PGE Portland General Electric Company

James E. Cross Vice President, Nuclear

November 14, 1990

Trojan Nuclear Plant
Docket 50-344
License NPF-1

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington DC 20555

Dear Sirs:

Adoption of NUREG/CR-5055 Methodology to Model
Atmospheric Diffusion for Control Room Habitability Assessments

In May 1988, the Nuclear Regulatory Commission (NRC) published NUREG/CR-5055 (PNL-6391), "Atmospheric Diffusion for Control Room Habitability Assessments". NUREG/CR-5055 presents the results of an evaluation of the procedure used by the NRC staff to assess control room habitability at nuclear power plants. The evaluation is based upon experimental data from tests at seven nuclear facilities.

Portland General Electric Company (PGE) has reviewed the methodology developed in NUREG/CR-5055 for estimating atmospheric diffusion and plans to implement the applicable portions in the Trojan Nuclear Plant control room habitability analysis (Design Basis of the Plant). The changes resulting from NUREG/CR-5055 considerations do not involve an unreviewed safety question nor a change to the Trojan Technical Specifications and therefore will be implemented in accordance with 10 CFR 50.59. These changes will be reflected in a future amendment to the Trojan Final Safety Analysis Report (FSAR). The revised FSAR pages are included as Attachment A. PGE is providing the NRC this advance notice of our intentions prior to implementation.

Based on the PGE review, the portions of the methodology which have been found to be applicable to the Trojan Nuclear Plant are:

1. The composite wake model for ground-level releases (Equation 14) including a factor of four conservatism as is recommended on NUREG/CR-5055 Page 59. Equation 14 as listed on NUREG/CR-5055 Page 40 is:

$$X/Q = 1/(Fo + Fp + Fw) \qquad \text{(Equation 14)}$$

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2. The equations for deriving the Equation 14 terms. These equations are described in NUREG/CR-5055 Pages 39 and 40.

   Fo = activity release rate/released activity concentration

   $1/Fw = k * x^{(a)} * A^{(b)} * U^{(c)} * S^{(d)}$ (Equation 10)

   $Fp = [Pi * U * sigma(y) * sigma(z)]$ (Equation 13)

3. The Equation 14 coefficients given on Page 37 of NUREG/CR-5055:

   k = 100
   a = -1.2
   b = -1.2
   c = 0.68
   d = 0.5

The safety evaluation performed for this change indicated that the estimated doses to control room personnel significantly decrease for all accidents except for a waste gas decay tank rupture. For a waste gas decay tank rupture the estimated inhalation thyroid dose to control room personnel increases insignificantly (from 24 mrem to 69 mrem) compared to the other accidents and the NRC acceptance criteria (30,000 mrem) such that it has been determined to not be an unreviewed safety question and does not involve significant hazards.

The cause of the increase in inhalation thyroid dose to control room personnel following a waste gas decay rupture is due to lower gamma dose rates in the control room. The lower gamma dose rates are not high enough to cause the control room Area Radiation Monitor (ARM-11) to isolate the control room. Since the control room normal ventilation system continues running until secured by operator action, the radioiodine buildup is assumed to instantly reach outside air concentrations. The initial higher radioiodine uptake causes the inhalation thyroid dose to be higher.

NUREG/CR-5055 proposes two diffusion models, the "new wake" model and the "composite wake" model. PGE has elected to use the composite wake model. NUREG/CR-5055, on Page 61, indicates the composite wake model should be used when the intake point is not on the same building as the release point. This is the case for Trojan and justifies the selection of the composite wake model.

NUREG/CR-5055 contains certain self-imposed limitations. PGE has found these limitations to be applicable to the use of the composite wake model at Trojan as described below.

1. The volumetric flow "Fo" shall be the larger of the release point and air intake flows when credit for volumetric flow is taken.

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2. Ultimately, the distance in the wake model (Fw) is to be limited to 20 times the square root of the projected building area. This limit roughly corresponds to the site exclusion area boundary shown on Trojan Technical Specification Figure 5.1-1. NUREG/CR-5055 is only used to estimate X/Q values within the site exclusion area boundary.

3. Discussions with the author indicate the absence of long-term meteorology corrections in NUREG/CR-5055 is not to be construed to mean such corrections are inappropriate. Rather, the use of the current long-term meteorology adjustment factors should be continued.

   The Murphy/Campe methodology, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19", (Thirteenth Atomic Energy Commission Air Cleaning Conference, August 1974) will continue to be used to calculate post-accident doses to control room personnel except for short-term X/Q estimates.

4. The composite wake model when used at Trojan should be limited to windspeeds above 0.5 meters per second.

Sincerely,

James E. Cross

Attachment

c: Mr. John B. Martin
Regional Administrator, Region V
U.S. Nuclear Regulatory Commission

Mr. David Stewart-Smith
State of Oregon
Department of Energy

Mr. R. C. Barr
NRC Resident Inspector
Trojan Nuclear Plant

Regulatory Guide 1.4. The atmospheric dispersion factors for the fission product cloud external to the Containment are given in Section 15.6.5.6. The maximum whole body doses from these pathways are given in Table 15.6-16.

6.4.4.1.2 Radiation Inside Control Room

An updated control room radiation analysis has been performed utilizing the assumptions outlined in "Nuclear Power Plants Control Room Ventilation System for Meeting General Design Criteria 19" by Murphy and Campe[1], and as outlined in NUREG/CR-5055 (Reference 35 of Section 15.6). Assumptions and results are provided in Section 15.6.5.6.

6.4.4.2 Toxic Gas Protection

6.4.4.2.1 Onsite Storage

The only significant toxic gas stored onsite is chlorine. Chlorine is stored in sixteen 1-ton cylinders in the Chlorine Building (see Figures 12.3-1 and 1.2-19). Sections 2.2.3 and 9.3.6 describe the chlorine system in detail. This location is approximately 330 feet from the control room intake and about 75 feet vertically below the control room ventilation intake.

