Union Electric

Rev. 5

CALLAWAY PLANT
OFFSITE DOSE CALCULATION MANUAL

February, 1988

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OFFSITE DOSE CALCULATION MANUAL

February, 1988

Approved: Blosser 2/29/88 — Chairman, Onsite Review Committee

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This document contains the following:

Pages	1	through	105
Tables	1	through	12
Figures	4.1, 5.1A, 5.1B, 5.2A, 5.2B, 5.3		

Table of Contents

Table of Contents (continued)

Table of Contents (continued)

Table of Contents (continued)

List of Figures

List of Tables

Record of Revisions

Revision Number	Date	Reason for Revision
Rev. 0	March 1983	
Rev. 1	November 1983	Revised to support the current RETS submittal and to incorporate NRC Staff comments
Rev. 2	March 1984	Revised to incorporate NRC Staff comments
Rev. 3	June 1985	Revised to incorporate errata identified by ULNRC-803 and changes to the Environmental Monitoring Program. Incorporate results of 1984 Land use Census.
Rev. 4	February 1987	Minor clarifications, incorporated 31-day projected dose methodology. Change in the utilization of areas within the Site Boundary.
Rev. 5	January 1988	Minor clarifications, revised descriptions of liquid and gaseous rad monitors, revised liquid setpoint methodology to incorporate monitor background, revised dose calculations for 40CFR190 requirements, Revised Table 6 and Figures 5.1A and 5.1B to refine descriptions of environmental TLD stations, incorporated description of environmental [illegible] testing required by Reg. Guide 4.13, revised Tables 1, 2, 4, and 5 to add additional nuclides, deleted redundant material from Chapter 6.

1.0 PURPOSE AND SCOPE

The Offsite Dose Calculation Manual (ODCM) describes the methodology and parameters used in the calculation of offsite doses and dose rates due to radioactive liquid and gaseous effluents and in the calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints. The ODCM also contains a list and description of the specific sample locations for the radiological environmental monitoring program.

Changes in the calculational methodologies or parameters will be incorporated into the ODCM and documented in the Semi Annual Radioactive Effluent Release Report. The ODCM does not replace any station implementing procedures.

2.0 LIQUID EFFLUENTS

2.1 Technical Specification 3.3.3.9

The radioactive liquid effluent monitoring instrumentation channels shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Technical Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be adjusted to the values determined in accordance with the methodology and parameters in the ODCM.

2.2 Liquid Effluent Monitors

Gross radioactivity monitors which provide for automatic termination of liquid effluent releases are present on the liquid effluent lines. Flow rate measurement devices are present on the liquid effluent lines and the discharge line (cooling tower blowdown). Setpoints, precautions, and limitations applicable to the operation of the Callaway Plant liquid effluent monitors are provided in the appropriate Plant Procedures, which are contained in Volume 6 of the Plant Operating Manual. Setpoint values are calculated to assure that alarm and trip actions occur prior to exceeding the Maximum Permissible Concentration (MPC) limits in 10 CFR Part 20 at the release point to the UNRESTRICTED AREA. The calculated alarm and trip action setpoints for the liquid effluent line monitors and flow measuring devices must satisfy the following equation:

$$\frac{cf}{F + f} \leq C \qquad (2.1)$$

Where:

C = the liquid effluent concentration limit (MPC) implementing Technical Specification 3.11.1.1 for the site in (μCi/ml).

c = The setpoint, in (μCi/ml), of the radioactivity monitor measuring the radioactivity concentration in the effluent line prior to dilution and subsequent release; the setpoint, which is inversely proportional to the volumetric flow of the effluent line and directly proportional to the volumetric flow of the dilution stream plus the effluent stream, represents a value, which, if exceeded, would result in concentrations exceeding the limits of 10 CFR Part 20 in the UNRESTRICTED AREA.

f = The flow setpoint as measured at the radiation monitor location, in volume per unit time, but in the same units as F, below.

F = The dilution water flow setpoint as measured prior to the release point, in volume per unit time. (If (F) is large compared to (f), then $F + f \approx F$).

(Ref. 9.8.1)

If no dilution is provided, then $c \leq C$.

The radioactive liquid waste stream is diluted by the plant discharge line prior to entry into the Missouri River. Normally, the dilution flow is obtained from the cooling tower blowdown, but should this become unavailable, the plant water treatment facility supplies the necessary dilution flow via a bypass line. The batch release limiting concentration (c) which corresponds to the liquid radwaste effluent line monitor setpoint is to be calculated using methodology from the expression above.

Thus, the expression for determining the setpoint on the liquid radwaste effluent line monitor becomes:

$$c \leq \frac{C(F + f)}{f} \quad (\mu Ci/ml) \tag{2.2}$$

2.2.1 Continuous Liquid Effluent Monitors

The radiation detection monitors associated with continuous liquid effluent releases are (Ref. 9.6.1, 9.6.2):

Monitor I.D.	Description
BM-RE-52	Steam Generator Blowdown Discharge Monitor
LE-RE-59	Turbine Building Drain Monitor

These effluent streams are not considered to be radioactive unless radioactivity has been detected by the associated effluent radiation monitor or by laboratory analysis. The sampling frequency, minimum analysis frequency, and type of analysis performed are as per Technical Specification Table 4.11-1.

The alarm/trip setpoints are determined through the use of Equation (2.2) methodology to ensure that the limits of Technical Specification 3.11.1.1 are not exceeded at the UNRESTRICTED AREA. The alert setpoints have been administratively established below the alarm/trip setpoints, thus providing an additional margin of safety.

The alarm/trip setpoint calculations are based on the minimum dilution flow rate (cooling tower blowdown, 5000 gpm), the maximum effluent stream flow rate, and the actual isotopic analysis. Due to the possibility of a simulataneous release from more than one release pathway, a portion of the total site release limit is allocated to each pathway. The determination and usage of the allocation factor is discussed in Section 2.3.1. In the event the alarm/trip setpoint is reached, the radiation monitor setpoint (c), will be reevaluated using the actual dilution flow rate (F), the actual effluent stream flow rate (f), and the actual isotopic analysis. This evaluation will then be used to ensure that Radiological Effluent Technical Specification 3.11.1.1 limits were not exceeded.

2.2.2 Radioactive Liquid Batch Release Effluent Monitor

The two radiation monitors which are associated with the liquid effluent batch release systems are (Ref. 9.6.4, 9.6.5):

MONITOR I.D.	Description
HB-RE-18	Liquid Radwaste Discharge Monitor
HF-RE-45	Secondary Liquid Waste System Monitor

The setpoint for these monitors is determined according to the methodology described by Equation (2.2) and is a function of the dilution flow rate (F), the radioactive effluent line flow rate (f) and the tank liquid effluent concentration, as determined by a pre-release isotopic analysis. Based on these factors, a setpoint is calculated for the appropriate monitor to ensure that the limits of Technical Specification 3.11.1.1 are not exceeded at the UNRESTRICTED AREA (Figure 5.2A).

2.3 Determination of Liquid Effluent Monitor Setpoints

The dependence of the setpoint (c), on the radionuclide distribution, yields, calibration, and monitor parameters, requires that several variables be considered in setpoint calculations. (Ref. 9.8.1)

The isotopic concentration of the release being considered must be determined. This is obtained from the sum of the measured concentrations as determined by the analysis required per Technical Specifications Table 4.11-1:

$$C_T = (\sum_i (C_g)_i) + C_a + C_S + C_t + C_F \qquad (2.3)$$

Where:

C_T = the total concentration of radionuclides as determined by the analysis of the waste sample.

$\sum_i (C_g)_i$ = the sum of the concentrations (C_g) of each measured gamma emitting nuclide observed by gamma-ray spectroscopy of the waste sample.

C_a* = the measured concentrations (C_a) of alpha emitting nuclides observed by gross alpha analysis.

C_s* = the measured concentrations of Sr-89 and Sr-90 in liquid waste as determined by analysis of the quarterly composite sample.

C_t* = the measured concentration of H-3 i liquid waste.

$C_F{}^*$ = the measured concentration of Fe-55 in liquid waste as determined by analysis of the quarterly composite sample.

The C_g term is included in the analysis of each batch; terms for alpha, Sr-89, Sr-90, Fe-55, and tritium are included as appropriate.

*Values for these concentrations will be based on previous composite sample analyses as required by Table 4.11-1 of Technical Specifications.

The measured radionuclide concentrations are used to calculate a Dilution Factor (F_d), which is the ratio of total dilution flow rate to tank flow rate required to assure that the limiting concentrations of Technical Specification 3.11.1.1 are met at the point of discharge. This is referred to as the required Dilution Factor and is determined according to:

$$F_d = \sum_i \left\{ \frac{(C_g)_i}{(MPC_g)_i} + \frac{C_a}{MPC_a} + \frac{C_s}{MPC_s} + \frac{C_t}{MPC_t} + \frac{C_F}{MPC_F} \right\} \div F_S \qquad (2.4)$$

Where:

C_g, C_a, C_s, C_t, C_F = measured concentrations as defined in 2.3.1.1. Terms C_a, C_s, C_F, and C_t will be included in the calculation as appropriate.

MPC_g, MPC_s, MPC_a, MPC_F, MPC_t = are limiting concentrations of the appropriate radionuclides from 10CFR 20, Appendix B, Table II, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} µCi/ml total activity.

F_s = the safety factor; a conservative factor used to compensate for statistical fluctuations and errors of measurements. (For example, F_S = 0.5 corresponds to a 100 percent variation.) Default value is F_S = 0.9.

For the case $F_d \leq 1$, the monitor tank effluent concentration meets the limits of Radiological Effluent Technical Specification 3.11.1.1 without dilution and the effluent may be released at any desired flow rate. If $F_d > 1$ then dilution is required to ensure compliance with Technical Specification 3.11.1.1 concentration limits. If simultaneous releases are occuring or are anticipated, a modified dilution factor (F_{dn}), must be determined so that available dilution flow may be apportioned among simultaneous discharge pathways.

$$F_{dn} = F_d \div F_a \qquad (2.5)$$

Where:

F_a = the allocation factor which will modify the required dilution factor such that simultaneous liquid releases may be made without exceeding the limits of Technical Specification 3.11.1.1.

The most straight-forward determination of the allocation factor is:

$$F_a = \frac{1}{n} \qquad (2.6)$$

Where:

n = the number of liquid discharge pathways for which $F_d > 1$ and which are planned for simultaneous release.

However, this value for F_a may be unnecessarily restrictive in that all release pathways are apportioned the same fraction of the available dilution stream, regardless of the relative concentrations of each of the sources.

Since the radionuclide concentration of the two continuous sources is less than that of the batch release source, it is acceptable to allocate smaller portions of the dilution stream to the continuous releases and a larger portion to the batch releases.

Therefore, F_a is necessarily defined as a flexible quantity with a default value of $1/\eta$, however, the sum of the allocation factors assigned to pathways for the simultaneous release must be ≤ 1.

The calculated maximum permissible waste tank effluent flow rate, (f_{max}), is based on the modified dilution factor, (F_{dn}), and the effective dilution flow rate, (F_{eff}). The effective dilution flow rate is given by:

$$F_{eff} = (0.9)F_e \qquad (2.7)$$

Where:

F_e = the cooling tower blowdown flow rate and/or bypass dilution flow.

A conservative value for F_e is the minimum allowable cooling tower blowdown of 5000gpm, which is used as a default value.

Having established the values of F_{dn} and F_{eff}, the calculated maximum permissible waste tank flow rate can be calculated by:

$$f_{max} \leq \frac{F_{eff} + f_p}{F_{dn}} \approx \frac{F_{eff}}{F_{dn}} \qquad (\text{for } f_p << F_{eff}) \qquad (2.8)$$

Where:

f_p = the expected undiluted effluent flow rate.

Thus, the effluent flow rate is set at or below f_{max}. Even though the value of f_{max} may be larger than the actual effluent pump capacity, (f_p), it does represent the upper limit to the effluent flow rate whereby the requirements of Technical Specification 3.11.1.1 may still be met. If $F_d \leq 1$, the effluent flow rate setpoint may be assigned any value since the waste tank effluent concentration meets the limits of Technical Specification 3.11.1.1 without dilution and the release may be made without regard to the setpoints for other release pathways. For those discharge pathways selected to be secured during the release under consideration, the flow rate setpoint should be set at as low a value as practicable to detect any inadvertent release.

