-2~~

Accident Event	FSAR Section	Boundary	Dose Limit	Radiological Analysis Code(s)	Isotopic Core Inventory Code(s)
~~Fuel Handling Inside Containment~~	~~15.5.22.2~~	~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~ ~~Control Room~~ ~~Thyroid~~ ~~Whole Body~~	~~75 rem~~ ~~6 rem~~ ~~30 rem~~ ~~5 rem~~	~~LOCADOSE~~	~~ORIGEN-2~~
Control Rod Ejection Accident (CREA)	15.5.23	EAB and LPZ Control Room~~EAB and LPZ~~ ~~Thyroid~~ ~~Whole Body~~ ~~Control Room~~ ~~Thyroid~~ ~~Whole Body~~	6.3 rem TEDE 5 rem TEDE ~~300 rem~~ ~~25 rem~~ ~~30 rem~~ ~~5 rem~~	RADTRAD 3.03~~EMERALD~~	SAS2 / ORIGEN-~~SEMERALD~~
Waste Gas Decay Tank Rupture	15.5.24	EAB and LPZ Thyroid Whole Body	300 rem 25 rem	EMERALD	EMERALD
Liquid Holdup Tank Rupture	15.5.25	EAB and LPZ Thyroid Whole Body	300 rem 25 rem	LOCADOSE	EMERALD
Volume Control Tank Rupture	15.5.26	EAB and LPZ Thyroid Whole Body	300 rem 25 rem	EMERALD	EMERALD

15.5.1.2 Assumptions associated with Loss of Offsite Power

The assumptions regarding the occurrence and timing of a Loss of Offsite Power (LOOP) during an accident are selected with the intent of maximizing the dose consequences. A LOOP is assumed for events that have the potential to cause grid perturbation.

i. The dose consequences of the LOCA, MSLB, SGTR, LRA, CREA and LOL event are evaluated with the assumption of a LOOP concurrent with reactor trip.

ii. 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.

In addition, in accordance with current DCPP licensing basis, the non-accident unit is assumed unaffected by the LOOP.

15.5.2 APPROACH TO ANALYSES OF RADIOLOGICAL EFFECTS OF ACCIDENTS

15.5.2.1 Introduction

The potential radiological effects of plant accidents are analyzed by the evaluation of all physical factors involved in each chain of events which might result in radiation exposures to humans. These factors include the meteorological conditions existing at the time of the accident, the radionuclide uptake rates, exposure times and distances, as well as the many factors which depend on the plant design and mode of operation. In these analyses, the factors affecting the consequences of each accident are identified and evaluated, and uncertainties in their values are discussed. Because some degree of uncertainty always exists in the prediction of these factors, it has become general practice to assume conservative values in making calculated estimates of radiation doses. For example, it is customarily assumed that the accident occurs at a time when very unfavorable weather conditions exist, and that the performance of the plant engineered safety systems is degraded by unexpected failures. The use of these unfavorable values for the various factors involved in the analysis provides assurance that each safety system has been designed adequately; that is, with sufficient capacity to cover the full range of effects to which each system could be subjected. For this reason, these conservative values for each factor have been called design basis values.

In a similar way, the specific chain of events in which all unfavorable factors are coincidentally assumed to occur has been called a design basis accident (DBA). The calculated doses for the DBA, provide a basis for determination of the design adequacy of the plant safety systems. In the process of safety review and licensing, the radiation exposure levels calculated for the DBA are compared to the regulatory limits ~~guideline values~~ established in 10 CFR 100.11 and 10 CFR 50.67including acceptance criteria proposed in regulatory guidance, and if these calculated exposures fall below the regulatory guidelines ~~levels~~, the plant safety systems are judged to be adequate.

~~The calculated exposures resulting from a DBA are generally far in excess of what would be expected and do not provide a realistic means of assessing the expected radiological effects of real plant accidents.~~

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

As noted in Section III.2.a of Standard Review Plan Section 15.0.1, Revision 0, (Reference 59), a full implementation of AST addresses a) all the characteristics of AST (i.e., the radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these nuclides), b) replaces the previous accident source term used in all design basis radiological analyses, and c) incorporates the Total Effective Dose Equivalent (TEDE) criteria of 10 CFR 50.67, and Section II of Standard Review Plan 15.0.1, Revision 0.

The dose consequences of the following accidents have been re-evaluated using AST in accordance with Regulatory Guide 1.183, July 2000.

1. Loss of Coolant Accident (LOCA) – Section 15.5.17
2. Fuel Handling Accident (FHA) – Section 15.5.22
3. Locked Rotor Accident (LRA) – Section 15.5.21
4. Control Rod Ejection Accident (CREA) – Section 15.5.23
5. Main Steam Line Break (MSLB) – Section 15.5.18
6. Steam Generator Tube Rupture (SGTR) – Section 15.5.20
7. Loss-of Load (LOL) Event – Section 15.5.10

The tank rupture events (i.e., Rupture of a Waste Gas Decay Tank, Section 15.5.24; Rupture of a Liquid Holdup Tank, Section 15.5.25; Rupture of a Volume Control Tank, Section 15.5.26) represent accidental release of radioactivity accumulated in tanks resulting from normal plant operations, thus the source term characteristics of AST are not applicable to these events.

The dose consequences for the remaining accidents are addressed by qualitative comparison to the seven accidents listed above (with the exception of the tank rupture events).

Note reference to Regulatory Guide 1.183, July 2000 is used extensively within this section, as a result any reference to "Regulatory Guide 1.183" within Section 15.5 refers to Regulatory Guide 1.183, July 2000.