The control room normal ventilation system (CB-2) is equipped with redundant chlorine detectors installed at the outside air intake duct. These detectors initiate isolation of the control room, within 19 seconds when chlorine concentration exceeds 5.0 ppm by volume. In addition, the Chlorine Building is equipped with a chlorine detector which alarms in the control room to allow necessary operator action before the chlorine gas can reach the control room (Section 9.4.1.2.1).

A Plant procedure describes the emergency procedures to be activated in the event of a chlorine release.

This isolation is described in detail in Section 9.4.2.

The only other significant toxic materials stored at the Plant are the acid (15,000 gallons) and caustic (4750 gallons) storage tanks located at Elevation 45 feet of the Turbine Building (see Figures 1.2-2 and 9.2-8). The principal hazard that would occur from this storage is if the acid (sulfuric acid) and caustic were somehow mixed, sulfur dioxide could then be formed by the reaction. This type of accident does not present a credible hazard at Trojan as discussed in Section 2.2.3.

6.4.4.2.2 Offsite Storage and Transport

Offsite storage and transport of toxic chemicals is discussed in Section 2.2.3.

6.4.4.2.3 Toxic Gas Analysis

Table 6.4-1 gives toxicity and minimum detectable limits for the chemicals analyzed. The analysis was performed for an instantaneous release of the contents of the maximum size storage or transportation container. Since there is onsite a portable compressor for filling self-contained breathing apparatus bottles, there is in essence an unlimited supply of replacement bottles available onsite for the operations team. Therefore no accidents involving long-term releases of toxic gases were analyzed.

The analytical models used to calculate the concentration of toxic gas in the control room atmosphere in the event of a spill are consistent with those described in NUREG-0570[2]. These models include the following bases and assumptions:

(1) Consistent with the criteria of Regulatory Guide 1.78, one container of toxic chemical (tank car or cylinder) was assumed to fail, releasing all of its contents.

(2) That fraction of the chemical which would flash to a gas at atmospheric pressure is assumed to be released as a puff. The remaining chemical is assumed to spread uniformly on the ground and evaporate over time. It is assumed conservatively that no losses of chemicals occur as a result of absorption into the ground, flow into the river, cleanup actions, or chemical reactions.

(3) The initial puff due to flashing, as well as the continuous plume due to evaporation, is transported (and diluted) by the wind to the control room air intake.

(4) Atmospheric dispersion factors are calculated using the methodology of Regulatory Guide 1.78 and NUREG-0570. Dilution due to building wake effects from the Plant structures using Reference 4 is conservatively considered. The building cross-sectional area orthogonal to the wind speed is applied to account for the building wake correction factor in the atmospheric dispersion factors calculation.

(5) Concentrations in the control room as a function of time were calculated assuming normal control room ventilation.

The incapacitation models of NUREG/CR-1741[3] were used in the analysis.

Credit was taken for operators recognizing toxic gases at the minimum detectable limit and donning self-contained breathing apparatus within 2 minutes thereafter. If the results of the analysis determined that toxic levels (i.e., 2-minute limits) would be reached in the control room before operators could don breathing apparatus (i.e., <2 minutes after the minimum detectable level is reached), additional protective measures were required for that particular substance. These measures are described in Section 9.4.1.2.

The results of this analysis indicate that chlorine transportation by railroad presents a sufficient potential hazard to the control room operators to require additional protective measures.

6.4.4.2.4 Additional Toxic Gas Analysis

In addition to the chlorine detectors described in Section 6.4.4.2.1, self-contained breathing apparatus for the operations team are stored in the control room in the following quantities:

Self-contained breathing apparatus:	5
Standby bottles:	8

About 20 additional standby bottles are stored in the Maintenance tool room, and two bottles are stored at the Visitors Information Center. Assuming one extra air bottle for every three stored bottles, and a 30-minute air supply per bottle (for non-strenuous activities), this gives a maximum air supply of about 2.5 hours for a five-man operations team. Additional unlimited onsite replenishment capacity is available from a trailer-mounted air bottle charger, which can be moved to an uncontaminated location if required.

(11) Whole body beta and gamma doses are based on immersion in a semi-infinite cloud. The breathing rate assumed for calculation of the inhalation thyroid dose is 3.47 x 10^{-4} m^3/sec for the 8-hour period. The dose model for both is given in Section 15.0.11.

(12) The steam releases to the atmosphere for the steam line break are given in Table 15.1-4.

Dose results. The above assumptions have been used in conjunction with the EMERALD[6] computer code to determine the potential doses resulting from a postulated steam line break. The whole body (beta plus gamma) and thyroid doses at the site boundary and the low population zone are given in Table 15.1-5. Figure 15.1-20 shows the whole body (gamma plus beta) doses as a function of distance and time after the accident. The resulting thyroid inhalation doses as a function of distance and time are shown in Figure 15.1-21. The doses resulting from this postulated accident are well within the siting guideline doses of 10 CFR 100 (25-rem whole body and 300-rem thyroid).

Doses to control room personnel. Radiation doses to control room personnel following a postulated main steam line break accident are based on the same shielding, ventilation, building wake dilution and dose model assumptions used for the LOCA. Activity in the control room is based on the 8-hour release of activity following an accident. The steam line break accident with the most severe radiological consequences will occur outside the Containment, hence there will be no buildup of activity in the Containment and no dose due to Containment shine. The control room personnel gamma whol body, beta skin and inhalation thyroid doses were estimated using the methods discussed in Murphy/Campe and NUREG/CR-5055 (References 33 and 35, respectively of Section 15.6). The analyzed doses with the control room emergency ventilation system operable are 0.0012 rem for gamma whole body, 0.027 rem for beta skin and 0.365 rem for inhalation thyroid doses. All these doses are below the dose limits specified in General Design Criterion 19.

The estimated doses with the control room isolated and with no charcoal filtration are 0.0026 rem for gamma whole body, 0.074 rem for beta skin and 43.8 rem for inhalation thyroid doses. The inhalation thyroid dose is higher than the dose limit specified in General Design Criteria 19. These estimated doses with the control room isolated are provided to emphasize that CB-1 must be operable when the conditions exist for a main steam line break accident.