The liquid radiation monitor setpoint may now be determined based on the values of C_T, and f_{max}, which were specified to provide compliance with the limits of Technical Specification 3.11.1.1.

The monitor response is primarily to gamma radiation, therefore, the actual setpoint is based on $\sum_i (C_g)_i$. The calculated monitor setpoint concentration is determined as follows:

$$c = B + A \sum_i (C_g)_i \frac{\mu Ci}{ml} \quad (2.9)$$

Where:

c = setpoint as previously defined (see Section 2.2)

B = monitor background prior to release initiation (μCi/ml).

NOTE

The monitor background is controlled at an appropriately small fraction of the limiting MPC of gamma emitting radionuclides to ensure that adequate monitor sensitivity is maintained.

A = Adjustment factor which will allow the setpoint to be established in a practical manner for convenience and to prevent spurious alarms.

$$A = \frac{f_{max}}{f_p} \quad (2.10)$$

If A > 1 Calculate the monitor setpont and proceed with the release.

If A ≤ 1: No release may be made. This condition must be flagged and the operator instructed to re-evaluate F_{dn} and F_{eff} (i.e., reduce effluent flow rate or return radwaste for reprocessing).

NOTE

If f_d < 1, the release may be made without regard to available dilution or simultaneous releases, and the Adjustment Factor may be calculated as follows:

$$A = \frac{1}{F_d} \qquad (2.11)$$

The methodology described above is used to determine setpoints for each of the radiation monitors assigned an effluent monitoring function. The limiting release concentration can be increased by reducing the discharge flow-rate and by increasing the cooling tower blowdown flow-rate.

2.4 Liquid Effluent Concentration Measurements

2.4.1 Technical Specification 3.11.1.1

The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2.0 E-04 uCi/ml total activity.

2.4.2 Liquid Effluent Concentration Measurements

Liquid batch releases are discharged as a discrete volume and each release is authorized based upon the sample analysis and the dilution flow rate existing in the discharge line at time of release. To assure representative sampling, each liquid monitor tank is isolated and thoroughly mixed by recirculation of tank contents prior to sample collection. The methods for mixing, sampling, and analyzing each batch are outlined in applicable plant procedures. The allowable release rate limit is calculated for each batch based upon the pre-release analysis, dilution flow-rate, and other procedural conditions, prior to authorization for release. The radwaste liquid effluent discharge is monitored prior to entering the dilution discharge line and will automatically be terminated if the preselected alarm/trip setpoint is exceeded. Concentrations are determined primarily from the gamma isotopic, H-3, & gross alpha analyses of the liquid batch sample. For Sr-89, Sr-90, & Fe-55, the measured concentration from the previous composite analysis is used. Composite samples are collected for each batch release and quarterly analyses are performed in accordance with Technical Specification Table 4.11-1.

Doses from liquids discharged as continuous releases are calculated by utilizing the last measured values of samples required in accordance with Technical Specifications Table 4.11-1.

2.5 Dose Due to Liquid Effluents

2.5.1 Technical Specification 3.11.1.2

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, to UNRESTRICTED AREAS shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the whole body and less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

2.5.2 The Maximum Exposed Individual

The cumulative dose determination considers the dose contributions from the maximum exposed individual's consumption of fish and potable water, as appropriate. Normally, the adult is considered to be the maximum exposed individual. (Ref. 9.8.3, 9.8.4)

The Callaway Plant's liquid effluents are discharged to the Missouri River. As there are no potable water intakes within 50 miles of the discharge point (Ref. 9.7.1, 9.6.6), this pathway does not require routine evaluation. Therefore, the dose contribution from fish consumption is expected to account for more than 95% of the total man-rem dose from discharges to the Missouri River. Dose from recreational activities is expected to contribute the additional 5%, which is considered to be negligible. (Ref. 9.6.7)

| 2.5.3 Calculation of Dose From Liquid Effluents

2.5.3.1 Calculation of Dose Contributions

The dose contributions for the total time period

$$\sum_{\ell=1}^{m} \Delta t_\ell$$

are calculated at least once each 31 days and a cumulative summation of the total body and individual organ doses is maintained for each calendar quarter. These dose contributions are calculated for all radionuclides identified in liquid effluents released to UNRESTRICTED AREAS using the following expression (Ref. 9.8.3)

$$D_\tau = \sum_i [A_{i\tau} \sum_{\ell=1}^{m} \Delta t_\ell \, C_{i\ell} \, F_\ell] \qquad (2.12)$$

Where:

D_τ = the cumulative dose commitment to the total body or any organ, τ, from the liquid effluents for the total period

$$\sum_{\ell=1}^{m} \Delta t_\ell$$

in mrem.

Δt_ℓ = the length of the ℓth time period over which $C_{i\ell}$ and F_ℓ are averaged for all liquid releases, in hours.

$C_{i\ell}$ = the average measured concentration of radionuclide, i, in undiluted liquid effluent during time period Δt_ℓ from any liquid release, in (μCi/ml).

$A_{i\tau}$ = the site related ingestion dose commitment factor to the total body or any organ τ for each identified principal gamma and beta emitter listed in Technical Specifications, Table 4.11-1, (in mrem/hr) per (μCi/ml). These factors are given in Table 1, as derived through the use of Equation (2.16).

F_ℓ = the near field average dilution factor for $C_{i\ell}$ during any liquid effluent release.

$$f_\ell = \frac{f_{max}}{(f_e + f_{max})\ 89.77}$$

Where:

f_{max} = maximum undiluted effluent flow rate during the release

f_e = average flow rate from the site discharge structure to unrestricted receiving waters (dilution flow)

89.77 = site specific applicable factor for the mixing effect of the discharge structure. (Ref 9.5.1)

The term $C_{i\ell}$ is the undiluted concentration of radioactive material in liquid waste at the common release point determined in accordance with Technical Specification 3.11.1.1, Table 4.11-1, "Radioactive Liquid Waste Sampling and Analysis Program". All dilution factors beyond the sample point(s) are included in the F_ℓ term.

2.5.3.2 Dose Factor Related to Liquid Effluents

| Calculating dose contributions via Equation (2.12) requires the use of a dose factor $A_{i\tau}$ for each nuclide, i, which embodies the dose factors, pathway transfer factors (e.g., bioaccumulation factors), pathway usage factors, and dilution factors for the points of pathway origin. The adult total body dose factor and the maximum adult organ dose factor for each radionuclide is used from Table E-11 of Regulatory Guide 1.109; thus, Table 1 contains critical organ dose factors for various organs. The dose factor is calculated according to (Ref. 9.8.4):

$$A_{i\tau} = k_o(U_w/D_w + U_F BF_i)DF_i \qquad (2.13)$$

Where:

$A_{i\tau}$ = composite dose parameter for the total body or critical organ of an adult for nuclide, i, for all appropriate pathways, as (mrem/hr) per (µCi/ml).

k_o = units conversion factor, derived according to: 1.14E05 = (1E06pCi/µCi x 1E03ml/kg) ÷ 8760 hr/yr.

U_F = adult fish consumption factor, equal
| to 21kg/yr (Reg. Guide 1.109, Table E-5)

BF_i = Bioaccumulation factor for nuclide, i, in fish (Table 2), as (pCi/kg) per (pCi/ℓ).

DF_i = Dose conversion factor for nuclide, i, for adults in pre-selected organ, τ, as (mrem/pCi)
| (Ref. 9.11.4 and 9.16.5).

U_w = receptor individual's water consumption by age group as per Regulatory Guide 1.109, Table E-5. For adults, U_w = 730kg/yr.

D_w = dilution factor from the near field area within one-quarter mile of the release point to the potable water intake for the adult water consumption.

NOTE

The nearest municipal potable water intake downstream from the liquid effluent discharge point into the Missouri River is located near the city of St. Louis, Mo., approximately 78 miles downstream. As there are currently no potable water intakes within 50 river miles of the discharge point, the drinking water pathway is not included in dose estimates to the maximally exposed individual, or in dose estimates to the population. Should future water intakes be constructed within 10 river miles downstream of the discharge point, then this manual will be revised to include this pathway in dose estimates. (Ref. 9.6.6). Therefore, it is not necessary to evaluate (U_w/D_w) at this time, and Equation (2.13) simplies to:

$$A_{i\tau} = k_o \ (U_F BF_i) DF_i \qquad (2.14)$$

Inserting the appropriate usage factors into Equation (2.14) yields the following expression:

$$A_{i\tau} = 1.14E05 \ (21BF_i) DF_i \qquad (2.15)$$

or $$A_{i\tau} = 2.39E06 \times BF_i \times DF_i \qquad (2.16)$$

| 2.5.4 Summary, Calculation of Dose Due to Liquid Effluents

The dose contribution for the total time period

$$\sum_{\ell=1}^{m} \Delta t_{\ell}$$

is determined by calculation at least once per 31 days and a cumulative summation of the total body and organ doses is maintained for each calendar quarter. The projected dose contribution from liquid effluents for which radionuclide concentrations are determined by periodic composite and grab sample analysis, may be approximated by using the last measured value. Dose contributions are determined for all radionuclides identified in liquid effluents released to UNRESTRICTED AREAS. Nuclides which are not detected in the analyses are reported as "less than" the nuclide's Minimum Detectable Activity (MDA) and are not reported as being present at the LLD level for that nuclide. The "less than" values are not used in the required dose calculations.

TABLE 1

INGESTION DOSE COMMITMENT FACTOR (A_{it}) FOR ADULT AGE GROUP

(mrem-hr per µci-ml)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	2.26E-01	2.26E-01	2.26E-01	2.26E-01	2.26E-01	2.26E-01
Be-7	1.30E-02	2.98E-02	1.45E-02	No Data	3.15E-02	No Data	5.16E+00
C-14	3.13E+04	6.26E+03	6.26E+03	6.26E+03	6.26E+03	6.26E+03	6.26E+03
Na-24	4.07E+02	4.07E+02	4.07E+02	4.07E+02	4.07E+02	4.07E+02	4.07E+02
P-32	4.62E+07	2.87E+06	1.78E+06	No Data	No Data	No Data	5.19E+06
Cr-51	No Data	No Data	1.27E+00	7.62E-01	2.[illegible]1-01	1.69E+00	3.2-E+02
Mn-54	No Data	4.38E+03	8.35E+02	No Data	1.30E+03	No Data	1.34E+04
Mn-56	No Data	1.10E+02	1.95E+01	No Data	1.40E+02	No Data	3.52E+03
Fe-55	6.57E+02	4.54E+02	1.06E+02	No Data	No Data	2.53E+02	2.61E+02
Fe-59	1.04E+03	2.44E+03	9.34E+02	No Data	No Data	6.81E+02	8.13E+03
Co-57	No Data	2.09E+01	3.48E+01	No Data	No Data	No Data	5.31E+02
Co-58	No Data	8.94E+01	2.00E+02	No Data	No Data	No Data	1.81E+03
Co-60	No Data	2.57E+02	5.66E+02	No Data	No Data	No Data	4.82E+03
Ni-63	3.11E+04	2.15E+03	1.04E+03	No Data	No Data	No Data	4.49E+02
Ni-65	1.26E+02	1.64E+01	7.48E+00	No Data	No Data	No Data	4.16E+02
Cu-64	NO Data	1.00E+01	4.69E+00	No Data	2.52E+01	No Data	8.52E+02
Zn-65	2.32E+04	7.38E+04	3.33E+04	No Data	4.93E+04	No Data	4.65E+04
Zn-69	4.93E+01	9.44E+01	6.56E+00	No Data	6.13E+01	No Data	1.42E+01
Br-82	No Data	No Data	2.87E+03	No Data	No Data	No Data	2.60E+03
Br-83	No Data	No Data	4.04E+01	No Data	No Data	No Data	5.81E+01
Br-84	No Data	No Data	5.26E+01	No Data	No Data	No Data	4.13E-04
Br-85	No Data	No Data	2.15E+00	No Data	No Data	No Data	0
Rb-86	No Data	1.01E+05	4.71E+04	No Data	No Data	No Data	1.99E+04
Rb-88	No Data	2.90E+02	1.54E+02	No Data	No Data	No Data	4.00E-09
Rb-89	No Data	1.92E+02	1.35E+02	No Data	No Data	No Data	1.12E-11
Sr-89	2.21E+04	No Data	6.35E+02	No Data	No Data	No Data	3.55E+03
Sr-90	5.44E+05	No Data	1.34E+05	No Data	No Data	No Data	1.57E+04
Sr-91	4.07E+02	No Data	1.64E+01	No Data	No Data	No Data	1.94E+03
Sr-92	1.54E+02	No Data	6.66E+00	No Data	No Data	No Data	3.06E+03
Y-90	5.75E-01	No Data	1.54E-02	No Data	No Data	No Data	6.10E+03
Y-91M	5.44E-03	No Data	2.10E-04	No Data	No Data	No Data	1.60E-02
Y-91	8.43E+00	No Data	1.[illegible]3E-01	No Data	No Data	No Data	4.64E+03
Y-92	5.05E-02	No Data	1.48E-03	No Data	No Data	No Data	8.85E+02
Y-93	1.60E-01	No Data	4.42E-03	No Data	No Data	No Data	5.08E+03
Zr-95	2.40E-01	7.70E-02	5.21E-02	No Data	1.21E-01	No Data	2.44E+02
Zr-97	1.33E-02	2.68E-03	1.22E-03	No Data	4.04E-03	No Data	8.30E+02
Nb-95	4.47E+02	2.48E+02	1.34E+02	No Data	2.46E+02	No Data	1.51E+06
Mo-99	No Data	1.03E+02	1.96E+01	No Data	2.33E+02	No Data	2.39E+02
Tc-99M	8.87E-03	2.51E-02	3.19E-01	No Data	3.81E-01	1.23E-02	1.48E+01
Tc-101	9.[illegible]1E-03	1.31E-02	1.29E-01	No Data	2.36E-01	6.70E-0[illegible]	0
Ru-103	4.42E+00	No Data	1.90E+00	No Data	1.69E+01	No Data	5.17E+02
Ru-105	3.68E-01	No Data	1.45E-01	No Data	4.76E+00	No Data	2.25E+02
Ru-106	6.37E+01	No Data	8.32E+00	No Data	1.27E+02	No Data	4.25E+03
Cd-109	No Data	5.34E+02	1.94E+01	No Data	5.31E+02	No Data	5.59E+03
Sb-124	6.69E+00	1.26E-01	2.65E+00	1.62E-02	No Data	5.21E+00	1.90E+02