The methodology used to assess the dose consequences of the DBAs, including the specific values of all important parameters, data, and assumptions used in the radiological exposure calculations are listed in the following sections. The computer programs used to assess the dose consequences of the DBAs are described briefly in Section 15.5.8.

As discussed previously, certain- radiological source terms for accidents and some of the releases resulting from Condition I and Condition II events have been included in Chapter 11.

15.5.2.3 Dose Acceptance Criteria

<u>EAB and LPZ Dose</u>

The dose acceptance criteria presented below for the EAB and LPZ reflect use of AST and are applicable to all accidents with the exception of the tank rupture events. The tank rupture events are evaluated against 100 CFR100.11 (refer to Sections 15.5.1.1 and 15.5.24 through 15.5.26 for detail)

The acceptance criteria for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) Dose are based on 10 CFR 50.67, and Section 4.4, Table 6 of Regulatory Guide 1.183:

(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 accident-specific TEDE value noted in Reference 55, Table 6.

(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 radiation dose in excess of the accident-specific TEDE value noted in Reference 55, Table 6.

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

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

Control Room Dose

The acceptance criterion for the control room dose is based on 10 CFR 50.67.

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

This criteria ensures that the dose criteria of GDC 19, 1999 and NUREG-0737, November 1980, Item III.D.3.4 (refer to Section 6.4.1) is met.

Technical Support Center Dose

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

15.5.2.4 Dose Calculation Methodology

The dose calculation methodology presented below reflects use of AST and is applicable to all accidents with the exception of the tank rupture events. The methodology used for the tank rupture events are discussed in the accident specific sections, i.e., Sections 15.5.24 through 15.5.26.

15.5.2.4.1 Inhalation and Submersion Doses from Airborne Radioactivity

Computer Code RADTRAD 3.03 is used to calculate the committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE) from submersion due to airborne radioactivity at offsite locations and in the control room. The summation of CEDE and EDE is reported as TEDE, in accordance with Section 4.1.4 of Regulatory Guide 1.183.

The CEDE is calculated using the inhalation dose conversion factors provided in Table 2.1 of Federal Guidance Report 11 (Reference 41).

The submersion EDE is calculated using the air submersion dose coefficients provided in Table III.1 of Federal Guidance Report 12 (Reference 42). The dose coefficients are derived based on a semi-infinite cloud model. The submersion EDE is reported as the whole body dose in the RADTRAD 3.03 output.

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

Computer Code PERC2 is used to calculate the CEDE from inhalation and the EDE from submersion due to airborne radioactivity in the TSC. PERC2 is a multiple

compartment activity transport code with the dose model consistent with Regulatory Guide 1.183. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included. The CEDE is calculated using the Federal Guidance Report No.11 (Reference 41) dose conversion factors. The EDE in the TSC is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the TSC. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes the integrated activity in the TSC air space, the photon energy release rates per energy group from activity airborne in the TSC, and the ANSI/ANS 6.1.1-1991 neutron and gamma-ray fluence-to-dose factors. (Reference 84)

Offsite Dose

In accordance with Regulatory Guide 1.183, for the first 8 hours, the breathing rate of the public located offsite is assumed to be 3.5×10^{-4} m^3/sec. From 8 to 24 hours following the accident, the breathing rate is assumed to be 1.8×10^{-4} m^3/sec. After that and until the end of the accident, the rate is assumed to be 2.3×10^{-4} m^3/sec. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is calculated and used in determining compliance with the dose criteria in 10 CFR 50.67. The LPZ TEDE is determined for the most limiting receptor at the outer boundary of the low population zone and is calculated for the entire accident duration.

Control Room Dose

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

$$EDE_{finite} = \frac{EDE_{\infty} \, V^{0.333}}{1173}$$

Technical Support Center Dose

The TSC inhalation CEDE is calculated by computer code PERC2 assuming the same breathing rate and occupancy factors as those used in determining the control room dose. The submersion EDE developed by PERC2 (which computes the photon fluence at the center of TSC and utilizes the ANSI/ANS 6.1.1-1991 fluence to effective dose conversion factors), is a close approximation of the dose determined using Table III.1 of Federal Guidance Report No. 12 (Reference 42) (refer to Section 4.1.4,

Regulatory Guide 1.183) and adjusted by the finite volume correction factor given in Regulatory Guide 1.183, Section 4.2.7.

15.5.2.4.2 Direct Shine Dose from External and Contained Sources

Computer program SW-QADCGGP is used to calculate the deep dose equivalent (DDE) in the control room, TSC and at the EAB due to external and contained sources following a LOCA. The calculated DDE is added to the inhalation (CEDE) and the submersion (EDE) dose due to airborne radioactivity to develop the final TEDE. Conservative build-up factors are used and the geometry models are prepared to ensure that un-accounted streaming/scattering paths were eliminated. The dose albedo method with conservative albedo values is used to estimate the scatter dose in situations where the scattering contributions are potentially significant. ANSI/ANS 6.1.1-1977 (Reference 83) is used to convert the gamma flux to the dose equivalent rate.