Filter loading. No recirculating or single pass filtration systems are used to clean up or control the fission products released by a main steam line rupture. Fission product iodine loadings for the control room emergency charcoal filters have been evaluated for the more severe conditions postulated in Section 15.6.5.6.1 and have been found to be negligible when compared to the design capacity of the system. (The control room emergency filter system is described in Section 6.4.).

15.1.5.2 Rupture of the Main Steam Line with Subsequent Control Rod Failure

15.1.5.2.1 Identification of Causes and Accident Description

The steam release arising from a rupture of the main steam pipe would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. With the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. Unlike the accident scenario discussed previously in Section 15.1.5.1, the automatic rod control system exhibits a consequential failure due to an adverse environment, which causes the control rods to begin stepping out prior to receipt of a reactor trip signal on overpower ΔT.

The assumptions required for this scenario are:

(1) Break occurs inside Containment between the steam generator nozzle and the Containment penetration.

(2) Intermediate steam line break (0.1 to 0.25 square feet per loop) at power levels from 70 to 100 percent.

(3) Adverse environment from the break impacts the Nuclear Instrumentation System (NIS) equipment (i.e., excore neutron detectors, cabling connectors, etc.) prior to reactor trip (i.e., within 2 minutes).

(4) Due to the adverse environment, the NIS initiates a spurious low-power signal without causing a reactor trip on negative flux rate.

Functions listed in Section 15.1.5.1.1 provides protection against a steam pipe rupture. This accident is considered a Condition IV event.

15.1.5.2.2 Analysis of Effects

Several factors tend to decrease the possibility of a significant consequential malfunction of the automatic rod control system due to a steam line break inside Containment.

(1) The four excore detectors are located in the reactor cavity at some distance from the main steam lines and from each other.

(2) The RPS includes features that protect against inappropriate rod withdrawal. A two out of four high flux trip logic is used. Therefore, three of the four excore detector outputs would have to fail low to preclude generating a normal high flux trip if the rod control system begins rod withdrawal. A rapid decrease in any two detector signals generates a negative rate trip; such a condition might result from an environmentally-induced failure of the detectors or cabling. Also, an overpower signal from any one excore detector blocks automatic rod withdrawal. These features of the RPS would tend to terminate inappropriate automatic rod withdrawal

resulting from consequential failures of the automatic rod control system.

(3) The automatic rod control system itself develops the automatic rod withdrawal logic based on auctioneered excore detector signals. That is, rod withdrawal is based on the highest of the four excore detector signals; therefore, all four detectors would have a fail low to lead to a spurious rod withdrawal.

For these reasons it is unlikely that, prior to reaching a normal high flux trip setpoint on at least two of the four excore detectors (or reaching another backup trip, such as high negative rate or high Containment pressure), environmental conditions could cause three or four of the indicator power outputs to fail low. Therefore, it is just as unlikely that the automatic rod withdrawal would have a chance to occur.

This accident is classified as an ANS Condition IV event as defined in Section 15.0.1.4.

15.1.5.2.3 Conclusions

Consistent with the assumptions made in Section 15.1.5.1, a typical bounding analysis of the intermediate steam line rupture accident has been performed by W to calculate the extent of fuel damage that could occur due to rod control system withdrawal prior to reactor trip. Based on the reduction in radial peaking factor with burnup and conservative EOL physics parameters, no fuel damage was calculated to occur following the intermediate steam line rupture with the consequential rod control system malfunction.

15.1.5.3 Minor Secondary System Pipe Breaks

15.1.5.3.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6-inch diameter break or smaller. This accident is considered an ANS Condition III event. See Section 15.0.1.3 for a discussion of Condition III events.

15.1.5.3.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Section 15.1.5 for a major secondary system pipe rupture also meet this criteria, separate analysis for minor secondary system pipe breaks is not required.

The analysis of the more probable accidental opening of a secondary system steam dump, relief or safety valve is presented in Section 15.1.4. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.1.5.3.3 Conclusions

The analysis presented in Section 15.1.5 demonstrate that the consequences of a minor secondary system pipe break are acceptable since the minimum DNBR is greater than the safety analysis limit value for a more critical major secondary system pipe break.

No radiological effects have been evaluated for this category of accident since other accidents having similar characteristics and comparable or worse consequences are evaluated (see Section 15.1.5.1.5 and 15.2.6.4).

15.2.6.4.3 Dose Results

The release of activity occurs only during the first 2 hr after the incident. Hence, all doses are for the first 2 hr. The calculated 2-hr site boundary inhalation thyroid dose is 0.143 rem, while at the low population zone distance the dose is 0.021 rem. For this incident the whole body beta and gamma doses are negligible. These doses are well below the siting guideline doses of 10 CFR 100 (25 rem whole body and 300 rem thyroid).

15.2.6.4.4 Doses to Control Room Personnel

Doses to control room personnel have been evaluated for a secondary side steam dump with loss of a-c power using the methods described in Murphy/Campe and NUREG/CR-5055 (References 33 and 35, respectively of Section 15.6). The results of the evaluation show the analyzed doses to control room personnel with the control room emergency ventilation system operable are 0.0008 rem for gamma whole body, 0.016 rem for beta skin and 0.475 rem for inhalation thyroid doses. The doses to control room personnel with no radiciodine protection provided (i.e., control room normal ventilation operating) are 0.0018 rem for gamma whole body, 0.039 rem for beta skin and 2.79 rem for inhalation thyroid doses. These estimated values are within the dose limits of General Design Criterion 19.

15.2.6.4.5 Filter Fission Product Loadings

No recirculating or single pass filtration systems are used to clean up or control the fission products released from the relief valves as a result of a loss of offsite power. Fission product iodine loadings for the control room emergency charcoal filters have been evaluated for the more severe conditions postulated in Section 15.6.5.6.1 and have been found to be negligible when compared to the design capacity of the system. (The control room emergency filter system is described in Section 6.4.)