TABLE 1 (Continued)

INGESTION DOSE COMMITMENT FACTOR (A_{it}) FOR ADULT AGE GROUP

(mrem-hr per uCi-ml)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Sb-125	4.28E+00	4.78E-02	1.02E+00	4.35E-03	No Data	3.30E+00	4.71E+01
Te-125M	2.57E+03	9.30E+02	3.44E+02	7.72E+02	1.04E+04	No Data	1.02E+04
Te-127M	6.47E+03	2.32E+03	7.90E+02	1.66E+03	2.63E+04	No Data	2.17E+04
Te-127	1.05E+02	3.76E+01	2.28E+01	7.80E+01	4.29E+02	No Data	8.30E+03
Te-129M	1.10E+04	4.11E+03	1.74E+03	3.78E+03	4.60E+04	No Data	5.54E+04
Te-129	3.01E+01	1.13E+01	7.33E+00	2.31E+01	1.26E+02	No Data	2.27E+01
Te-131M	1.66E+03	8.09E+02	6.75E+02	1.28E+03	8.21E+03	No Data	8.03E+04
Te-131	1.89E+01	7.88E+00	5.96E+00	1.55E+01	8.25E+01	No Data	2.67E+00
Te-132	2.41E+03	1.56E+03	1.47E+03	1.72E+03	1.50E+04	No Data	7.38E+04
I-130	2.71E+01	8.01E+01	3.16E+01	6.79E+03	1.25E+02	No Data	6.89E+01
I-131	1.49E+02	2.14E+02	1.22E+02	7.00E+04	3.66E+02	No Data	5.64E+01
I-132	7.29E+00	1.95E+01	6.82E+00	6.82E+02	3.11E+01	No Data	3.66E+00
I-133	5.10E+01	8.87E+01	2.70E+01	1.30E+04	1.55E+02	No Data	7.97E+01
I-134	3.81E+00	1.03E+01	3.70E+00	1.79E+02	1.64E+01	No Data	9.01E-03
I-135	1.59E+01	4.16E+01	1.54E+01	2.75E+03	6.68E+01	No Data	4.70E+01
Cs-134	2.98E+05	7.09E+05	5.80E+05	No Data	2.29E+05	7.62E+04	1.24E+04
Cs-136	3.12E+04	1.23E+05	8.86E+04	No Data	6.85E+04	9.39E+03	1.40E+04
Cs-137	3.82E+05	5.22E+05	3.42E+05	No Data	1.77E+05	5.89E+04	1.01E+04
Cs-138	2.64E+02	5.22E+02	2.59E+02	No Data	3.84E+02	3.79E+01	2.23E-03
Ba-139	9.29E-01	6.62E-04	2.72E-02	No Data	6.19E-04	3.76E-04	1.65E+00
Ba-140	1.94E+02	2.44E-01	1.27E+01	No Data	8.31E-02	1.40E-01	4.00E+02
Ba-141	4.50E-01	3.40E-04	1.52E-02	No Data	3.16E-04	1.93E-04	2.12E-10
Ba-142	2.04E-01	2.09E-04	1.28E-02	No Data	1.77E-04	1.19E-04	0
La-140	1.50E-01	7.53E-02	1.99E-02	No Data	No Data	No Data	5.53E+03
La-142	7.65E-03	3.48E-03	8.66E-04	No Data	No Data	No Data	2.54E+01
Ce-141	2.24E-02	1.51E-02	1.72E-03	No Data	7.03E-03	No Data	5.78E+01
Ce-143	3.94E-03	2.92E+00	3.23E-04	No Data	1.28E-03	No Data	1.07E+02
Ce-144	1.17E+00	4.88E-01	6.26E-02	No Data	2.89E-01	No Data	3.94E+02
Pr-143	5.50E-01	2.21E-01	2.73E-02	No Data	1.27E-01	No Data	2.41E+03
Nd-147	3.76E-01	4.35E-01	2.60E-02	No Data	2.54E-01	No Data	2.09E+03
Hf-181	3.99E-02	1.94E-01	1.80E-02	No Data	4.17E-02	No Data	2.21E+02
W-187	2.96E+02	2.47E+02	8.64E+01	No Data	No Data	No Data	8.09E+04
Np-239	2.84E-02	2.80E-03	1.54E-03	No Data	8.72E-03	No Data	5.74E+02

TABLE 2
BIOACCUMULATION FACTOR (BF_i) USED IN THE ABSENCE
OF SITE-SPECIFIC DATA[a]
(pCi/kg) per (pCi/liter)

Element	BF_i Fish (Freshwater)
H	9.0 E - 01
Be	2.0 E + 00
C	4.6 E + 03
Na	1.0 E + 02
P	1.0 E + 05
Cr	2.0 E + 02
Mn	4.0 E + 02
Fe	1.0 E + 02
Co	5.0 E + 01
Ni	1.0 E + 02
Cu	5.0 E + 01
Zn	2.0 E + 03
Br	4.2 E + 02
Rb	2.0 E + 03
Sr	3.0 E + 01
Y	2.5 E + 01
Zr	3.3 E + 00
Nb	3.0 E + 04
Mo	1.0 E + 01
Tc	1.5 E + 01
Ru	1.0 E + 01
Rh	1.0 E + 01
Cd	2.0 E + 02
Sb	1.0 E + 00
Te	4.0 E + 02
I	1.5 E + 01
Cs	2.0 E + 03
Ba	4.0 E + 00
La	2.5 E + 01
Ce	1.0 E + 00
Pr	2.5 E + 01
Nd	2.5 E + 01
Hf	3.3 E + 00
W	1.2 E + 03
Np	1.0 E + 01

(a) Values from Regulatory Guide 1.109, Rev 1, Table A-1 and UE Safety Analysis Calculation 88-002-00-F.

2.6 LIQUID RADWASTE TREATMENT SYSTEM

2.6.1 Technical Specification 3.11.1.3

The LIQUID RADWASTE TREATMENT SYSTEM Shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivty when the projected doses due to the liquid effluent, to UNRESTRICTED AREAS, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

2.6.2 OPERABILITY Of The LIQUID RADWASTE TREATMENT SYSTEM

The LIQUID RADWASTE TREATMENT SYSTEM is capable of varying treatment, depending on waste type and product desired. It is capable of concentrating, gas stripping, and distillation of liquid wastes through the use of the evaporator system. The demineralization system is capable of removing radioactive ions from solutions to be reused as makeup water. Filtration is performed on certain liquid wastes and it may, in some cases, be the only required treatment prior to release. The system has the ability to absorb halides through the use of charcoal filters prior to their release.

The design and operation requirements of the LIQUID RADWASTE TREATMENT SYSTEM provide assurance that releases of radioactive materials in liquid effluents will be kept "As Low As Reasonably Achievable" (ALARA).

The OPERABILITY of the LIQUID RADWASTE TREATMENT SYSTEM ensures this system will be available for use when liquids require treatment prior to their release to the environment. OPERABILITY is demonstrated through compliance with Technical Specifications 3.11.1.1 and 3.11.1.2.

Projected doses due to liquid releases to UNRESTRICTED AREAS are determined each 31 days by dividing the cummulative annual total by the number of elapsed months.

3.0 GASEOUS EFFLUENTS

3.1 Technical Specification 3.3.3.10

The radioactive gaseous effluent monitoring instrumentation channels shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be adjusted to the values determined in accordance with the methodology and parameters in the ODCM.

3.2 Technical Specification 3.11.2.1

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For Iodine - 131 and 133 for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr. to any organ, from the inhalation pathway only.

3.3 Gaseous Effluent Monitors

Noble gas activity monitors are present on the containment building ventilation system, plant unit ventilation system, and radwaste building ventilation system.

The alarm/trip (alarm & trip) setpoint for any gaseous effluent radiation monitor is determined based on the instantaneous noble gas total body and skin dose rate limits of Technical Specification 3.11.2.1, at the SITE BOUNDARY location with the highest annual average X/Q value. (Figure 5.1B)

Each monitor channel is provided with a two level system which provides sequential alarms on increasing radioactivity levels. These setpoints are designated as alert setpoints and alarm/trip setpoints. (Ref. 9.6.3)

The radiation monitor alarm/trip setpoints for each release point are based on the radioactive noble gases in gaseous effluents. It is not considered practicable to apply instantaneous alarm/trip setpoints to integrating radiation monitors sensitive to radioiodines, radioactive materials in particulate form and radionuclides other than noble gases. Conservative assumptions may be necessary in establishing setpoints to account for system variables, such as the measurement system efficiency and detection capabilities during normal, anticipated, and unusual operating conditions, the variability in release flow and principal radionuclides, and the time lag between alarm/trip action and the final isolation of the radioactive effluent. (Ref. 9.8.6.) Technical Specifications Table 4.3-13 provides the instrument surveillance requirements, such as calibration, source checking, functional testing, and channel checking.

3.3.1 <u>Continuous Release Gaseous Effluent Monitors</u>

The radiation detection monitors associated with continuous gaseous effluent releases are (Ref. 9.6.8, 9.6.9):

Monitor I.D.	Description
GT-RE-21	Unit Vent
GH-RE-10	Radwaste Building Vent

Each of the above continuously monitors gaseous radioactivity concentrations downstream of the last point of potential influent, and therefore measures effluents and not inplant concentrations.

The Unit Vent monitor continuously monitors the effluent from the unit vent for gaseous radioactivity. The Unit Vent, via ventilation exhaust systems, continuously purges various tanks and sumps normally containing low-level radioactive aerated liquids that can potentially generate airborne activity.

The exhaust systems which supply air to the unit vent are from the fuel building, auxiliary building, the access control area, the containment purge, and the condenser air discharge.

The Unit Vent monitor provides alarm functions only, and does not terminate releases from the Unit Vent.