~~The specific values of all important parameters, data, and assumptions used in the radiological exposure calculations are listed in the following sections. The details of the implementation of the equations, models, and parameters for accidents evaluated using the original licensing basis computer code EMERALD are described in the description of the EMERALD computer program (Reference 4) and the EMERALD NORMAL computer program (Reference 5), which are described briefly in Section 15.5.8.1.~~

~~As discussed earlier, some of the radiological source terms for accidents and some of the releases resulting from Condition I and Condition II events have been included in Chapter 11.~~

15.5.3 ACTIVITY INVENTORIES IN THE PLANT PRIOR TO ACCIDENTS

15.5.3.1 Design Basis Accidents Excluding Tank Ruptures

The fission product inventories in the reactor core, the fuel rod gaps, and the primary coolant prior to an accident have been conservatively calculated based on plant operation at 105% of the current licensed rated thermal power of 3411 MWth, with current licensed values of fuel enrichment and fuel burnup. ~~using the same assumptions, models, and physical data described in Section 11.1, but for different core and plant operating conditions. The pre-accident inventories were calculated using the EMERALD computer code and are similar to those calculated for Tables 11.1-1 through 11.1-12 by the EMERALD-NORMAL code, except for slight differences in some nuclides due to different initial core inventories and irradiation times in the accident calculation.~~

~~The steam system operating conditions assumed for the calculation of pre-accident secondary system inventories are listed in Table 11.1-23. It should be noted that these~~

~~steam system flowrates and masses are approximate lumped values, used for activity balances only, and assume gross lumping of feedwater system component flows and masses. While these values are adequate for activity balances, they should not be used in the context of actual plant flow and energy balances. The activity inventories and concentrations existing in the secondary system are listed in Table 11.1-26.~~

15.5.3.1.1 Core Activity Inventory

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

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

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

The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with three different burnups, i.e., approximately 1/3 of the core is subjected to one fuel cycle, 1/3 of the core to two fuel cycles and 1/3 of the core to three fuel cycles. This approach has been demonstrated to develop an isotopic core inventory that is a reasonable and conservative approximation of a core inventory developed using DCPP specific fuel management history data. Minor variations in fuel irradiation time and duration of refueling outages will have a slight impact on the estimated inventory of long-lived isotopes in the core. However, these inventory changes will have an insignificant impact on the radiological consequences of postulated accidents. A 4% margin has been included in the final isotopic radioactive inventories in support of bounding analyses and to address minor changes in future fuel management schemes.

A 19 month fuel cycle length was utilized in the analysis. The 19-month average fuel cycle is an artifact of the current DCPP fuel management scheme which specifies 3 fuel cycles every 5 years and refueling outages in Spring or Fall.

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

i. A power level of 3580 MWth inclusive of power uncertainty.

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

iii. A maximum core average burnup of 50 GWD/MTU.

The core inventory developed by ORIGEN-S using the above methodology includes over 800 isotopes. The DCPP equilibrium core fission product inventory of dose significant isotopes relative to LWR accidents is presented in Table 15.5-77.

15.5.3.1.2 Coolant Activity Inventory

1. Design Basis Primary and Secondary Coolant Activity Concentrations

Computer code, ACTIVITY2, is used to calculate the design basis primary coolant activity concentrations for both DCPP Unit 1 and Unit 2 based on the core inventory developed using ORIGEN-S and discussed in Section 15.5.3.1. The source terms for the primary coolant fission product activity include leakage from 1% fuel defects and the decay of parent and second parent isotopes. The depletion terms of the primary coolant fission product activity include radioactive decay, purification of the letdown flow and neutron absorption when the coolant passes the reactor core. The nuclear library includes 3rd order decay chains of approximately 200 isotopes.

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

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

the SG liquid concentrations and the partition coefficients recommended in NUREG 0017, Revision 1 (Reference 56).

2. Technical Specification Primary and Secondary Coolant Activity Concentrations

In accordance with Technical Specifications the primary coolant Technical Specification activities for iodines and noble gases are based on 1.0 µCi/gm Dose Equivalent (DE) I-131 and 270 µCi/gm DE Xe-133, respectively.

The Technical Specification based primary coolant isotopic activity reflect the following:

a. Isotopic compositions based on the design basis primary coolant equilibrium concentrations at 1% fuel defects.
b. Iodine concentrations based on the thyroid inhalation weighting factors for I-131, I-132, I-133, I-134, and I-135 obtained from Federal Guidance Report 11 (Reference 41).
c. Noble gas concentrations based on the submersion weighting factors for Xe-133, Xe-133m, Xe-135m, Xe-135, Xe-138, Kr-85m, Kr-87 and Kr-88 obtained from Federal Guidance Report 12 (Reference 42)

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

$$DEI_{131}(i)\,(\mu Ci/gm) = \frac{C(i) \times CT_{tot}}{\sum\{F(i) \times C(i)\}} \qquad (15.5\text{-}1)$$

Where:

F(i) = DCF(i) / DCF $_{I\text{-}131}$
DCF(i)= Federal Guidance Report-11, Table 2-1 (Reference 41) Thyroid Dose Conversion Factor per Nuclide (Rem/Ci)
C(i) = design basis primary coolant equilibrium iodine concentration per nuclide (µCi/gm)
CT_{tot}= primary coolant total (DE I-131) Technical Specification iodine concentration (µCi/gm).

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

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

The Technical Specification secondary liquid iodine concentration is determined using methodology similar to that described above for the primary coolant where CT_{tot} is

0.1 µCi/gm DE I-131, and C(i) is the design basis secondary coolant equilibrium concentrations per nuclide.

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

$$DEX_{133} = \Sigma\{F(i) \times C(i)\} \qquad (15.5\text{-}2)$$

Where:

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

DCF(i) = EPA Federal Guidance Report No. 12 (Reference 42) Table III.1, Dose Coefficient per Nuclide [(rem-m^3)/(Ci-sec)]

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

The noble gas and halogen primary and secondary coolant Technical Specification Activity Concentrations for Unit 1 and Unit 2 are presented in Table 15.5-78. The pre-accident iodine spike concentrations and the equilibrium iodine appearance rates (utilized to develop accident initiated iodine spike values), are presented in Table 15.5-79

15.5.3.1.3 Gap Fractions for Non-LOCA Events

Regulatory Guide 1.183, July 2000, Table 3 provides the gap fractions for Non-LOCA events (with the exception of the CREA) for AST applications. The referenced gap fractions are contingent upon meeting Note 11 of Regulatory Guide 1.183. The burnup criterion associated with the maximum allowable linear heat generation rate is applicable to the peak rod average burnup in any assembly and is not limited to assemblies with an average burnup that exceeds 54 GWD/MTU.