15.6.3.5.2 Dose Results

The above assumptions have n used in conjunction with the EMERALD[2] computer code to determine the potential doses resulting from a postulated steam generator tube rupture. The whole body (beta plus gamma) and thyroid doses at the site boundary and the low population zone are given in Table 15.6-3. Figure 15.6-6 shows the whole body (gamma plus beta) doses as a function of distance and time after the accident. The resulting thyroid inhalation doses as a function of distance and time are shown in Figure 15.6-7. The doses resulting from this postulated accident are well within the siting guidelines doses of 10 CFR 100.

15.6.3.5.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated steam generator tube rupture are based on the same shielding, ventilation, cavity dilution, and dose model assumptions as used for the Loss-of-Coolant Accident (LOCA). Buildup of activity in the control room is based on the 8-hr release of activity following the accident. Steam is dumped directly to the atmosphere, hence there is no buildup of activity in the Containment and no dose due to Containment shine. The control room personnel gamma whole body, beta skin and inhalation thyroid doses were determined using the analytical method discussed in Murphy/Campe[33] and NUREG/CR-5055[35]. The doses determined by these analyses are 0.0097 rem for gamma whole body, 0.203 rem for beta skin, and 1.5 rem for inhalation thyroid doses with the control room emergency system operable. The analyzed doses with t control room isolated and with no charcoal filtration are 0.018 rem for gamma whole body, 0.39 rem for beta skin, and 20.9 rem for inhalation thyroid doses. All these doses are below the dose limits of General Design Criterion 19.

(2) Atmospheric dispersion and control room occupancy factors (seconds/cubic meters) for Containment leakage:

0-8 hours:	6.499 E-4
8-24 hours:	3.834 E-4
1-4 days:	1.495 E-4
4-30 days:	4.289 E-5

(3) Atmospheric dispersion and control room occupancy factors (seconds/cubic meters) for recirculation leakage:

0-8 hours:	5.866 E-4
8-24 hours:	3.473 E-4
1-4 days:	1.354 E-4
4-30 days:	3.885 E-5

(4) Control room ventilation rates:

(a) With one train of emergency ventilation (CB-1A or B):

Recirculation - 3,000 cfm
Filtered Makeup - 525 cfm
Unfiltered Inleakage - 10 cfm

(b) With two trains of emergency ventilation (CB-1A and B):

Recirculation - 6,000 cfm
Filtered Makeup - 1,050 cfm
Unfiltered Inleakage - 20 cfm

(5) Control room filter iodine efficiency = 95 percent for elemental and organic iodine; 99 percent for particulate iodine.

(6) Control room volume = 81,300 ft^3.

(7) Containment leak rate = 0.1 percent per 24 hours by air weight for the first 24 hours, and 0.05 percent per 24 hours thereafter.

(8) Total post-LOCA fluid leakage outside Containment = 1,772 cm^3 per hour.

Inhalation thyroid, gamma whole body, and beta skin doses were calculated using the Equations 15.0-9, 15.0-7, and 15.0-8, respectively, with modifications to these equations based upon the Murphy and Campe formalism. Given the values in assumption four above, the iodine protection factors and inhalation thyroid doses for one train of CB-1 operating (3,000 cfm of recirculation) and for two trains (6,000 cfm of recirculation) operating are identical. Table 15.6-16 gives the results of the analysis. The doses with the control room emergency ventilation system operable are less than the General Design Criterion 19 limits. The estimated inhalation thyroid dose with the control room isolation and with no charcoal filtration is higher than the dose limit specified in General Design Criteria 19. These doses are provided to emphasize the requirement that the control room emergency ventilation systems be operable when the conditions exist for a LOCA.

Filter Loadings:

No recirculating or single-pass filters are used for fission product cleanup and control within the Containment following a postulated LOCA. The only filter system expected to be operating under post-LOCA conditions is the control room emergency filter system. Charcoal filter iodine loadings for this system are not expected to be significant since fission product dilution will occur in the Containment wake prior to entering the control room ventilation system intake. In addition, the control room emergency filter system is located in a cubicle on the 105-foot elevation of the Control Building with a solid concrete cubicle wall of 3 feet 5 inches and a solid concrete floor of 2 feet 10 inches between the filters and personnel in the control room. For these reasons, the radiation doses to control room personnel on the 93-foot elevation of the Control Building from the radionuclides on the control room emergency filter system would be insignificant.

Activity Airborne in Containment:

Activity present in the Containment atmosphere is based on the formalism leading to Equation 15.6-2 and the iodine removal efficiency of the Containment base-borate spray system as discussed in the fission product spray removal model part of Section 15.6.5.6.1. Activities as a function of time in the Containment atmosphere are presented in Table 15.6-18.

15.6.5.6.2 Small Break LOCA

For breaks $\leq$0.5 sq ft in area, the maximum release of fission products to the Containment is restricted to those in the primary coolant. The radiological consequences are presented below.

Fission Product Release Assumptions:

The following conservative assumptions were used to evaluate the activity release from a postulated LOCA due to a small pipe break:

(1) The noble gases released to Containment atmosphere are equal to 100 percent of the equilibrium reactor coolant inventory.

(2) The iodine released to Containment is equal to 100 percent of the equilibrium reactor coolant inventory.

(3) The iodine air-water partition factor (hot water) is 10[27].

(4) The equilibrium reactor coolant inventories are based on 1-percent failed fuel.

(5) The Containment leak rate is assumed to be 0.10 percent/day for the first 24 hours and 0.05 percent/day for the period 1 to 30 days.

(6) No credit is taken for iodine removal by Containment base-borate sprays.

The equilibrium noble gas and iodine inventories present in the primary coolant for a 1-percent failed fuel condition (reference Section 11.1) are given in Table 15.6-19.

Fission Product Release Model:

The differential equations and their solutions which describe the release of activity from the Containment to the environment are presented in Section 15.6.5.6.1 for the design bases LOCA. These equations are solved by the computer code EMERALD[34] to determine activities released to the environment as a function of time and to calculate resulting doses.