The Radwaste Building Ventilation effluent monitor continuously monitors for gaseous radioactivity in the effluent duct downstream of the exhaust filter and fans. The flow path provides ventilation exhaust for all parts of the building structure and components within the building and provides a discharge path for the waste gas decay tank release line. These components represent potential sources for the release of gaseous and air particulate and iodine activities in addition to the drainage sumps, tanks, and equipment purged by the waste processing system.

This monitor will isolate the waste gas decay tank discharge line upon a high gaseous radioactivity alarm.

The continuous gaseous effluent monitor setpoints are established using the methodology described in Section 3.4. Since there are two continuous gaseous effluent release points, a fraction of the total dose rate limit (DRL) will be allocated to each release point. Neglecting the batch releases, the plant Unit Vent monitor has been allocated 0.7 DRL and the Radwaste Building Vent monitor has been allocated 0.3 DRL. These allocation factors may be changed as required to support plant operational needs, but shall not be allowed to exceed unity (i.e. 1.0). Therefore, a particular monitor reaching the setpoint would not necessarily mean the dose rate limit at the SITE BOUNDARY is being exceeded; the alarm only indicates that the specific release point is contributing a greater fraction of the dose rate limit than was allocated to the associated monitor, and will necessitate an evaluation of both systems.

3.3.2 Batch Release Gaseous Monitors

The radiation monitors associated with batch release gaseous effluents are (Ref. 9.6.9, 9.6.10, 9.6.11):

Monitor I.D.	Description
GT-RE-22 GT-RE-33	Containment Purge System
GH-RE-10	Radwaste Building Vent

The Containment Purge System monitors continuously monitor the containment purge exhaust duct during purge operations for gaseous radioactivity. The primary purpose of these monitors is to isolate the containment purge system on high gaseous activity via the ESFAS.

The sample points are located outside the containment between the containment isolation dampers and the containment purge filter adsorber unit.

The Radwaste Building Vent monitor was previously described in Section 3.3.1.

Setpoints for the batch gaseous effluent monitors are calculated using the methodology described in Section 3.4.

A pre-release isotopic analysis is performed for each batch release to determine the identity and quantity of the principal radionuclides. The alarm/trip setpoint(s) is adjusted accordingly to ensure that the limits of Technical Specification 3.11.2.1 are not exceeded.

| 3.4 Determination of Gaseous Effluent Monitor Setpoints

The alarm/trip setpoint for gaseous effluent monitors is determined based on the lesser of the total body dose rate and skin dose rate, as calculated for the SITE BOUNDARY.

3.4.1 Total Body Dose Rate Setpoint Calculations

To ensure that the limits of Technical Specification 3.11.2.1 are met, the alarm/trip setpoint based on the total body dose rate is calculated according to:

$$S_{tb} \leq D_{tb} R_{tb} F_s F_a \qquad (3.1)$$

Where:

S_{tb} = the alarm/trip setpoint based on the total body dose rate (μCi/cc).

D_{tb} = Technical Specification 3.11.2.1 limit of 500 mrem/yr, conservatively interpreted as a continuous release over a one year period.

F_s = the safety factor; a conservative factor used to compensate for statistical fluctuations and errors of measurement. (For example, F_s = 0.5 corresponds to a 100% variation.) Default value is F_s = 1.0.

F_a = the allocation factor which will modify the required dilution factor such that simultaneous gaseous releases may be made without exceeding the limits of Radiological Effluent Technical Specification 3.11.2.1. The default value is 1/n, where n is the number of pathways planned for release.

R_{tb} = factor used to convert dose rate to the effluent concentration as measured by the effluent monitor, in (μCi/cc) per (mrem/yr) to the total body, determined according to:

$$R_{tb} = C \div [(\overline{X/Q}) \sum_i K_i Q_i] \quad (3.2)$$

Where:

C = monitor reading of a noble gas monitor corresponding to the sample radionuclide concentrations for the batch to be released. Concentrations are determined in accordance with Technical Specifications Table 4.11-2. The mixture of radionuclides determined via grab sampling of the effluent stream or source is correlated to a calibration factor to determine monitor response. The monitor response is based on concentrations, not release rate and is in units of (μCi/cc).

$\overline{X/Q}$ = the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY in (sec/m³). Refer to Tables 9, 10, and 12.

K_i = the total body dose factor due to gamma emissions for each identified noble gas radionuclide, in (mrem/yr) per (μCi/m³). (Table 3)

Q_i = rate of release of noble gas radionuclide, i, in (μCi/sec).

Q_i is calculated as the product of the ventilation path design flow rate and the measured activity of the effluent stream as determined by grab sampling. Flow rates for the ventilation pathways can be found in references 9.6.21, 9.6.22, 9.6.23, and 9.6.24.

3.4.2 Skin Dose Rate Setpoint Calculation

To ensure that the limits of Technical Specification 3.11.2.1 are met, the alarm/trip setpoint based on the skin dose rate is calculated according to:

$$S_s \le D_s R_s F_s F_a \qquad (3.3)$$

Where:

F_s and F_a are as previously defined in Section 3.4.1.1.

S_s = the alarm/trip setpoint based on the skin dose rate.

D_s = Technical Specification 3.11.2.1 limit of 3000 mrem/yr, conservatively interpreted as a continuous release over a one year period.

R_s = factor used to convert dose rate to the effluent concentration as measured by the effluent monitor, in (μCi/cc) per (mrem/yr) to the skin, determined according to:

$$R_s = C \div [(\overline{X/Q}) \sum_i (L_i + 1.1M_i) Q_i] \qquad (3.4)$$

Where:

L_i = the skin dose factor due to beta emissions for each identified noble gas radionuclide, in (mrem/yr) per (μCi/m^3). (Table 3)

1.1 = conversion factor: 1 mrad air dose = 1.1 mrem skin dose.

M_i = the air dose factor due to gamma emissions for each identified noble gas radionuclide, in (mrad/yr) per (μCi/m^3). (Table 3)

C, $(\overline{X/Q})$ and Q_i are as previously defined.

3.4.3 Gaseous Effluent Monitors Setpoint Determination

The results of Equation (3.1) and Equation (3.3) are compared. The setpoint is then selected as the lesser of the two values.

TABLE 3

DOSE FACTORS FOR EXPOSURE TO A SEMI-INFINITE CLOUD OF NOBLE GASES[a]

Radionuclide	Total Body Dose Factor K_i (mrem/yr) per (μCi/m³)	Skin Dose Factor L_i (mrem/yr) per (μCi/m³)	Gamma Air Dose Factor M_i (mrad/yr) per (μCi/m³)	Beta Air Dose Factor N_i (mrad/yr) per (μCi/m³)
Kr-83m	7.56 E-02	- - -	1.93 E+01	2.88 E+02
Kr-85m	1.17 E+03	1.46 E+03	1.23 E+03	1.97 E+03
Kr-85	1.61 E+01	1.34 E+03	1.72 E+01	1.95 E+03
Kr-87	5.92 E+03	9.73 E+03	6.17 E+03	1.03 E+04
Kr-88	1.47 E+04	2.37 E+03	1.52 E+04	2.93 E+03
Kr-89	1.66 E+04	1.01 E+04	1.73 E+04	1.06 E+04
Kr-90	1.56 E+04	7.29 E+03	1.63 E+04	7.83 E+03
Xe-131m	9.15 E+01	4.76 E+02	1.56 E+02	1.11 E+03
Xe-133m	2.51 E+02	9.94 E+02	3.27 E+02	1.48 E+03
Xe-133	2.94 E+02	3.06 E+02	3.53 E+02	1.05 E+03
Xe-135m	3.12 E+03	7.11 E+02	3.36 E+03	7.39 E+02
Xe-135	1.81 E+03	1.86 E+03	1.92 E+03	2.46 E+03
Xe-137	1.42 E+03	1.22 E+04	1.51 E+03	1.27 E+04
Xe-138	8.83 E+03	4.13 E+03	9.21 E+03	4.75 E+03
Ar-41	8.84 E+03	2.69 E+03	9.30 E+03	3.28 E+03

(a) The listed dose factors are derived from Reg. Guide 1.109, Table B-1 (Rev. 1, 1977).

3.4.4 Summary, Gaseous Effluent Monitors Setpoint Determination

The gaseous effluent monitors setpoints are calculated according to equations (3.1) and (3.2), as described in Section 3.4. However, it should be noted that a batch release will alter the flow rate characteristics at the Unit Vent and therefore the concentration as sensed by the monitor. For example, in the case of a mini-purge, the setpoint for the Unit Vent monitor must be re-calculated to include both the continuous and batch sources.

3.5 Calculation of Dose From Gaseous Effluents

Dose rate calculations are performed for gaseous effluents to ensure compliance with Technical Specification 3.11.2.1.

3.5.1 Calculation of Dose Rate

The following methodology is applicable to the location (SITE BOUNDARY or beyond) characterized by the values of the parameter (X/Q) which results in the maximum total body or skin dose rate. In the event that the analysis indicates a different location for the total body and skin dose limitations, the location selected for consideration is that which minimizes the allowable release values. (Ref. 9.8.7)

The factors K_i, L_i, and M_i relate the radionuclide airborne concentrations to various dose rates, assuming a semi-infinite cloud model, and are tabulated in Table 3.

3.5.1.1 Noble Gases

The release rate limit for noble gases is determined according to the following general relationships (Ref. 9.8.7):

$$D_{tb} = \sum_i [K_i((\overline{X/Q})Q_i)] \leq 500 \text{ mrem/yr} \qquad (3.5)$$

$$D_s = \sum_i [(L_i + 1.1\ M_i)((\overline{X/Q})Q_i)] \leq 3000\ \text{mrem/yr} \qquad (3.6)$$

Where:

D_{tb} = Total body dose rate, conservatively averaged over a period of one year.

K_i = Total body dose factor due to gamma emissions for each identified noble gas radionuclide, in (mrem/yr) per ($\mu Ci/m^3$). (Table 3)

$(\overline{X/Q})$ = The highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY. Refer to Tables 9, 10, and 12.

Q_i = The release rate of noble gas radionuclides, i, in gaseous effluents, from all vent releases in (μCi/sec).

Q_i is calculated as the product of the ventilation path design flow rate and the measured activity of the effluent stream as determined by grab sampling. Flow rates for the ventilation pathways can be found in references 9.6.21, 9.6.22, 9.6.23, and 9.6.24.

D_s = Skin dose rate, conservatively averaged over a period of one year.

L_i = Skin dose factor due to beta emissions for each identified noble gas radionuclide, in (mrem/yr) per ($\mu Ci/m^3$) (Table 3).

1.1 = Units conversion factor; 1 mrad air dose = 1.1 mrem skin dose.

M_i = Air dose factor due to gamma emissions for each identified noble gas radionuclide, in (mrad/yr) per ($\mu Ci/m^3$) (Table 3).

3.5.1.2 Radionuclides Other Than Noble Gases

The release rate limit for Iodine-131 and-133, for tritium, and for all radioactive materials in particulate form with half lives greater than 8 days is determined according to (Ref. 9.8.8):

$$D_o = \sum_i P_i[(\overline{X/Q})Q_i] \leq 1500 \text{ mrem/yr} \quad (3.7)$$

Where:

D_o = Dose rate to any critical organ, in (mrem/yr).

P_i = Dose parameter for radionuclides other than noble gases for the inhalation pathway for the child, based on the critical organ, in (mrem/yr) per ($\mu Ci/m^3$). (Table 4)

Q_i = The release rate of radionuclide, i, in gaseous effluents, from all vent releases, in ($\mu Ci/sec$). Q_i is calculated as the product of the ventilation path design flow rate and the measured activity of the effluent stream as determined by grab sampling. Flow rates for the ventilation pathways can be found in references 9.6.21, 9.6.22, 9.6.23, and 9.6.24.

$(\overline{X/Q})$ is as previously defined.

The dose parameter (P_i) includes the internal dosimetry of radionuclide, i, and the receptor's breathing rate, which are functions of the receptor's age. Therefore the child age group has been selected as the limiting age group.