To support flexibility of fuel management with respect to meeting the Regulatory Guide 1.183 criteria for the linear heat generation rate, the gap fractions used for Non-LOCA events (with the exception of the CREA) are based on the bounding (higher) values per isotope / isotope class provided in Safety Guide 25, March 1972 (Reference 25), NUREG/CR-5009, January 1988 (Reference 62) and Regulatory Guide 1.183. Safety Guide 25, March 1972 was traditionally the regulatory guidance used for gap fractions at LWRs for Non-LOCA events with the exception of the CREA. NUREG/CR-5009, January 1988, Section 3.2.2 addresses the impact of extended burn fuel on the gap fractions.

Nuclide Group	Regulatory Guide 1.183 Gap Fraction for Non-LOCA events	DCPP Gap Fraction for Non-LOCA events
I-131	0.08	0.12
Kr-85	0.10	0.30

Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.17

In accordance with Regulatory Guide 1.183, the gap fraction associated with the CREA is as follows:

Noble Gases: 10%
Halogens: 10%

Refer to Tables 15.5-80 for the isotopic concentrations in the gap assumed for the LRA and CREA. The isotopic concentrations assumed for the FHA are presented in Table 15.5-47C.

15.5.3.2 Tank Rupture Events

Activity inventories ~~in various radwaste system tanks~~used for the tank rupture events are ~~also listed~~provided in sections of Chapters 11 and 12 and ~~will be~~are cross-referenced in the sections of this chapter dealing with accidental releases from these tanks.

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

~~*Refueling shutdown studies at operating Westinghouse PWRs indicate that, during cooldown and depressurization of the RCS, a release of activated corrosion products and fission products from defective fuel has been found to increase the coolant activity level above that experienced during steady state operation. An increased core activity release of this sort, commonly referred to as "spiking," could be expected to occur during the depressurization of the RCS as the result of an accident, and should therefore be taken into account in the calculation of post-accident releases of primary coolant to the environment.*~~

~~*Table 15.5-1 illustrates the anticipated coolant activity increases of several isotopes for DCPP during shutdown. This table lists the expected activities during steady state operation and anticipated peak activities during plant cooldown operations. These data are based on measurements from an operating PWR that is similar in design to the DCPP and has operated with significant fuel defects. The measured activity levels for the operating plant are also included in Table 15.5-1.*~~

~~*The dominant nongaseous fission product released to the coolant during system depressurization is I-131. The activity level in the coolant was observed to be higher than the normal operating level for nearly a week following initial plant shutdown with the system purification rate varying between approximately 1 x 10^{-5} and 3 x 10^{-5} per second. Although lesser in magnitude, the other fission product particulates (cesium isotopes) exhibited a similar pattern of release and removal by purification. It is reasonable to project these data to the DCPP since the purification constants are*~~

~~*similar, and it is standard operating procedure to purify the coolant through the demineralizers during plant cooldown.*~~

~~*Fission gas data from operating plants indicate a maximum increase of approximately 1.5 over the normal coolant gas activity concentration. However, system degassification procedures are implemented prior to and during shutdowns, and have proven to be an effective means for reducing the gaseous activity concentration and controlling the activity to levels lower than the steady state value during the entire cooldown and depressurization procedure. Although a steady state Xe-133 concentration of 127 μCi/gm was observed prior to degassification procedures (see Table 15.5-1), the maximum coolant concentration during the reactor depressurization was 65 μCi/gm. Further, the coolant activity was then reduced to approximately 1 μCi/gm in less than two days of degassification.*~~

~~*The corrosion product activity releases have been determined to be predominantly dissolved Co-58. From Table 15.5-1, it is noted that this contribution is less than 1 percent of the total expected coolant activity and is, therefore, considered to be a minor contribution.*~~

~~For the calculation of the effect of spiking on accidental plant releases, the original licensing basis assumed the dominant isotopes were iodines, and all others were neglected. Using the measured I-131 concentrations given in Table 15.5-1 and a primary purification rate of 1 x 10^{-5} per second, effective I-131 fuel escape rate to the reactor coolant during a spike of 30 times the normal equilibrium value was calculated. This value was then applied to all iodine isotopes. Subsequent analyses assumed an accident-initiated spike of 500 times (for MSLB, refer to Section 15.5.18) and 335 times (for SGTR, refer to Section 15.5.20) the normal equilibrium value, as described in the appropriate subsections that follow, to be consistent with Technical Specification 3.4.16.~~

~~The duration of the spike was assumed to be 8 hours. This assumption can be justified by examining graphs of I-131 coolant concentration versus time during shutdowns for operating BWR plants (Reference 14). The assumption that the fuel escape rate continues at the elevated rates discussed above for the full 8 hours of the spike is conservative. The effect of iodine spiking was included in all accidents that involved leakage of primary coolant directly or indirectly to the environment.~~

15.5.4 ~~EFFECTS OF PLUTONIUM INVENTORY ON POTENTIAL ACCIDENT DOSES~~DELETED

~~HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.~~
~~*Because of the somewhat higher fission yields of some isotopes associated with thermal fissions in Pu-239, a sensitivity study was conducted to determine the possible influence of this effect on potential accident doses.*~~

~~*This study demonstrates that accident doses are only slightly affected by the incorporation of Pu-239 fission yields into total core fission yields, even using the EOL plutonium inventories. The resulting differences, listed in Table 15.5-2, indicate that thyroid doses generally increase from 4 to 6 percent, and whole body doses generally decreased, from 2 to 5 percent, assuming the accident occurred at EOL.*~~