Dose Calculation Model:

Fission products released due to Containment leakage are assumed to escape at ground level to the building wake and to be diulated by both wake and meteorological diffusion effects as they travel to the nearest site boundary (662 meters) and the LPZ outer boundary (4000 meters). The meteorological conditions and models used to determine the dispersion factors used in the accident analyses of this section are discussed in detail in Section 2.3. The dispersion factors themselves are repeated in tabular form in Table 15.0-9.

The whole body beta and gamma and inhalation thyroid doses were calculated assuming receptor submersion in a semi-infinite cloud of fission products. The dose equations, isotope average beta and gamma energies, curie-to-rem conversion factors, and breathing rates by time period are presented, discussed and referenced in Section 15.0.11.

These parameters and equations correspond to or are identical to the parameters and equations suggested by Safety Guide 4[26].

Dose Results:

The calculated 2 hour site boundary whole body (gamma plus beta) and inhalation thyroid doses are 1.12×10^{-4} rem and 2.06×10^{-3} rem, respectively. The 30 day whole body (gamma plus beta) and inhalation thyroid doses at the LPZ distance are 5.55×10^{-5} rem and 1.13×10^{-3} rem, respectively. These doses are well below the siting guideline doses of 10 CFR 100 (25 rem whole body and 300 rem thyroid). A detailed analysis of the more severe design basis LOCA is presented in Section 15.6.5.6.1.

Doses to Control Room Personnel:

Doses to control room personnel have been evaluated for a more severe radioactivity release (Section 15.6.5.6.1) than that postulated above and have been found to be within the dose limits of General Design Criterion 19. Since fission product activity releases for the small LOCA are several orders of magnitude less than for the design basis LOCA, the requirements of General Design Criterion 19 are met with a large margin.

Filter Fission Product Loadings:

No recirculating or single-pass filtration systems are used to cleanup or control the fission products released to the Containment as a result of a postulated LOCA. Fission product iodine loadings for the control room emergency charcoal filters have been evaluated for the more severe conditions postulated in Section 15.6.5.6.1 and have been found to be negligible when compared to the design capacity of the system. (Control room emergency filter system is described in Section 6.4.)

TABLE 15.6-15

LOSS-OF-COOLANT ACCIDENT
CONTROL ROOM PERSONNEL DOSE DUE TO RECIRCULATION WATER LEAKAGE WITH CONTROL ROOM EMERGENCY VENTILATION OPERATING

Time Period	Thyroid Dose (rem)
0-8 hours	0.45
8-24 hours	0.44
1-4 days	0.53
4-30 days	0.41

TABLE 15.6-16

CONTROL ROOM PERSONNEL DOSES FOR 30-DAY DURATION
UPDATED CALCULATION

Control Room Emergency Ventilation System Operating

Source	Gamma Whole Body Dose (Rem)	Beta Skin Dose (Rem)	Inhalation Thyroid Dose 1 Train Operating (Rem)	Inhalation Thyroid Dose 2 Trains Operating (Rem)
Internal cloud	0.35	5.0	11.2	11.2
Containment shine	0.24	-	-	-
External cloud shine	0.003	-	-	-
Total	0.60	5.0	11.2	11.2

Control Room Isolated With No Charcoal Filtration

Source	Gamma Whole Body Dose (Rem)	Beta Skin Dose (Rem)	Inhalation Thyroid Dose (Rem)
Internal Cloud	0.39	6.5	1045.0
Containment Shine	0.24	-	-
External Cloud Shine	0.003	-	-
Total	0.64	6.5	1045.0

population zone (4000 meters) are 1.47 rem and 0.215 rem, respectively. These doses are well below the siting guideline doses of 10 CFR 100. Figure 15.7-1 shows the calculated whole body (gamma plus beta) dose as a function of distance.

15.7.1.4 Dose to Control Room Personnel

In addition to the offsite dose evaluations, analyses of the doses to Plant personnel in the control room for the duration of the WGDT failure incident were performed. The analysis with CB-1 operable was based on the same control room ventilation, filtration and shielding parameters as presented in Section 15.6.5.6.1 and with the formalism of Murphy/Campe and NUREG/CR-5055 (References 33 and 35, respectively of Sections 15.6). The analysis without charcoal filtration, but with the control room able to be isolated takes no credit for the control room ventilation and filtration parameters of Section 15.6.5.6.1.

The estimated values with CB-1 operable were 0.018 rem for gamma whole body, 0.423 rem for beta skin and 0.069 rem for inhalation thyroid doses. The estimated values with no radioiodine protection provided (i.e., control room normal ventilation operating) are 0.018 rem for gamma whole body, 0.423 rem for beta skin and 0.271 rem for inhalation thyroid doses. These doses to control room personnel from the WGDT failure incident are below the dose requirements of General Design Criterion 19.

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

Analysis of this accident was not required as a part of the Trojan Nuclear Plant design basis.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID TANK FAILURES

15.7.3.1 Identification of Causes and Accident Description

This accident is defined as the uncontrolled release of the contents of a Chemical and Volume Control System (CVCS) holdup tank due to the postulated rupture of the tank. This tank is the highest potential atmospheric release source in the CVCS due to its large volume and the fact that it is assumed to contain reactor coolant.

15.7.3.2 Assumptions or Conditions

An analysis of the consequences of a holdup tank rupture accident was performed in accordance with the requirements of Regulatory Guide 1.29[2] for the purpose of determining whether the holdup tanks should be classified as Seismic Category I. The assumptions used in this analysis are:

(1) The reactor has been operating at full power with 1-percent defective fuel. Reactor coolant noble gas and iodine inventories are given in Table 15.6-20.

(2) A CVCS mixed bed demineralizer decontamination factor of 10.

(3) No degassification.

(4) Holdup tank volume = 60,000 gallons; contents released over a 2-hour period.

(5) Iodine partition factor = 0.0075.

(6) CVCS liquid temperature = 120°F.