For the child exposure, separate values of P_i are tabulated in Table 4 for the inhalation pathway. These values were calculated according to (Ref. 9.8.9):

$$P_i = K' (BR) DFA_i \quad (3.8)$$

Where:

K' = Units conversion factor: 1µCi = 1E06 pCi.

BR= The breathing rate of the maximum exposed child age group, 3700 m^3/yr. (Regulatory Guide 1.109, Table E-5).

DFA_i = The maximum organ inhalation dose factor for the child age group for the ith radionuclide, in (mrem/pCi). The total body is considered as an organ in the selection of DFA_i. (Ref. 9.11.5 and 9.16.5)

Note: All radioiodines are assumed to be released in elemental form. (Ref.9.8.8)

TABLE 4

DOSE PARAMETER (P_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Inhalation Pathway

(mrem-hr) per ($\mu Ci/m^3$)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	ND	1.12E3	1.12E3	1.12E3	1.12E3	1.12E3	1.12E3
Be-7	8.47E2	1.44E3	9.25E2	ND	ND	6.48E4	2.55E3
C-14	3.59E4	6.73E3	6.73E3	6.73E3	6.73E3	6.73E3	6.73E3
Na-24	1.61E4	1.61E4	1.61E4	1.61E4	1.61E4	1.61E4	1.61E4
P-32	2.60E6	1.14E5	9.88E4	ND	ND	ND	4.22E4
Cr-51	ND	ND	1.54E2	8.55E1	2.43E1	1.70E4	1.08E3
Mn-54	ND	4.29E4	9.51E3	ND	1.00E4	1.58E6	2.29E4
Mn-56	ND	1.66E0	3.12E-1	ND	1.67E0	1.31E4	1.23E5
Fe-55	4.74E4	2.52E4	7.72E3	ND	ND	1.11E5	2.87E3
Fe-59	2.07E4	3.34E4	1.67E4	ND	ND	1.27E6	7.07E4
Co-57	ND	9.03E2	1.07E3	ND	ND	5.07E5	1.32E4
Co-58	ND	1.77E3	3.16E3	ND	ND	1.11E6	3.44E4
Co-60	ND	1.31E4	2.26E4	ND	ND	7.07E6	9.26E4
Ni-63	8.21E5	4.63E4	2.80E4	ND	ND	2.75E5	6.33E3
Ni-65	2.99E0	2.96E-1	1.64E-1	ND	ND	8.18E3	8.40E4
Cu-64	ND	1.99E0	1.07E0	ND	6.03E0	9.58E3	3.67E4
Zn-65	4.26E4	1.13E5	7.03E4	ND	7.14E4	9.95E5	1.63E4
Zn-69	6.70E-2	9.66E-2	8.92E-3	ND	5.85E-2	1.42E3	1.02E4
Br-82	ND	ND	4.18E3	ND	ND	ND	ND
Br-83	ND	ND	4.74E2	ND	ND	ND	0
Br-84	ND	ND	5.48E2	ND	ND	ND	0
Br-85	ND	ND	2.53E1	ND	ND	ND	0
Rb-86	ND	1.98E5	1.14E3	ND	ND	ND	7.99E3
Rb-88	ND	5.62E2	3.66E2	ND	ND	ND	1.72E1
Rb-89	ND	3.45E2	2.90E2	ND	ND	ND	1.89E0
Sr-89	5.99E5	ND	1.72E4	ND	ND	2.16E6	1.67E5
Sr-90	1.01E8	ND	6.44E6	ND	ND	1.48E7	3.43E5
Sr-91	1.21E2	ND	4.59E0	ND	ND	5.32E4	1.74E5
Sr-92	1.31E1	ND	5.25E-1	ND	ND	2.40E4	2.42E5
Y-90	4.11E3	ND	1.11E2	ND	ND	2.62E5	2.68E5

TABLE 4 (Cont'd.)

DOSE PARAMETER (P_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Inhalation Pathway

(mrem-hr) per ($\mu Ci/m^3$)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Y-91m	5.07E-1	ND	1.84E-2	ND	ND	2.81E3	1.72E3
Y-91	9.14E5	ND	2.44E4	ND	ND	2.63E6	1.84E5
Y-92	2.04E1	ND	5.81E-1	ND	ND	2.39E4	2.39E5
Y-93	1.86E2	ND	5.11E0	ND	ND	7.44E4	3.89E5
Zr-95	1.90E5	4.18E4	3.70E4	ND	5.96E4	2.23E6	6.11E4
Zr-97	1.88E2	2.72E1	1.60E1	ND	3.89E1	1.13E5	3.51E5
Nb-95	2.33E4	9.18E3	6.55E3	ND	8.62E3	6.14E5	3.70E4
Mo-99	ND	1.72E2	4.26E1	ND	3.92E2	1.35E5	1.27E5
Tc-99m	1.78E-3	3.48E-3	5.77E-2	ND	5.07E-2	9.51E2	4.81E3
Tc-101	8.10E-5	8.51E-5	1.08E-3	ND	1.45E-3	5.85E2	1.63E1
Ru-103	2.79E3	ND	1.07E3	ND	7.03E3	6.62E5	4.48E4
Ru-105	1.53E0	ND	5.55E-1	ND	1.34E0	1.59E4	9.95E4
Ru-106	1.36E5	ND	1.69E4	ND	1.84E5	1.43E7	4.29E5
Ag-110m	1.69E4	1.14E4	9.14E3	ND	2.12E4	5.48E6	1.00E5
Cd-109	ND	5.48E5	2.59E4	ND	4.96E5	1.05E6	2.78E4
Sb-124	5.74E4	7.40E2	2.00E4	1.26E2	ND	3.24E6	1.64E5
Sb-125	9.84E4	7.59E2	2.07E4	9.10E1	ND	2.32E6	4.03E4
Te-125m	6.73E3	2.33E3	9.14E2	1.92E3	ND	4.77E5	3.38E4
Te-127m	2.49E4	8.55E3	3.02E3	6.07E3	6.36E4	1.48E6	7.14E4
Te-127	2.77E0	9.51E-1	6.11E-1	1.96E0	7.07E0	1.00E4	5.62E4
Te-129m	1.92E4	6.85E3	3.04E3	6.33E3	5.03E4	1.76E6	1.82E5
Te-129	9.77E-2	3.50E-2	2.38E-2	7.14E-2	2.57E-1	2.93E3	2.55E4
Te-131m	1.34E2	5.92E1	5.07E1	9.77E1	4.00E2	2.06E5	3.08E5
Te-131	2.17E-2	8.44E-3	6.59E-3	1.70E-2	5.88E-2	2.05E3	1.33E3
Te-132	4.81E2	2.72E2	2.63E2	3.17E2	1.77E3	3.77E3	1.38E5
I-130	8.18E3	1.64E4	8.44E3	1.85E6	2.45E4	ND	5.11E3
I-131	4.81E4	4.81E4	2.73E4	1.62E7	7.88E4	ND	2.84E3
I-132	2.12E3	4.07E3	1.88E3	1.94E5	6.25E3	ND	3.20E3
I-133	1.66E4	2.03E4	7.70E3	3.85E6	3.38E4	ND	5.48E3
I-134	1.17E3	2.16E3	9.95E2	5.07E4	3.30E3	ND	9.55E2
I-135	4.92E3	8.73E3	4.14E3	7.92E5	1.34E4	ND	4.44E3

TABLE 4 (Cont'd.)

DOSE PARAMETER (P_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Inhalation Pathway

(mrem-hr) per ($\mu Ci/m^3$)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Cs-134	6.51E5	1.01E6	2.25E5	ND	3.03E5	1.21E5	3.85E3
Cs-136	6.51E4	1.71E5	1.16E5	ND	9.55E4	1.45E4	4.18E3
Cs-137	9.07E5	8.25E5	1.28E5	ND	2.72E5	1.04E5	3.62E3
Cs-138	6.33E2	8.40E2	5.55E2	ND	6.22E2	6.81E1	2.70E2
Ba-139	1.84E0	9.84E-4	5.37E-2	ND	8.62E-4	5.77E3	5.77E4
Ba-140	7.40E4	6.48E1	4.33E3	ND	2.11E1	1.74E6	1.02E5
Ba-141	2.19E-1	1.09E-4	6.36E-3	ND	9.47E-5	2.92E3	2.75E2
Ba-142	5.00E-2	3.60E-5	2.79E-3	ND	2.91E-5	1.64E3	2.74E0
La-140	6.44E2	2.25E2	7.55E1	ND	ND	1.83E5	2.26E5
La-142	1.30E0	4.11E-1	1.29E-1	ND	ND	8.70E3	7.59E4
Ce-141	3.92E4	1.95E4	2.90E3	ND	8.55E3	5.44E5	5.66E4
Ce-143	3.66E2	1.99E2	2.87E1	ND	8.36E1	1.15E5	1.27E5
Ce-144	6.77E6	2.12E6	3.61E5	ND	1.17E6	1.20E7	3.89E5
Pr-143	1.85E4	5.55E3	9.14E2	ND	3.00E3	4.33E5	9.73E4
Pr-144	5.96E-2	1.85E-2	3.00E-3	ND	9.77E-3	1.57E3	1.97E2
Nd-147	1.08E4	8.73E3	6.81E2	ND	4.81E3	3.28E5	8.21E4
Hf-181	2.78E4	1.01E5	1.25E4	ND	2.05E4	1.06E6	6.62E4
W-187	1.63E1	9.66E0	4.33E0	ND	ND	4.11E4	9.10E4
Np-239	4.66E2	3.34E1	2.35E1	ND	9.73E1	5.81E4	6.40E4

(a) The child age group; Table E-9 Reg. Guide 1.109, Rev. 1, 1977 and UE Safety Analysis Calculation 88-002-00-F.

3.5.2 Dose Due To Gaseous Effluents

3.5.2.1 Technical Specification 3.11.2.2

The air dose due to noble gases released in gaseous effluents, to areas at and beyond the SITE BOUNDARY shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

3.5.2.1.1 Noble Gases

The air dose at the SITE BOUNDARY due to noble gases released from the site is calculated according to the following methodology (Ref. 9.8.10):

During any calendar quarter, for gamma radiation:

$$D_g = 3.17\ \text{E-08} \sum_i [M_i ((\overline{X/Q})\, Q_i + (X/q)\, q_i)] \leq 5 \text{ mrad} \qquad (3.9)$$

During any calendar quarter, for beta radiation:

$$D_b = 3.17\ \text{E-08} \sum_i [N_i ((\overline{X/Q})\, Q_i + (X/q)\, q_i)] \leq 10 \text{ mrad} \qquad (3.10)$$

During any calendar year, for gamma radiation:

$$D_g = 3.17\ \text{E-08} \sum_i [M_i ((\overline{X/Q})\, Q_i + (X/q)\, q_i)] \leq 10 \text{ mrad} \qquad (3.11)$$

During any calendar year, for beta radiation:

$$D_b = 3.17\ \text{E-08} \sum_i [N_i ((\overline{X/Q})\, Q_i + (X/q)\, q_i)] \leq 20 \text{ mrad} \qquad (3.12)$$

Where:

D_g = Air dose from gamma radiation due to noble gases released in gaseous effluent.

D_b = Air dose from beta radiation due to noble gases released in gaseous effluents.

(X/q) = The relative concentration for areas at or beyond the SITE BOUNDARY for short-term releases (equal to or less than 500 hrs/year). Refer to Tables 9, 10, 11, and 12.

q_i = The average release of noble gas radionuclides, i, in gaseous effluents from all vent releases for short-term releases (equal to or less than 500 hrs/year), in (μCi). Releases are cumulative over the calendar quarter or year, as appropriate.

N_i = The air dose factor due to beta emissions for each identified noble gas radionuclide, i, in (mrad/yr) per (μCi/m³). (Table 3)

Q_i = The average release of noble gas radionuclides, i, in gaseous effluents from all vent releases for long-term releases (greater than 500 hrs/year), in (μCi). Releases are cumulative over the calendar quarter or year, as appropriate.

$(\overline{X/Q})$ = The highest calculated annual average relative concentration for areas at or beyond the SITE BOUNDARY for long-term releases (greater than 500 hrs/yr). Refer to Tables 9, 10, and 12.

3.17E-08 = The inverse of the number of seconds per year.