~~*In this study, total core fission yields were calculated by a mass weighting of U-235 fission yields and Pu-239 fission yields. Because the core mass of U-235 is considerably greater than the core mass of Pu-239, total core fission yields are close to U-235 fission yields. The masses of U-238 and Pu-241 that fission are extremely small, and thus U-238 and Pu-241 have essentially no effect on the total core fission yields.*~~

15.5.5 POST-ACCIDENT METEOROLOGICAL CONDITIONS

15.5.5.1 Design Basis Accidents (Excluding Tank Ruptures)

The EAB and the LPZ atmospheric dispersion factors (χ/Q) utilized in the dose consequence analyses have been developed using Regulatory Guide 1.145, Revision 1 methodology and a continuous, temporally representative 5-year period of hourly meteorological data from the DCPP onsite meteorological tower; January 1, 2007 through December 31, 2011. Refer to Section 2.3.5.2.1 and Table 2.3-145.

Using the same hourly meteorological data, the χ/Q values applicable to on-site locations such as the control room and TSC, have been calculated using the "Atmospheric Relative CONcentrations in Building Wakes" (ARCON96) methodology (Reference 61). Refer to Section 2.3.5.2.2.

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

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

Note that the specific control room χ/Q values used in each of the accident analyses (and the specific TSC χ/Q values used for the LOCA) are presented in the accident-specific tables presented in Chapter 15.5. The χ/Q values selected for use in the dose consequence analyses are intended to support bounding analyses for an accident that occurs at either unit. They take into consideration the various release points-receptors applicable to each accident in order to identify the bounding χ/Q values and reflect the allowable adjustments and reductions in the values as discussed earlier and further summarized in the notes of Tables 2.3-147 through 2.3-149.

15.5.5.2 Tank Rupture Events

For the analyses of offsite doses from the ~~DBAs~~tank rupture events, the rare and unfavorable set of atmospheric dilution factors assumed in the ~~NRC~~ Regulatory Guide 1.4, Revision 1 (Reference 6) was used. On the basis of meteorological data collected at the DCPP site, these unfavorable dilution factors, assumed for the design bases cases, are not expected to exist for onshore wind directions more than 5 percent of the time. The particular values used for this site are given in Table 15.5-3.

~~For the analyses of offsite doses from the expected case accidents, the assumed atmospheric dilution factors are listed in Table 15.5-4. For these cases, 10 percent of the design basis case numbers were used. On the basis of study of the site data at DCPP, this assumption will result in calculated exposures higher than would be expected.~~

~~Because of the low probability of occurrence associated with these assumed dilution factors, significant downwind decay, variable shifts in population distribution due to possible emergency evacuation, and large variations in concentrations due to downwind topographical characteristics, appropriate assumptions for population exposure (man-rem) estimates following a significant accidental release are difficult to select. It is clear that using the same factors of conservatism established for individual exposures at locations near the site (the regulatory guide dilution factors) would yield calculated population exposures much higher than could physically occur. For these reasons, the population exposures (man-rem) for the expected cases have been calculated using the long-term dilution factors given in Table 15.5-5, and ten times these values have been assumed for the DBA cases.~~

Effects of release duration on downwind ground level concentration have been measured directly and determined theoretically from knowledge of the horizontal and vertical spectrum of turbulence. Both the observations and theory generally agree that only the horizontal components of turbulence near ground level contain any significant amount of energy in periods longer than a few seconds. As a result, only the lateral dimension of the cloud need be modified for concentration estimates for noncontinuous releases. Slade (Reference 7) using the approach recommended by Cramer, gives a time-dependent adjustment of the lateral component of turbulence to be:

$$\sigma_\theta(T) = \sigma_\theta(T_o)\,(T/T_o)^{0.2} \qquad (15.5\text{-}\cancel{13})$$

where:

$\sigma_\theta(T)$ = lateral intensity of turbulence of a time period T, where T is a value less than 10 minutes

$\sigma_\theta(T_o)$ = lateral intensity of turbulence measured over a time period T_o, where T_o is on the order of 10 minutes

Near a source there is a direct linear relationship between σ_θ and the plume crosswind dimension σ_y so that the σ_y versus distance curves presented by Slade can be directly scaled by the factor $(T/T_o)^{0.2}$ to provide estimates of a reference σ_y at about 100 meters downwind from the source. Beyond this distance, the lateral expansion rates for continuous and noncontinuous point source releases are approximately the same, and thus the ratio of short-term release concentration to continuous release concentration for point sources is independent of stability class, downwind distance, or windspeed. For distances less than a few thousand meters the ratio approaches unity as the volume of the source increases.

Using the above scaling concept, the dilution equation in Regulatory Guide 1.4, and the cloud dimension curves given by Slade, the ratio of short-term release concentration to continuous release concentration was calculated for several different release durations (Figure 15.5-1). For a 10-second duration, the short-term dilution factor is only 2.3 higher than the continuous release dilution factor, and thus the appropriate short-term release correction is within the uncertainty limits of the continuous release dilution factor.