15.7.3.3 Dose Results

The fission products were assumed to be released from the Auxiliary Building at ground level and mixed in the building wake. The 2-hour atmospheric diffusion factors (cloud centerline concentration) for a ground-level release are presented in Table 15.0-9. Whole body gamma doses and inhalation thyroid doses were computed on the basis of submersion in a semi-infinite for the duration of the incident cloud passage. A breathing rate of 2.3×10^{-4} m^3/sec and dose conversion factors from Regulatory Guide 1.109 were used.

The 2-hour whole body and thyroid doses at the nearest site boundary (662 meters) resulting from the rupture of a holdup tank are 0.325 rem and 0.122 rem, respectively. Ruptures in other components of the CVCS, including the volume control tank (VCT) and CVCS piping, were also analyzed, and the resulting site boundary doses were found to be less than for the holdup tank rupture accident.

15.7.4 DESIGN BASIS FUEL HANDLING ACCIDENTS

15.7.4.1 Identification of Causes and Accident Description

The possibility of a fuel handling accident is very remote because of the many administrative controls and physical limitations imposed on the fuel handling operations (see Section 9.1). All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in handling fuel and nuclear safety. When transferring irradiated fuel from the core to the spent fuel pool (SFP) for storage, the reactor cavity and refueling canal are filled with borated water at a boron concentration equal to that in the SFP which ensures subcritical conditions in the core even if all rod cluster control assemblies (RCCA) were withdrawn. After the reactor head and rod cluster control drive shafts are removed, fuel assemblies are lifted from the core, transferred vertically to the refueling cavity, placed horizontally on a conveyor car and pulled through the transfer tube and canal, upended and transferred through the pool gate, then lowered into stainless steel racks for storage in the SFP in a pattern which prevents any possibility of a criticality accident. Fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position that provides an adequate water depth shield for radiation protection of operating personnel.

The Containment, Auxiliary Building, refueling cavity, canal, and SFP are designed Seismic Category I which prevents the structures themselves from failing in the event of an earthquake. They are also designed to prevent any credible external missile from entering the buildings and reaching the stored irradiated fuel, and any internal missile from penetrating the walls of these structures. The fuel-handling manipulators, cranes, trollies, bridges, and associated equipment above the water cavities through which the fuel assemblies move are designed to prevent this equipment from generating missiles and damaging the fuel. The construction of the fuel assemblies precludes damage to the fuel should portable or hand tools drop on an assembly.

The two times a fuel handling accident could occur are during transfer of a fuel assembly from the core to its storage position in the SFP or from its storage position in the SFP into a fuel cask. The facility is designed so that heavy objects such as the fuel cask cannot be carried over the irradiated fuel stored in the SFP and only one fuel assembly can be handled at a time. Movement of equipment handling the fuel is kept at low speeds while exercising caution that the fuel does not strike another object or structure during transfer from the core to its storage position. In the unlikely occurrence that an assembly became stuck in the transfer tube, natural convection will maintain adequate cooling.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 60 pounds on each fuel rod. If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling is not of a magnitude great enough to breach the fuel rod cladding.

If the fuel rods are not in contact with the bottom plate of the assembly, the rods would have to slide against the 60-pound friction force. This would have the effect of absorbing a shock and thus limit the force on the individual fuel rods. After the reactor is shut down, the fuel rods contract during the subsequent cooldown and would not be in contact with the bottom plate of the assembly.

Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, it is felt that it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breach the integrity of the cladding.

If, during handling, the fuel assembly strikes against a flat surface, the loads would be distributed across the fuel assemblies and grid clips and essentially no dam[illegible] would be expected in any fuel rods. If the fuel assembly were to strike a sharp object, it is possible that the sharp object might damage the fuel rods with which it cor[illegible] in contact, but breaching of the cladding is not expected.

Analyses have been made assuming the extremely remote situation where a fuel assembly is dropped on another, and where one assembly strikes a sharp object. The analysis of a fuel assembly assumed to be dropped and striking a flat surface considered the stresses the fuel cladding was subjected to and any possible buckling of the fuel rods between the grip clip supports.

The results showed that the buckling load at the bottom section of the fuel rod, which would receive the highest loadings, was below the critical buckling load and the stresses were relatively low and below the yield stress. For the case where one assembly is dropped on top of another fuel assembly, the loads will be transmitted through the end plates and the RCCA guide tubes of the struck assembly before any of the loads reach the fuel rods

The end plates and guide thimbles absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. Also, energy is absorbed in the struck assembly top end plate before any load can be transmitted to the fuel rods. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling loads and the stresses in the cladding were relatively low and below yield.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the fact that no fuel cladding integrity failures are expected to occur during fuel handling operations. However, the above analysis indicates that if the unlikely event of a fuel accident could occur, it would result from the dropping of a fuel assembly either in the Containment or Fuel Building.

15.7.4.1.1 Containment Accident

During the fuel handling operations, the Containment is kept in an isolated condition with all penetrations to the outside atmosphere either closed or capable of being closed on an alarm signal from a radiation monitor indicating that radioactivity is above prescribed limits. In addition, the Containment purge system is not operated when irradiated fuel is being moved inside Containment during the first 285 hours following reactor shutdown.

At least one of the two interlock doors on the personnel locks is kept closed. In addition to the area radiation monitors located in the Containment, portable monitors capable of sounding audible alarms are to be located in the fuel handling area. ' a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors would sound an audible alarm, and the personnel would be evacuated. The Containment air cleanup filter system can be used to remove any radioactive iodine released.

15.7.4.1.2 Fuel Building Accident

In the Fuel Building a fuel assembly could be dropped in the transfer canal or the SFP. However, supply air for the SFP area is swept across the fuel pool and transfer canal and exhausted through the vent. In addition to the area radiation monitor located on the bridge over the SFP, portable radiation monitors capable of emitting audible alarms are located in this area during fuel handling operations. Doors to the outside atmosphere in the Auxiliary and Fuel Buildings are closed to maintain controlled leakage characteristics in the SFP region during refueling operations involving irradiated fuel. Should a fuel assembly be dropped in the canal or in the pool and release radioactivity above a prescribed level, the radiation monitors will sound an alarm and the SFP ventilation exhaust through charcoal filters will remove most of the halogens prior to discharging any effluent to the atmosphere.