M_i is as previously defined. (Refer to Section 3.4.1.2)

3.5.2.2 Technical Specification 3.11.2.3

The dose to a MEMBER OF THE PUBLIC from Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, to areas at and beyond the SITE BOUNDARY shall be limited to the following (Ref. 9.8.10):

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

3.5.2.2.1. Radionuclides Other Than Noble Gases

The dose to a MEMBER OF THE PUBLIC from Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, to areas at and beyond the SITE BOUNDARY, is calculated according to the following expressions:

During any calendar quarter:

$$D_i = 3.17E\text{-}08 \sum_i R_i \, [W \, Q_i + w \, q_i] \leq 7.5 \text{ mrem} \qquad (3.13)$$

During any calendar year:

$$D_i = 3.17E\text{-}08 \sum_i R_i \, [W \, Q_i + w \, q_i] \leq 15 \text{ mrem} \qquad (3.14)$$

Where:

D_i = Dose to a MEMBER OF THE PUBLIC from radionuclides other than noble gases.

| Q_i = The releases of radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases, i, in gaseous effluents, for all vent releases for long-term releases (greater than 500 hrs/yr), in (μCi). Releases are cumulative over the calendar quarter or year as appropriate.

| q_i = The releases of radioiodines, radioactive materials in particulate form and radionuclides other than noble gases, i, in gaseous effluents for all vent releases for short-term releases (equal to or less than 500 hrs/yr), in (μCi). Releases are cumulative over the calendar quarter or year as appropriate.

R_i = The dose factor for each identified radionuclide, i, in m^2(mrem/yr) per (μCi/sec) or (mrem/yr) per (μCi/m^3). (Table 5)

W = The dispersion parameter for estimating the dose to an individual at the controlling location for long-term releases (greater than 500 hrs/yr):

W = $(\overline{X/Q})$ for the inhalation and tritium pathways, in(sec/m^3).

W = $(\overline{D/Q})$ for the food and ground plane pathways, in(meters^{-2}). Refer to Tables 9, 10, and 12.

w = The dispersion parameter for estimating the dose to an individual at the controlling location for short-term releases (equal to or less than 500 hrs/yr):

w = (X/q) for the inhalation pathway, in(sec/m^3)

w = (D/q) for the food and ground plane pathway, in (meters^{-2}). Refer to Tables 9, 10, 11, and 12.

3.17 E-08 = The inverse of the number of seconds per year.

($\overline{D/Q}$) = the average relative deposition of the effluent at or beyond the SITE BOUNDARY, considering depletion of the plume during transport, for long term releases (greater than 500 hrs/yr), in (meters^{-2}).

(D/q) = the relative deposition of the effluent at or beyond the SITE BOUNDARY, considering depletion of the plume during transport, for short term releases (less than or equal to 500 hrs/yr), in ($meters^{-2}$).

Note: For the direction sectors with existing pathways within 5 miles from the site, the appropriate R_i values are used. If no real pathway exists within 5 miles from the center of the building complex, the cow-milk R_i value is used, and it is assumed that this pathway exists at the 4.5 to 5.0 mile distance in the limiting-case sector. If the R_i for an existing pathway within 5 miles is less than a cow-milk R_i at 4.5 to 5.0 miles, then the value of the cow-milk R_i at 4.5 to 5.0 miles is used. (Rev. 9.8.10.)

Although the annual average relative concentration (X/Q) and the average relative deposition rate (D/Q) are generally considered to be at the approximate receptor location in lieu of the SITE BOUNDARY for these calculations, it is acceptable to consider the ingestion, inhalation, and ground plane pathways to coexist at the location of the nearest residence with the highest value of (X/Q). (Ref. 9.8.10) The Total Body dose from ground plane deposition is added to the dose for each individual organ. (Ref. 9.11.3)

TABLE 5

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Inhalation Pathway

(mrem-hr) per (μCi/m³)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	ND	1.12E3	1.12E3	1.12E3	1.12E3	1.12E3	1.12E3
Be-7	8.47E2	1.44E3	9.25E2	ND	ND	6.48E4	2.55E3
C-14	3.59E4	6.73E3	6.73E3	6.73E3	6.73E3	6.73E3	6.73E3
Na-24	1.61E4	1.61E4	1.61E4	1.61E4	1.61E4	1.61E4	1.61E4
P-32	2.60E6	1.14E5	9.88E4	ND	ND	ND	4.22E4
Cr-51	ND	ND	1.54E2	8.55E1	2.43E1	1.70E4	1.08E3
Mn-54	ND	4.29E4	9.51E3	ND	1.00E4	1.58E6	2.29E4
Mn-56	ND	1.66E0	3.12E-1	ND	1.67E0	1.31E4	1.23E5
Co-57	ND	9.03E2	1.07E3	ND	ND	5.07E5	1.32E4
Fe-55	4.74E4	2.52E4	7.72E3	ND	ND	1.11E5	2.87E3
Fe-59	2.07E4	3.34E4	1.67E4	ND	ND	1.27E6	7.07E4
Co-58	ND	1.77E3	3.16E3	ND	ND	1.11E6	3.44E4
Co-60	ND	1.31E4	2.26E4	ND	ND	7.07E6	9.26E4
Ni-63	8.21E5	4.63E4	2.80E4	ND	ND	2.75E5	6.33E3
Ni-65	2.99E0	2.96E-1	1.64E-1	ND	ND	8.18E3	8.40E4
Cu-64	ND	1.99E0	1.07E0	ND	6.03E0	9.58E3	3.67E4
Zn-65	4.26E4	1.13E5	7.03E4	ND	7.14E4	9.95E5	1.63E4
Zn-69	6.70E-2	9.66E-2	8.92E-3	ND	5.85E-2	1.42E3	1.02E4
Br-82	ND	ND	4.18E3	ND	ND	ND	ND
Br-83	ND	ND	4.74E2	ND	ND	ND	0
Br-84	ND	ND	5.48E2	ND	ND	ND	0
Br-85	ND	ND	2.53E1	ND	ND	ND	0
Rb-86	ND	1.98E5	1.14E5	ND	ND	ND	7.99E3
Rb-88	ND	5.62E2	3.66E2	ND	ND	ND	1.72E1
Rb-89	ND	3.45E2	2.90E2	ND	ND	ND	1.89E0
Sr-89	5.99E5	ND	1.72E4	ND	ND	2.16E6	1.67E5
Sr-90	1.01E8	ND	6.44E6	ND	ND	1.48E7	3.43E5
Sr-91	1.21E2	ND	4.59E0	ND	ND	5.33E4	1.74E5
Sr-92	1.31E1	ND	5.25E-1	ND	ND	2.40E4	2.42E5
Y-90	4.11E3	ND	1.11E2	ND	ND	2.62E5	2.68E5

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Inhalation Pathway

(mrem-hr) per (μCi/m³)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Y-91m	3.07E-1	ND	1.84E-2	ND	ND	2.81E3	1.72E3
Y-91	9.14E5	ND	2.44E4	ND	ND	2.63E6	1.84E5
Y-92	2.04E1	ND	5.81E-1	ND	ND	2.39E4	2.39E5
Y-93	1.86E2	ND	5.11E0	ND	ND	7.44E4	3.89E5
Zr-95	1.90E5	4.18E4	3.70E4	ND	5.96E4	2.23E6	6.11E4
Zr-97	1.88E2	2.72E1	1.60E1	ND	3.89E1	1.13E5	3.51E5
Nb-95	2.33E4	9.18E3	6.55E3	ND	8.62E3	6.14E5	3.70E4
Mo-99	ND	1.72E2	4.26E1	ND	3.92E2	1.35E5	1.27E5
Tc-99m	1.78E-3	3.48E-3	5.77E-2	ND	5.07E-2	9.51E2	4.81E3
Tc-101	8.10E-5	8.51E-5	1.08E-3	ND	1.45E-3	5.85E2	1.63E1
Ru-103	2.79E3	ND	1.07E3	ND	7.03E3	6.62E5	4.48E4
Ru-105	1.53E0	ND	5.55E-1	ND	1.34E0	1.59E4	9.93E4
Ru-106	1.36E5	ND	1.69E4	ND	1.84E5	1.43E7	4.29E5
Ag-110m	1.69E4	1.14E4	9.14E3	ND	2.12E4	5.48E6	1.00E5
Cd-109	ND	5.48E5	2.59E4	ND	4.96E5	1.05E6	2.78E4
Sb-124	5.74E4	7.40E2	2.00E4	1.26E2	ND	3.24E6	1.64E5
Sb-125	9.84E4	7.59E2	2.07E4	9.10E1	ND	2.32E6	4.03E4
Te-125m	6.73E3	2.33E3	9.14E2	1.92E3	ND	4.77E5	3.38E4
Te-127m	2.49E4	8.55E3	3.02E3	6.07E3	6.36E4	1.48E6	7.14E4
Te-127	2.77E0	9.51E-1	6.11E-1	1.96E0	7.07E0	1.00E4	5.62E4
Te-129m	1.92E4	6.85E3	3.04E3	6.33E3	5.03E4	1.76E6	1.82E5
Te-129	9.77E-2	3.50E-2	2.38E-2	7.14E-2	2.57E-1	2.93E3	2.55E4
Te-131m	1.34E2	5.92E1	5.07E1	9.77E1	4.00E2	2.06E5	3.08E5
Te-131	2.17E-2	8.44E-3	6.59E-3	1.70E-2	5.88E-2	2.05E3	1.33E3
Te-132	4.81E2	2.72E2	2.63E2	3.17E2	1.77E3	3.77E5	1.38E5
I-130	8.18E3	1.64E4	8.44E3	1.85E6	2.45E4	ND	5.11E3
I-131	4.81E4	4.81E4	2.73E4	1.62E7	7.88E4	ND	2.84E3
I-132	2.12E3	4.07E3	1.88E3	1.94E5	6.25E3	ND	3.20E3
I-133	1.66E4	2.03E4	7.70E3	3.85E6	3.38E4	ND	5.48E3
I-134	1.17E3	2.16E3	9.95E2	5.07E4	3.30E3	ND	9.55E2

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Inhalation Pathway

(mrem-hr) per (μCi/m³)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
I-133	4.92E3	8.73E3	4.14E3	7.92E5	1.34E4	ND	4.44E3
Cs-134	6.51E5	1.01E6	2.25E5	ND	3.03E5	1.21E5	3.85E3
Cs-136	6.51E4	1.71E5	1.16E5	ND	9.55E4	1.45E4	4.18E3
Cs-137	9.07E5	8.25E5	1.28E5	ND	2.72E5	1.04E5	3.62E3
Cs-138	6.03E2	8.40E2	5.55E2	ND	.22E2	6.81E1	2.70E2
Ba-139	1.84E0	9.84E-4	5.37E-2		.62E-4	5.77E3	5.77E4
Ba-140	7.40E4	6.48E1	4.33E3		2.11E1	1.74E6	1.02E5
Ba-141	2.19E-1	1.09E-4	6.36E-3		9.47E-5	2.02E3	2.75E2
Ba-142	5.00E-2	3.60E-5	2.79E-3	[illegible]	2.91E-5	1.64E3	2.74E0
La-140	6.44E2	2.25E2	7.55E1	ND	ND	1.83E5	2.26E5
La-142	1.30E0	4.11E-1	1.29E-1	ND	ND	8.70E3	7.59E4
Ce-141	3.92E4	1.95E4	2.90E3	ND	8.55E3	5.44E5	5.66E4
Ce-143	3.66E2	1.99E2	2.87E1	ND	8.36E1	1.15E5	1.27E5
Ce-144	6.77E6	2.12E6	3.61E5	ND	1.17E6	1.20E7	3.89E5
Pr-143	1.85E4	5.55E3	9.14E2	ND	3.00E3	4.33E5	9.73E4
Pr-144	5.96E-2	1.85E-2	3.00E-3	ND	9.77E-3	1.57E3	1.97E2
Nd-147	1.08E4	8.73E3	6.81E2	ND	4.81E3	3.28E5	8.21E4
Hf-181	2.78E4	1.01E5	1.25E4	ND	2.05E4	1.06E6	6.62E4
W-187	1.63E1	9.66E0	4.33E0	ND	ND	4.11E4	9.10E4
Np-239	4.66E2	3.34E1	2.35E1	ND	9.73E1	5.81E4	6.40E4

(a) The child age group; Table E-9 Reg. Guide 1.109, Rev. 1, 1977, and UE Safety Analysis Calculation 88-002-00-F.