~~The various plant accidents considered in Sections 15.2, 15.3, and 15.4 may result in activity release through various pathways: containment leakage, secondary steam dumping, ventilation discharge, and radioactive waste system discharge.~~

~~Post-accident containment leakage is a slow continuous process, and thus continuous release dilution factors apply for these cases.~~

~~Because of secondary loop isolation capabilities and because significant activity release is accompanied by large steam release, secondary steam dumping accidents release significant quantities of activity only through relief valves. Relief valve flow limitations combined with large steam release result in activity releases of long duration. Thus continuous release dilution factors apply for these cases.~~

~~The approximate duration of a ventilation discharge activity release can be estimated by dividing the volume of contaminated air by the discharge flowrate. Because estimates of~~ The release duration for liquid holdup tank rupture, gas decay tank rupture~~,~~ and the volume control tank rupture, ~~and fuel handling area accident~~ are all over ~~in~~less than 10

minutes~~,~~. As discussed above, continuous release dilution factors apply for these cases~~.~~

~~Continuous release dilution factors have been applied to all Conditions II, III, and IV accidents discussed in Chapter 15 for the following reasons:~~

~~Almost all Conditions II, III, and IV releases are definitely long-term releases~~

~~Releases that might be considered short-term releases result in exposures well within 10 CFR Part 100 limits~~

~~Short-term release dilution factors are only about twice as high as continuous release dilution factors~~

~~The appropriate short-term release corrections are within the range of the uncertainties in the continuous release dilution factors~~

~~Furthermore, the above reasons indicate that~~ and a more sophisticated or complex short-term release dilution model is not justified.

~~The atmospheric dispersion factors for pressurization and infiltration air flows to the control room are analyzed using the modified Halitsky χ/Q methodology, which is discussed below.~~

~~As a result of the TMI accident, the NRC, in NUREG-0737 Section III.D.3.4, asked all nuclear power plants to review their post-LOCA control room habitability designs using the guidance of Standard Review Plan (SRP) 6.4 and the 1974 Murphy-Campe (M-C) paper (Reference 17). These reviews concluded that the atmospheric dispersion factor (χ/Q) methodology recommended in the M-C paper was overly conservative and inappropriate for most of the plant designs. The M-C equations are based primarily on the Halitsky data for round-topped EBR-II (PWR type) containments and are valid only for intake locations at least a half containment diameter from the containment wall. In most cases, however, the intake locations are closer to the building causing the wake. Thus, review of recent literature on building wake χ/Qs, models, wind tunnel tests, and field measurements resulted in the modified Halitsky χ/Q model.~~

~~Historically, the preliminary work on building wake χ/Qs was based on a series of wind tunnel tests by James Halitsky et al. Halitsky summarized these results in Meteorology and Atomic Energy 1968, D. H. Slade, Editor (Reference 7). In 1974 K. Murphy and K. Campe of the NRC published their paper based on a survey of existing data. This χ/Q methodology, which presented equations without derivation or justification, was adopted as the interim methodology in SRP 6.4 in 1975. Since that time, a series of actual building wake χ/Q measurements have been conducted at Rancho Seco~~

~~(Reference 25), and several other papers have been published documenting the results of additional wind tunnel tests (see References 26 through 31).~~

~~The Diablo Canyon plant complex is composed of square-edged buildings and two cylindrical containment buildings. Infiltration air into the control room would come from the auxiliary building, which has air intakes slightly above the control room. This intake of air will be subject to building wake caused by the portion of the containment building above the highest roof elevation of the auxiliary building. Pressurization air for the control room is provided from intakes on the turbine building. The intake will be subject to building wake caused by a portion of the containment building above the turbine building roof and a portion of the turbine building wall facing west and the wall facing north.~~

~~J. Halitsky's efforts, summarized in Reference 7, present the basic equation as follows:~~

$$\cancel{\chi / Q = K / A\bar{u}} \qquad \cancel{(15.5\text{-}2)}$$

~~where~~

~~A = cross sectional area, m^2 orthogonal to $\bar{u}$~~

~~$\bar{u}$ = wind speed, m/s~~

~~K = isopleth (concentration coefficient - dimensionless)~~

~~It is found in many cases that the above Halitsky equation still provides a reasonable estimate of χ/Q. The following correction factors can be applied to this equation to account for situation and plant-specific features:~~

- ~~Stream line flows are used in most wind tunnel tests~~
- ~~Release points are generally much higher than 10 meters above ground~~
- ~~Null wind velocity is observed at certain periods of time~~
- ~~Isothermal temperatures are used in wind tunnel tests~~
- ~~Buoyancy and jet momentum effects are ignored~~

~~Typical 1 hr field tests account for plume meander effects, while 3 to 5 minute wind tunnel tests do not.~~

~~A modified Halitsky χ/Q methodology, formulated by R. Bhatia, et al (Reference 32), is presented below.~~

$$\chi/Q = \frac{K}{A\bar{u}} \times f_1 \times f_2 \times f_3 \times f_4 \times f_5 \times f_6 \ (sec/m^3) \qquad (15.5\text{-}3)$$

This modified Halitsky methodology is inherently conservative because the wind is assumed to be blowing towards the control room during the first or worst part of the accident, and because 5 percent wind speeds are used rather than 50 percent. In addition, the adjustment factors are always biased towards the minimum reduction that the data justifies.

As a test of the modified Halitsky method, calculated values of χ/Q, without using factors f_4 and f_5 due to their uncertainty, were compared to the 1 hour field test χ/Q data from Rancho Seco. Only one χ/Q was found to be higher than the calculated value. This was due to an external wake influence caused by wind channeling between the nearby cooling towers. The wind channeling prevented the normal wake turbulence and variation effects over time, which normally spread the plume over a wide area. In most cases the modified Halitsky χ/Q was found to be a conservative estimate of the measured χ/Q; in some cases it was significantly higher.

The choice of K factors and the suggested modifying factors, f_1, f_2, etc., are discussed below.