If the discharge vent radiation monitor indicates that the radioactivity in the vent discharge is greater than the prescribed levels, an alarm sounds and the supply and exhaust ventilation systems servicing the SFP area can be shut down limiting the leakage to the atmosphere.

Any movement of the fuel cask in the SFP area is under administrative control. Interlocks prevent the crane from moving the cask over stored irradiated fuel and limit cask movement (see Section 9.1.2.3).

The probability of a fuel handling accident is very low because of the safety features, administrative controls and design characteristics of the facility as previously mentioned. The shock absorbing analyses presented above indicate that in most incidents where an assembly is struck against another object, the outer row of fuel rods would experience greater loads and stresses than the inner rows. Therefore, if a fuel assembly is dropped it does not necessarily mean that all the fuel rods break. Nevertheless, for a fuel handling accident analysis the assumption is made that the cladding of all the fuel rods in one fuel assembly break, suddenly releasing all the gaseous fission products in the voids between the pellets.

(3) The maximum centerline operating fuel temperature for the highest power assembly is <4500°F.

(4) The average region burnup for the peak assembly is <25,000 MWd/MTU.

(5) The water depth between the top of the damaged fuel rods and the SFP surface is >23 feet.

(6) All fuel rods in one assembly (264) are assumed to be damaged as a result of the handling accident.

(7) All of the gap activity in the damaged rods is released to the pool water and consists of 10 percent of the total noble gases other than Kr-85, 30 percent of the Kr-85 and 10 percent of the total radioactive iodine in the rods.

(8) Assembly fission product inventories are based on full-power operation at the end of core life immediately preceding shutdown (see Table 15.0-5), a radial peaking factor of 1.65.

(9) The iodine gap inventory is composed of 0.25 percent organic and 99.75 percent inorganic species.

(10) The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100. An experimental test program was conducted to evaluate the extent of removal of iodine released from a damaged irradiated fuel assembly.

Iodine removal from the released gas takes place as the gas rises through the body of solution in the SFP to the pool surface. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution.

In order to obtain all the necessary information regarding this mass transfer process, a number of small-scale tests were conducted, using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry) and data was collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression for iodine decontamination factor in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large-scale tests were also performed with carbon dioxide. The small-scale carbon dioxide tests also resulted in a mathematical expression for the decontamination factor in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full-size fuel assembly simulator was fabricated and placed in a deep pool for testing, where gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas, and overall decontamination factors were measured as a function of the total gas volume released. These measurements, combined with the analytical expression derived from small-scale tests with carbon dioxide, permitted an in situ measurement of both the effective bubble diameter and rise time, both as a function of the volume of gas released.

Having measured the characteristics of large-scale gas releases, the decontamination factor for iodine was obtained, using the analytical expression from small-scale iodine testing.

$$\text{Decontamination factor} = 7.3e^{0.313\ t/d}$$

where

t = rise time

d = effective bubble diameter.

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the SFP solution and that the efficiency of removal will depend on the volume of gas released instantaneously from the full void space.

The pool decontamination factor for iodine is indicated to be a minimum of 760 for a 26-foot SFP depth. For conservatism, a lower decontamination factor of 100 has been selected to provide for reasonable deviation in the factors which control iodine adsorption by the SFP water.

(11) Noble gas decontamination factor is 1.0 for the SFP.

(12) All radioactive material that escapes from the SFP to the building is released from the building over a 2-hour time period through the SFP and Auxiliary Building ventilation systems.

(13) Filter efficiencies of 90 percent for inorganic species and 70 percent for organic species are assumed for the

2-inch tray-type charcoal absorbers on the building exhaust.

(14) The effluent from the filter system passes directly to the exhaust system without mixing in the building atmosphere.

The Regulatory Guide 1.25 fission product source and release assumptions (listed above) assume that the average burnup for the peak assembly does not exceed 25,000 MWd/MTU. An evaluation was performed to determine if the Regulatory Guide assumptions are valid for assembly average burnups up to 44,000 MWd/MTU. Based on an evaluation of extended burnup fuel by Westinghouse(4) and the results of measurements of fission gas gap fractions on operating reactors(5), it was concluded that the Regulatory Guide 1.25 assumptions for fraction of fuel rod activity in the fuel rod gap and spent fuel pool iodine decontamination factor are still conservative in assembly average burnups up to 44,000 MWd/MTU. The only possible difference in the radiological consequences from increasing burnup would come from the change in fuel rod radionuclide inventories.

15.7.4.2.1.2 Dose results. The fission products were assumed to be released from the Fuel Building at ground level and mixed in the building wake. The 2-hour atmospheric diffusion factors (cloud centerline concentration) for a ground-level release are presented in Table 15.0-9. Whole body beta and gamma doses and inhalation thyroid doses were computed on the basis of submersion in a semi-infinite cloud for the duration of the incident cloud passage. A breathing rate of 3.47×10^{-4} cubic meters per second was used. The above assumptions were used in conjunction with the EMERALD computer code(6) to determine potential doses from a fuel handling accident. The decay constants, dose conversion factors, average gamma and beta energies for significant isotopes and dose models are given in Section 15.0.

The 2-hour whole body (beta plus gamma) dose at the 662 meter site boundary is 2.03 rad equivalent man (rem). At the 4000 meter low population zone distance it is 0.296 rem. The inhalation thyroid dose for the first 2 hours at the site boundary is 18.1 rem, while at the low population zone distance it is 2.6 rem. Without credit for the 90-percent efficiency of the Fuel Building charcoal filters, the thyroid doses would be increased by a factor of 10. Even then, these doses are well below the siting guidelines doses of 10 CFR 100 (25 rem whole body and 300 rem thyroid for a 2-hour duration at the site boundary). These results indicate that if no credit is taken for the Fuel Building ventilation exhaust filters, the resulting doses will still be below the siting guidelines doses of 10 CFR 100.