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Ground Plane Pathway

(M^2 mrem/yr) per (μCi/sec)

Nuclide	Total Body	Skin
Na-24	1.19E7	1.39E7
Cr-51	4.65E6	5.51E6
Mn-54	1.39E9	1.62E9
Mn-56	9.03E5	1.07E6
Fe-59	2.72E8	3.20E8
Co-58	3.79E8	4.44E8
Co-60	2.15E10	2.53E10
Ni-65	2.97E5	3.45E5
Cu-64	6.07E5	6.88E5
Zn-65	7.47E8	8.59E8
Br-83	4.87E3	7.08E3
Br-84	2.03E5	2.36E5
Rb-86	8.99E6	1.03E7
Rb-88	3.31E4	3.78E4
Rb-89	1.23E5	1.48E5
Sr-89	2.16E4	2.51E4
Sr-91	2.15E6	2.51E6
Sr-92	7.77E5	8.63E5
Y-90	4.49E3	5.31E3
Y-91m	1.00E5	1.16E5
Y-91	1.07E6	1.21E6
Y-92	1.80E5	2.14E5
Y-93	1.83E5	2.51E5
Zr-95	2.45E8	2.84E8
Zr-97	2.96E6	3.44E6
Nb-95	1.37E8	1.61E8
Mo-99	3.98E6	4.62E6
Tc-99m	1.84E5	2.11E5
Tc-101	2.04E4	2.26E4
Ru-103	1.08E8	1.26E8

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Ground Plane Pathway

(M^2 mrem/yr) per (µCi/sec)

Nuclide	Total Body	Skin
Ru-105	6.36E5	7.21E5
Ru-106	4.22E8	5.07E8
Ag-110m	3.44E9	4.01E9
Te-125m	1.55E6	2.13E6
Te-127m	9.16E4	1.08E5
Te-127	2.98E3	3.28E3
Te-129m	1.98E7	2.31E7
Te-129	2.6[illegible]	3.10E4
Te-131m	8.[illegible]	9.46E6
Te-131	2.9[illegible]	3.45E4
Te-132	4.23E6	4.98E6
I-130	5.51E6	6.69E6
I-131	1.72E7	2.09E7
I-132	1.23E6	1.45E6
I-133	2.45E6	2.98E6
I-134	4.47E5	5.30E5
I-135	2.51E6	2.93E6
Cs-134	6.86E9	8.00E9
Cs-136	1.53E8	1.74E8
Cs-137	1.03E10	1.20E10
Cs-138	3.59E5	4.10E5
Ba-139	1.06E5	1.19E5
Ba-140	2.05E7	2.35E7
Ba-141	4.15E4	4.73E4
Ba-142	4.44E4	5.06E4
La-140	1.92E7	2.18E7
La-142	7.40E5	8.89E5
Ce-141	1.37E7	1.54E7
Ce-143	2.31E6	2.63E6
Ce-144	6.96E7	8.04E7

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Ground Plane Pathway

(M^2 mrem/yr) per (μCi/sec)

Nuclide	Total Body	Skin
Pr-144	1.84E3	2.11E3
Nd-147	8.41E6	1.01E7
W-187	2.36E6	2.74E6
Np-239	1.71E6	1.98E6

(a) Data from Reg. Guide 1.109, Appendix E.

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

MEAT PATHWAY

(m^2 mrem/yr) per (μCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	ND	2.34E2	2.34E2	2.34E2	2.34E2	2.34E2	2.34E2
Be-7	7.37E3	1.26E4	8.06E3	ND	1.23E4	ND	7.00E5
C-14	3.83E8	7.67E7	7.67E7	7.67E7	7.67E7	7.67E7	7.67E7
Na-24	1.78E-3	1.78E-3	1.78E-3	1.78E-3	1.78E-3	1.78E-3	1.78E-3
P-32	7.41E9	3.47E8	2.86E8	ND	ND	ND	2.05E8
Cr-51	ND	ND	8.79E3	4.88E3	1.33E3	8.91E3	4.66E5
Mn-54	ND	8.01E6	2.13E6	ND	2.25E6	ND	6.72E6
Mn-56	ND	0	0	ND	0	ND	0
Fe-55	4.57E8	2.42E8	7.51E7	ND	ND	1.37E8	4.49E7
Fe-59	3.76E8	6.09E8	3.03E8	ND	ND	1.76E8	6.34E8
Co-57	ND	5.92E6	1.20E7	ND	ND	ND	4.85E7
Co-58	ND	1.64E7	5.02E7	ND	ND	ND	9.58E7
Co-60	ND	6.93E7	2.04E8	ND	ND	ND	3.84E8
Ni-63	2.91E10	1.56E9	9.91E8	ND	ND	ND	1.05E8
Ni-65	0	0	0	ND	ND	ND	0
Cu-64	ND	2.97E-7	1.79E-7	ND	7.17E-7	ND	1.39E-5
Zn-65	3.75E8	1.00E9	6.22E8	ND	6.30E8	ND	1.76E8
Zn-69	0	0	0	ND	0	ND	0
Br-82	ND	ND	1.52E3	ND	ND	ND	ND
Br-83	ND	ND	ND	ND	ND	ND	ND
Br-84	ND	ND	ND	ND	ND	ND	ND
Br-85	ND	ND	ND	ND	ND	ND	ND
Rb-86	ND	5.82E8	3.58E8	ND	ND	ND	3.74E7
Rb-88	ND	0	0	ND	ND	ND	0
Rb-89	ND	0	0	ND	ND	ND	0
Sr-89	4.82E8	ND	1.38E7	ND	ND	ND	1.86E7
Sr-90	1.04E10	ND	2.64E9	ND	ND	ND	1.40E8
Sr-91	2.40E-10	ND	0	ND	ND	ND	5.29E-10
Sr-92	0	ND	0	ND	ND	ND	0
Y-90	1.71E2	ND	4.59E0	ND	ND	ND	4.88E5

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

MEAT PATHWAY

(m^2 mrem/yr) per (µCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Y-91m	0	ND	0	ND	ND	ND	0
Y-91	1.80E6	ND	4.82E4	ND	ND	ND	2.40E8
Y-92	0	ND	0	ND	ND	ND	0
Y-93	0	ND	0	ND	ND	ND	1.55E-[illegible]
Zr-95	2.66E6	5.85E5	5.21E5	ND	8.38E5	ND	6.11E8
Zr-97	3.20E-5	4.63E-6	2.73E-6	ND	6.65E-6	ND	7.02E-1
Nb-95	3.09E6	1.20E6	8.61E5	ND	1.13E6	ND	2.23E9
Mo-99	ND	1.15E5	2.84E4	ND	2.46E5	ND	9.51E4
Tc-99m	0	0	0	ND	0	0	0
Tc-101	0	0	0	ND	0	0	0
Ru-103	1.55E8	ND	5.96E7	ND	3.90E8	ND	4.01E9
Ru-105	0	ND	0	ND	0	ND	0
Ru-106	4.44E9	ND	5.54E8	ND	5.99E9	ND	5.90E10
Ag-110m	8.40E6	5.67E6	4.53E6	ND	1.06E7	ND	6.75E8
Cd-109	ND	1.90E6	8.83E4	ND	1.70E6	ND	6.18E6
Sb-124	2.92E7	3.79E5	1.02E7	6.45E4	ND	1.62E7	1.83E8
Sb-125	2.85E7	2.20E5	5.97E6	2.64E4	ND	1.59E7	6.80E7
Te-125m	5.69E8	1.54E8	7.59E7	1.60E8	ND	ND	5.49E8
Te-127m	1.77E9	4.78E8	2.11E8	4.24E8	5.06E9	ND	1.44E9
Te-127	4.11E-10	1.11E-10	0	2.85E-10	1.17E-9	ND	1.61E-8
Te-129m	1.79E9	4.99E8	2.77E8	5.76E8	5.25E9	ND	2.18E9
Te-129	0	0	0	0	0	ND	0
Te-131m	7.00E2	2.42E2	2.58E2	4.98E2	2.34E3	ND	9.82E3
Te-131	0	0	0	0	0	ND	0
Te-132	2.09E6	9.26E5	1.12E6	1.35E6	8.60E6	ND	9.33E6
I-130	3.04E-6	6.13E-6	3.16E-6	6.76E-4	9.17E-6	ND	2.87E-6
I-131	1.66E7	1.66E7	9.46E6	5.50E9	2.73E7	ND	1.46E8
I-132	0	0	0	0	0	ND	0
I-133	6.16E-1	7.61E-1	2.88E-1	1.41E2	1.27E0	ND	3.07E-1
I-134	0	0	0	0	0	ND	0

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (k_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

MEAT PATHWAY

(m^2 mrem/yr) per (μCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
I-135	0	0	0	0	0	ND	0
Cs-134	9.22E8	1.51E9	3.19E8	ND	4.69E8	1.68E8	8.16E6
Cs-136	1.61E7	4.43E7	2.86E7	ND	2.36E7	3.51E6	1.56E6
Cs-137	1.33E9	1.28E9	1.88E8	ND	4.16E8	1.50E8	7.99E6
Cs-138	0	0	0	ND	0	0	0
Ba-139	0	0	0	ND	0	0	0
Ba-140	4.38E7	3.84E4	2.56E6	ND	1.25E4	2.29E4	2.22E7
Ba-141	0	0	0	ND	0	0	0
Ba-142	0	0	0	ND	0	0	0
La-140	5.69E-2	1.99E-2	6.70E-3	ND	ND	ND	5.54E2
La-142	0	0	0	ND	ND	ND	0
Ce-141	2.22E4	1.11E4	1.64E3	ND	4.85E3	ND	1.38E7
Ce-143	3.17E-2	1.72E1	2.49E-3	ND	7.21E-3	ND	2.52E2
Ce-144	2.32E6	7.26E5	1.24E5	ND	4.02E5	ND	1.89E8
Pr-143	3.35E4	1.00E4	1.66E3	ND	5.44E3	ND	3.61E7
Pr-144	0	0	0	ND	0	ND	0
Nd-147	1.17E4	9.50E3	7.35E2	ND	5.21E3	ND	1.50E7
Hf-181	4.76E6	1.73E7	2.15E6	ND	3.52E6	ND	6.40E9
W-187	3.35E-2	1.98E-2	8.91E-3	ND	ND	ND	2.79E0
Np-239	4.20E-1	3.02E-2	2.12E-2	ND	8.72E-2	ND	2.23E3

(a) The child age group; data from Reg. Guide 1.109, Appendix E, and UE Safety Analysis Calculation 88-002-00-F.