K factors:

The choice of an appropriate K factor from the wind tunnel test data is critical for the χ/Q estimate to be valid. Halitsky in Reference 7 has several sets of K isopleths for round-topped containments (for PWRs) and block buildings (for BWRs). Multiple building complexes must be simulated by single equivalent structures. The effluent velocity to wind speed ratio of approximately 1 is valid for most power coolant systems. Various angles of wind incidence are shown to account for vortexing that could result in worse conditions than a wind normal to the building face. K factors should be estimated for various combinations of wind incidence angle and the appropriate effective building cross-sectional area causing the wake (not just the containment area) to determine the peak value, as was done by Walker (Reference 26).

The K factors were determined from Figure 5.29c in Reference 7, based on a conservative analysis of the locations for infiltration and pressurization intake airflows and the appropriate dimensions relative to the containment. A single pressurization intake nearest the containment was assumed. The selected K factors and appropriate building cross-sectional areas used for the base χ/Q values are given below. The 5 percent wind speed was derived from an analysis of Diablo Canyon meteorological data over a period of 10 years.

Case	K	$\bar{u}$ (meter/second)	A(m^2)	Base χ/Q (sec/m^3)

~~Pressurization~~	~~4~~	~~1~~	~~3690~~	~~1.084×10^{-3}~~
~~Infiltration~~	~~5~~	~~1~~	~~1661~~	~~3.01×10^{-3}~~

~~$\bar{u}$, *wind speed:*~~

~~Halitsky's K values are based on wind speeds measured at the top of the containment or building. Therefore, the M-C 5 percent wind speed at a 10 meter height should be adjusted to the actual speed at the top of containment or release point. The 5 percent wind speed is adjusted using the formulation presented by Wilson (Reference 30) as follows.~~

$$\bar{u}_T = \bar{u}\left(\frac{Z}{Z_{Ref}}\right)^{.23} \qquad (15.5\text{-}4)$$

~~where:~~

~~$\bar{u}_T$ = wind speed at height Z~~
~~Z_{Ref} = 10 meters (5% wind speed reference height)~~

~~*f_1, wind speed change factor/ f_2, wind direction change factor:*~~

~~The factors shown below were used. They are based on Diablo Canyon meteorological data for a 10 year period of record.~~

Time Periods	f_1	f_2
~~0 - 8 hrs~~	~~1.0~~	~~1.0~~
~~8 - 24 hrs~~	~~0.83~~	~~0.92~~
~~24 - 96 hrs~~	~~0.66~~	~~0.84~~
~~96 - 720 hrs~~	~~0.48~~	~~0.67~~

~~*f_3, wind turbulence effect:*~~

~~Wilson in Reference 30 and field tests confirm Halitsky's statement that his K isopleths are a factor of 5 to 10 too conservative due to not accounting for random fluctuations of the wind approaching the building. Therefore, a factor of 0.2 was used for f_3.~~

~~*f_4, elevated release effect:*~~

~~Bouwmeester et al. (Reference 31) indicate that there are up to 10 null wind speed conditions during an hour of data collection. During these periods the effects of jet momentum, plume rise and buoyancy would result in the radioactive effluent being discharged above the effective wake boundary and thus not entering the wake cavity. A reduction factor of 1 was used.~~

~~*F_5, time averaging effects:*~~

~~Wind speed variations and wind direction meandering effects are not modeled in wind tunnel tests to account for this effect. Reference 31 indicates the use of the following equation:~~

$$C_p = C_m \left(\frac{t_p}{t_m} \right)^{-1/2} \quad (15.5\text{-}5)$$

~~where:~~

~~C_p = prototype concentration~~
~~C_m = model concentration~~
~~t_p = prototype sampling time~~
~~t_m = model equivalent sampling time~~

~~Normal wind tunnel data is taken for 3 to 10 minute samples. Thus, for a 1-hour field test, C_p = 0.22 to 0.41 C_m, and for an 8 hour field test, C_p = 0.08 to 0.14. C_m A value of 0.5 was conservatively assumed for f_5.~~

~~*f_6, adjustments to top of containment:*~~

~~To account for wind speed at the top of containment, instead of the M-C 5 percent wind speed at 10 meter height, the factor $f_6 = \bar{u}/\bar{u}_T$ was included. The f_6 value equals 0.65.~~

~~Table 15.5-6 presents the resultant atmospheric dispersion factors (χ/Q) calculated using the modified Halitsky χ/Q methodology. These dispersion factors do not take credit for dual pressurization inlets and do not include the control room occupancy factors.~~

15.5.6 RATES OF ISOTOPE INHALATION

15.5.6.1 Design Basis Accidents (Excluding Tank Ruptures)

The breathing rates used in the calculations of inhalation doses are listed in Table 15.5-7A. These values are based on the average daily breathing rates provided in Section 4.1.3 of Regulatory Guide 1.183.

15.5.6.2 Tank Rupture Events

The breathing rates used in the calculations of inhalation doses are listed in Table 15.5-7. These values are based on the average daily breathing rates assumed in

ICRP Publication 2 (Reference 8) which are also used in Regulatory Guide 1.4, Revision 1. ~~The active breathing rates are used for all onsite dose calculations, which are based on expected exposure times.~~

15.5.7 DELETED ~~POPULATION DISTRIBUTION~~

~~The distribution of population surrounding the plant site, which was used for the population exposure calculations, is discussed in Section 2.1, and the population distribution used is listed in Table 15.5-8. The actual post-accident population distribution could be significantly lower if any evacuation plan were implemented.~~

15.5.8 RADIOLOGICAL ANALYSIS PROGRAMS

15.5.8.1 ~~DESCRIPTION of the~~ EMERALD (Revision I) and EMERALD-NORMAL (Tank Rupture Events) ~~Program~~

EMERALD is used to develop the source term for the tank rupture events and assess the dose consequences at the EAB and LPZ following a waste gas decay tank rupture and a volume control tank rupture.