An evaluation was performed to determine the effects on radiological consequences of increasing the assembly average burnup to 44,000 MWd/MTU. As discussed in Section 15.7.4.2.1.1, the only change in the radiological consequences would be from differences in the fuel rod radionuclide inventories. Table 15.7-2 shows the percentage increase in key fuel rod noble gas and iodine isotope activities calculated using the ORIGEN2[7] computer code for an assembly average burnup increase from 25,000 to 44,000 MWd/MTU. Calculated increases in 2-hour whole body and inhalation thyroid doses for an assembly average burnup of 44,000 MWd/MTU were approximately 3 percent, which is considered to be negligible given the overall accuracy of the calculation.

15.7.4.2.1.3 Dose to control room personnel. In addition to the offsite dose evaluation, analyses of the doses to Plant personnel in the control room for the duration of the fuel handling incident were performed. The analysis with control room emergency ventilation operable was based on the same control room ventilation, filtration, and shielding parameters and the Murphy/Campe and NUREG/CR-5055 (References 33 and 35, respectively from Section 15.6) formalism as presented in Section 15.6.5.6.1. The analysis without control room emergency ventilation charcoal filtration, but with the control room able to be isolated takes no credit for the control room ventilation and filtration parameters of Section 15.6.5.6.1.

The estimated values for a Fuel Building fuel handling accident with control room emergency ventilation operable are 0.016 rem for gamma whole body, 0.497 rem for beta skin and 0.422 rem for inhalation thyroid doses. The estimated values without charcoal filtration, but with the control room able to be isolated are 0.006 rem for gamma whole body, 0.172 rem for beta skin and 5.28 rem for inhalation thyroid doses. These doses are below the dose requirements of General Design Criterion 19.

15.7.4.2.2 Containment Accident

An analysis of the activity releases from a fuel handling accident inside Containment was performed based on the fission product source and release assumptions of Regulatory Guide 1.25. The assumptions and conditions used in the analysis were the same as those used for the Fuel Building accident (see Section 15.7.4.2.1).

The resulting 2-hour whole body (beta plus gamma) dose at the 662-meter site boundary was 2.0 rem. The inhalation thyroid dose at the site boundary was 120 rem. These doses are less than the 10 CFR 100 limits.

The NRC staff performed a separate analysis of the fuel handling accident in Containment, using the Regulatory Guide 1.25 models and assumptions. However, the NRC staff used a χ/Q of 6.9×10^{-4} sec/m^3, instead of the value of 4.26×10^{-4} sec/m^3 used by Portland General Electric Company. The resulting site boundary inhalation thyroid doses were concluded to be acceptably low (100 rem) only if a minimum of 285 hours of decay following reactor shutdown was assumed. Therefore, a change to the Plant Technical Specifications was made[(8)] to require isolation of all Containment vent and purge lines during refueling until 285 hours after reactor shutdown.

In addition to the offsite dose evaluation, analyses of the doses to Plant personnel in the control room were performed for the duration of the Containment fuel handling incident. The analysis with control room emergency ventilation operable was based on the same control room ventilation filtration and shielding parameters and the Murphy/Campe and NUREG/CR-5055 (References 33 and 35, respectively of Section 15.6) formalism presented in Section 15.6.5.6.1. The estimated values are 0.006 rem for gamma whole body, 0.203 rem for beta skin and 1.38 rem inhalation thyroid doses. The analysis without charcoal filtration, but with the control room able to be isolated takes no credit for the control room ventilation or filtration parameters of Section 15.6.5.6.1. The estimated values are 0.002 rem for gamma whole body, 0.075 rem for beta skin and 20.5 rem for inhalation thyroid doses. These doses are below the dose requirements of General Design Criterion 19.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

Analysis of this accident was not required as part of the Trojan Nuclear Plant design basis.

23. Information on the long-ter 'ld-up of boric acid in the core region following a postulate ..., transmitted from C. L. Caso, Westinghouse NES, to T. M. No' ., NRC, as enclosures to Letter CLC-NS-309 (April 1, 1975)

24. Supplemental information on the long-term build-up of boric acid in the core region following a postulated LOCA transmitted from J. O. Cermak, Westinghouse NES, to T. M. Novak, NRC, as Letter JOC-NS-364 (July 23, 1975).

25. Letter from D. B. Vassallo, NRC, to C. Eicheldinger, Westinghouse NES (May 30, 1975).

26. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors, Regulatory Guide 1.4, Directorate of Regulatory Standards, U. S. Atomic Energy Commission (June 1973).

27. M. A. Styrikovich and O. I. Martynova, "Transfer of Iodine from Aqueous Solutions to Saturated Vapor", Atomniya Energiya, 17, 1, (July 1964) pp 45-49.

28. L. P. Parsly, Design Considerations of Reactor Containment Spray Systems - Part IV. Calculation of Iodine - Water Partition Coefficients, ORNL TM-2412, January 1970.

29. D. H. Slade, Meteorology and Atomic Energy, TID-24190, U. S. Atomic Energy Commission (1972).

30. D. S. Duncan and A. B. Speir, GRACE II - A Program for Computing Gamma - Ray Attenuation and Heating in Cylindrical and Spherical Geometrics, Atomics International (1959).

31. D. S. Duncan and A. B. Speir, GRACE I - A Program Designed for Computing Gamma - Ray Attenuation and Heating in Reactor Shields, Atomics International (1959).

32. USNRC Standard Review Plan for the Review of Safety Analysis Reports, NUREG-0800, Section 6.4, July 1981.

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

34. W. K. Brunot, EMERALD - A Program for the Calculation of Activity Releases and Potential Doses from a Pressurized Water Reactor Plant, Pacific Gas & Electric Company (October 1971).

35. NUREG/CR-5055, Atmospheric Diffusion for Control Room Habitability Assessments (May 1988).