The EMERALD program (Reference 4) is designed for the calculation of radiation releases and exposures resulting from abnormal operation of a large PWR. The approach used in EMERALD is similar to an analog simulation of a real system. Each component or volume in the plant that contains a radioactive material is represented by a subroutine, which keeps track of the production, transfer, decay, and absorption of radioactivity in that volume. During the course of the analysis of an accident, activity is transferred from subroutine to subroutine in the program as it would be transferred from place to place in the plant. ~~For example, in the calculation of the doses resulting from a LOCA, the program first calculates the activity built up in the fuel before the accident, then releases some of this activity to the containment volume. Some of this activity is then released to the atmosphere.~~ The rates of transfer, leakage, production, cleanup, decay, and release are read in as input to the program.

Subroutines are also included that calculate the onsite and offsite radiation exposures at various distances for individual isotopes and sums of isotopes. The program contains a library of physical data for 25 isotopes of most interest in licensing calculations, and other isotopes can be added or substituted. Because of the flexible nature of the simulation approach, the EMERALD program can be used for most calculations involving the production and release of radioactive materials, including design, operational and licensing studies. The complete description of the program, including models and equations, is contained in Reference 4.

The EMERALD-NORMAL program (Reference 5) is a program incorporating the features of EMERALD, but designed specifically for releases from normal and near-

normal operating conditions. It contains an expanded library of isotopes, including all those of interest in gaseous and liquid environmental exposures. Models for a radwaste system are included, using the specific configuration of radwaste system components in the DCPP. The program contains a subroutine for doses via liquid release pathways developed by the Bechtel Corporation and a tritium subroutine. The code calculates activity inventories in various radwaste tanks and plant components which are used for the initial conditions for accidents involving these tasks. In addition, it is used in some near-normal plant conditions classified in this document as Condition I and Condition II and discussed in Chapter 11.

15.5.8.2 ~~Description of the~~ LOCADOSE ~~Program~~

The LOCADOSE program (Reference 47) is designed to calculate radionuclide activities, integrated activities, and releases from a number of arbitrarily specified regions. One region is specified as the environment. Doses and dose rates for five organs (thyroid, lung, bone, beta skin, and whole body) can be calculated for each region, and for a number of offsite locations with specified atmospheric dispersion factors. The control room can be specified as a special region for convenience in modeling airborne doses to the control room operators.

LOCADOSE is also used to assess the dose consequences at the EAB and LPZ following a liquid holdup tank rupture.

15.5.8.3 DELETED~~Description of the ORIGEN-2 Program~~

~~The core inventory and gamma ray energy spectra of post-accident fission products for selected accidents (See Section 15.5.1) were computed using the ORIGEN-2 computer program. ORIGEN-2 (Reference 50) is a versatile point depletion and decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions of materials contained therein. This code represents a revision and update of the original ORIGEN computer code which has been distributed world-wide beginning in the early 1970s. Included in it are provisions for incorporating data generated by more sophisticated reactor physics codes, free-format input, the ability to simulate a wide variety of fuel cycle flowsheets, and more flexible and controllable output features~~

15.5.8.4 DELETED~~Description of the ISOSHLD Program~~

~~ISOSHLD (Reference 9) is a computer code used to perform gamma ray shielding calculations for isotope sources in a wide variety of source and shield configurations. Attenuation calculations are performed by point kernel integration; for most geometries this is done by Simpson's rule numerical integration. Source strength in uniform or exponential distribution (where applicable) may be calculated by the linked fission product inventory code RIBD or by other options as desired. Buildup factors are calculated by the code based on the number of mean free paths of material between the source and detector points, the effective atomic number of a particular shield region (the~~

~~last unless otherwise chosen), and the point isotropic Nuclear Development Associates (NDA) buildup data available as Taylor coefficients in the effective atomic number range of 4 to 82. Other data needed to solve most isotope shielding problems of practical interest are linked to ISOSHLD in various libraries.~~

15.5.8.5 Description of the ISOSHLD II Program

ISOSHLD II (Reference 11) is a shielding code that is principally intended for use in calculating the radiation dose, at a field point, from bremsstrahlung and/or decay gamma rays emitted by radioisotope sources. This program, with the newly-added bremsstrahlung mode, is an extension of the earlier version (ISOSHLD). Five shield regions can be handled with up to twenty materials per shield; the source is considered to be the first shield region, i.e., bremsstrahlung and decay gamma rays are produced only in the source. Point kernel integration (over the source region) is used to calculate the radiation dose at a field point.

ISOSHLD II is used to determine the dose to the control room operator due to direct shine from the airborne activity inside the containment following a LOCA during daily ingress / egress for the duration of the accident.

15.5.8.6 DELETED~~Description of the RADTRAD Program~~

~~RADTRAD (Reference 52) uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation. The RADTRAD code can be used to assess occupational radiation exposure, typically in the control room, as well as site boundary doses, and to estimate the dose attenuation due to modification of a facility or accident sequence.~~

15.5.8.7 SAS2 / ORIGEN-S

ORIGEN-S is part of the SCALE 4.3 suite of codes which was developed by Oak Ridge National Laboratory (ORNL) for the NRC to perform standardized computer analyses for licensing evaluations. SAS2 is a control module that provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. SAS2 / ORIGEN-S (Reference 64) calculates the time-dependent neutron flux and the buildup of fissile trans-uranium nuclides. It properly accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions. It can calculate accurately the neutron-activated products, the actinides and the fission products in a reactor core.

SAS2/ORIGEN-S is used to develop the equilibrium core activity inventory and the decayed fuel inventories after shutdown utilized to assess the design basis accidents excluding the tank ruptures.