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	ND	1.57E3	1.57E3	1.57E3	1.57E3	1.57E3	1.57E3
Be-7	7.50E3	1.27E4	8.19E3	ND	1.25E4	ND	7.12E5
C-14	1.19E9	2.39E8	2.39E8	2.39E8	2.39E8	2.39E8	2.39E8
Na-24	8.89E6	8.89E6	8.89E6	8.89E6	8.89E6	8.89E6	8.89E6
P-32	7.77E10	3.64E9	3.00E9	ND	ND	ND	2.15E9
Cr-51	ND	ND	1.03E5	5.65E4	1.56E4	1.04E5	5.40E6
Mn-54	ND	2.10E7	5.59E6	ND	5.88E6	ND	1.76E7
Mn-56	ND	1.29E-2	2.90E-3	ND	1.56E-2	ND	1.86E0
Fe-55	1.12E8	5.93E7	1.84E7	ND	ND	3.35E7	1.10E7
Fe-59	1.20E8	1.94E8	9.69E7	ND	ND	5.64E7	2.02E8
Co-57	ND	3.84E6	7.76E6	ND	ND	ND	3.15E7
Co-58	ND	1.21E7	3.71E7	ND	ND	ND	7.07E7
Co-60	ND	4.32E7	1.27E8	ND	ND	ND	2.39E8
Ni-63	2.96E10	1.59E9	1.01E9	ND	ND	ND	1.07E8
Ni-65	1.66E0	1.56E-1	9.01E-2	ND	ND	ND	1.91E1
Cu-64	ND	7.46E4	4.51E4	ND	1.80E5	ND	3.50E6
Zn-65	4.13E9	1.10E10	6.85E9	ND	6.94E9	ND	1.93E9
Zn-69	0	0	0	ND	0	ND	1.12E-9
Br-82	ND	ND	1.15E8	ND	ND	ND	ND
Br-83	ND	ND	ND	ND	ND	ND	ND
Br-84	ND	ND	ND	ND	ND	ND	ND
Br-85	ND	ND	ND	ND	ND	ND	ND
Rb-86	ND	8.80E9	5.41E9	ND	ND	ND	5.66E8
Rb-88	ND	0	0	ND	ND	ND	0
Rb-89	ND	0	0	ND	ND	ND	0
Sr-89	6.62E9	ND	1.89E8	ND	ND	ND	2.56E8
Sr-90	1.12E11	ND	2.83E10	ND	ND	ND	1.51E9
Sr-91	1.30E5	ND	4.92E3	ND	ND	ND	2.80E5
Sr-92	2.18E0	ND	8.75E-2	ND	ND	ND	4.13E1
Y-90	3.22E2	ND	8.62E0	ND	ND	ND	9.17E5

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Y-91m	0	ND	0	ND	ND	ND	0
Y-91	3.90E4	ND	1.04E3	ND	ND	ND	5.20E6
Y-92	2.5[illegible]-4	ND	7.24E-6	ND	ND	ND	7.31E0
Y-93	1.05E0	ND	2.90E-2	ND	ND	ND	1.57E4
Zr-95	3.83E3	8.42E2	7.50E2	ND	1.21E3	ND	8.79E5
Zr-97	1.92E0	2.77E-1	1.64E-1	ND	3.98E-1	ND	4.20E4
Nb-95	3.18E5	1.24E5	8.84E4	ND	1.16E5	ND	2.29E8
Mo-99	ND	8.14E7	2.01E7	ND	1.74E8	ND	6.73E7
Tc-99m	1.32E1	2.59E1	4.29E2	ND	3.76E2	1.32E1	1.47E4
Tc-101	0	0	0	ND	0	0	0
Ru-103	4.28E3	ND	1.65E3	ND	1.08E4	ND	1.11E5
Ru-105	3.82E-3	ND	1.39E-3	ND	3.36E-2	ND	2.49E0
Ru-106	9.24E4	ND	1.15E4	ND	1.25E5	ND	1.44E6
Ag-110m	2.09E8	1.41E8	1.13E8	ND	2.63E8	ND	1.68E10
Cd-109	ND	3.86E6	1.79E5	ND	3.45E6	ND	1.25E7
Sb-124	1.08E8	1.41E6	3.81E7	2.40E5	ND	6.03E7	6.79E8
Sb-125	9.70E7	6.71E5	1.83E7	8.06E4	ND	4.85E7	2.08E8
Te-125m	7.38E7	[illegible].00E7	9.84E6	2.07E7	ND	ND	7.12E7
Te-127m	2.[illegible]8E8	5.60E7	2.47E7	4.97E7	5.93E8	ND	1.68E8
Te-127	3.05E3	8.22E2	6.54E2	2.11E3	8.67E3	ND	1.19E5
Te-129m	2.71E8	7.57E7	4.21E7	8.74E7	7.96E8	ND	3.31E8
Te-129	0	0	0	0	2.90E-9	ND	6.17E-8
Te-131m	1.60E6	5.53E5	5.89E5	1.14E6	5.35E6	ND	2.24E7
Te-131	0	0	0	0	0	ND	0
Te-132	1.02E7	4.52E6	5.46E6	6.58E6	4.20E7	ND	4.55E7
I-130	1.73E6	3.49E6	1.80E6	3.84E8	5.22E6	ND	1.63E6
I-131	1.30E9	1.31E9	7.45E8	4.33E11	2.15E9	ND	1.17E8
I-132	6.02E-1	1.11E0	5.08E-1	5.13E1	1.69E0	ND	1.30E0
I-133	1.74E7	2.15E7	8.13E6	3.99E9	3.58E7	ND	8.66E6
I-134	0	0	0	0	0	ND	0

TABLE 5 (Cont'd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (μCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
I-135	5.40E4	9.72E4	4.60E4	8.61E6	1.49E5	ND	7.40E4
Cs-134	2.26E10	3.72E10	7.84E9	ND	1.15E10	4.13E9	2.00E8
Cs-136	1.01E9	2.77E9	1.79E9	ND	1.48E9	2.20E8	9.74E7
Cs-137	3.22E10	3.09E10	4.56E9	ND	1.01E10	3.62E9	1.93E8
Cs-138	0	0	0	ND	0	0	0
Ba-139	1.89E-7	0	5.48E-9	ND	0	0	1.09E-5
Ba-140	1.17E8	1.03E5	6.84E6	ND	3.34E4	6.12E4	5.93E7
Ba-141	0	0	0	ND	0	0	0
Ba-142	0	0	0	ND	0	0	0
La-140	1.95E1	6.80E0	2.29E0	ND	ND	ND	1.90E5
La-142	0	0	0	ND	ND	ND	2.90E-6
Ce-141	2.19E4	1.09E4	1.62E3	ND	4.78E3	ND	1.36E7
Ce-143	1.87E2	1.02E5	1.47E1	ND	4.26E1	ND	1.49E6
Ce-144	1.62E6	5.09E5	8.66E4	ND	2.82E5	ND	1.33E8
Pr-143	7.19E2	2.16E2	3.57E1	ND	1.17E2	ND	7.75E5
Pr-144	0	0	0	ND	0	ND	0
Nd-147	4.45E2	3.61E2	2.79E1	ND	1.98E2	ND	5.71E5
Hf-181	6.44E2	2.35E3	2.90E2	ND	4.75E2	ND	8.65E5
W-187	2.91E4	1.73E4	7.73E3	ND	ND	ND	2.42E6
Np-239	1.72E1	1.23E0	8.68E-1	ND	3.57E0	ND	9.14E4

(a) The child age group; data from Reg. Guide 1.109, Appendix E and UE Safety Analysis Calculation 88-002-00-F.

TABLE 5 (Contd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES [a]

Grass-Goat-Milk Pathway

(m^2 mrem/yr) per (μCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	ND	3.20E3	3.20E3	3.20E3	3.20E3	3.20E3	3.20E3
C-14	1.19E9	2.39E8	2.39E8	2.39E8	2.39E8	2.39E8	2.39E8
Na-24	1.07E6	1.07E6	1.07E6	1.07E6	1.07E6	1.07E6	1.07E6
P-32	9.33E10	4.37E9	3.60E9	ND	ND	ND	2.58E9
Cr-51	ND	ND	1.23E4	6.78E3	1.87E3	1.25E4	6.48E5
Mn-54	ND	2.52E6	6.70E5	ND	7.06E5	ND	2.11E6
Mn-56	ND	1.54E-3	3.49E-4	ND	1.87E-3	ND	2.24E-1
Fe-55	1.45E6	7.71E5	2.39E5	ND	ND	4.36E5	1.43E5
Fe-59	1.56E6	2.53E6	1.26E6	ND	ND	7.33E5	2.63E6
Co-58	ND	1.45E6	4.45E6	ND	ND	ND	8.49E6
Co-60	ND	5.18E6	1.53E7	ND	ND	ND	2.87E7
Ni-63	3.56E9	1.90E8	1.21E8	ND	ND	ND	1.28E7
Ni-65	1.99E-1	1.87E-2	1.09E-2	ND	ND	ND	2.29E0
Cu-64	ND	8.31E3	5.02E3	ND	2.01E4	ND	3.90E5
Zn-65	4.96E8	1.32E9	8.22E8	ND	8.33E8	ND	2.32E8
Zn-69	0	0	0	ND	0	ND	1.35E-10
Br-83	ND	ND	ND	ND	ND	ND	ND
Br-84	ND	ND	ND	ND	ND	ND	ND
Br-85	ND	ND	ND	ND	ND	ND	ND
Rb-86	ND	1.06E9	6.50E8	ND	ND	ND	6.80E7
Rb-88	ND	0	0	ND	ND	ND	0
Rb-89	ND	0	0	ND	ND	ND	0
Sr-89	1.39E10	ND	3.97E8	ND	ND	ND	5.38E8
Sr-90	2.35E11	ND	5.95E10	ND	ND	ND	3.16E9
Sr-91	2.74E5	ND	1.03E4	ND	ND	ND	6.04E5

TABLE 5 (Contd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES [a]

Grass-Goat-Milk Pathway

(m^2 mrem/yr) per (μCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Sr-92	4.58E0	ND	1.84E-1	ND	ND	ND	8.68E1
Y-90	3.87E1	ND	1.03E0	ND	ND	ND	1.10E5
Y-91m	0	ND	0	ND	ND	ND	0
Y-91	4.68E3	ND	1.25E2	ND	ND	ND	6.24E-5
Y-92	3.04E-5	ND	8.69E-7	ND	ND	ND	8.77E-1
Y-93	1.27E-1	ND	3.48E-3	ND	ND	ND	1.89E3
Zr-95	4.60E2	1.01E2	9.00E1	ND	1.45E2	ND	1.05E5
Zr-97	2.30E-1	3.33E-2	1.96E-2	ND	4.78E-2	ND	5.04E3
Nb-95	3.81E4	1.48E4	1.06E4	ND	1.39E4	ND	2.75E7
Mo-99	ND	9.76E6	2.42E6	ND	2.09E7	ND	8.08E6
Tc-99m	1.59E0	3.11E0	5.15E1	ND	4.52E1	1.58E0	1.77E3
Tc-101	0	0	0	ND	0	0	0
Ru-103	3.14E2	ND	1.98E2	ND	1.29E3	ND	1.33E4
Ru-105	4.58E-4	ND	1.66E-4	ND	4.03E-3	ND	2.99E-1
Ru-106	1.11E4	ND	1.38E3	ND	1.50E4	ND	1.72E5
Ag-110m	2.51E7	1.69E7	1.35E7	ND	3.15E7	ND	2.01E9
Te-125m	8.85E6	2.40E6	1.18E6	2.48E6	ND	ND	8.54E6
Te-127m	2.50E7	6.72E6	2.96E6	5.97E6	7.12E7	ND	2.02E7
Te-127	3.66E2	9.86E1	7.85E1	2.53E2	1.04E3	ND	1.43E4
Te-129m	3.25E7	9.09E6	5.05E6	1.05E7	9.55E7	ND	3.97E7
Te-129	0	0	0	0	0	ND	7.40E-9
Te-131m	1.92E5	6.64E4	7.07E4	1.37E5	6.43E5	ND	2.69E6
Te-131	0	0	0	0	0	ND	0
Te-132	1.23E6	5.42E5	6.55E5	7.90E5	5.04E6	ND	5.46E6
I-130	2.07E6	4.19E6	2.16E6	4.61E8	6.26E6	ND	1.96E6

TABLE 5 (Contd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Grass-Goat-Milk Pathway

(m^2 mrem/yr) per (μCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
I-131	1.56E9	1.57E9	8.94E8	5.20E11	2.58E9	ND	1.40E8
I-132	7.22E-1	1.33E0	6.10E-1	6.15E1	2.03E0	ND	1.56E0
I-133	2.09E7	2.58E7	9.76E6	4.79E9	4.30E7	ND	1.04E7
I-134	0	0	0	0	0	ND	0
I-135	6.48E4	1.17E5	5.52E4	1.03E7	1.79E5	ND	8.88E4
Cs-134	6.79E10	1.11E11	2.35E10	ND	3.45E10	1.24E10	6.01E8
Cs-136	3.03E9	8.32E9	5.38E9	ND	4.43E9	6.61E8	2.92E8
Cs-137	9.67E10	9.26E10	1.37E10	ND	3.02E10	1.09E10	5.80E8
Cs-138	0	0	0	ND	0	0	0
Ba-139	2.27E-8	0	0	ND	0	0	1.31E-6
Ba-140	1.41E7	1.23E4	8.20E5	ND	4.01E3	7.34E3	7.12E6
Ba-141	0	0	0	ND	0	0	0
Ba-142	0	0	0	ND	0	0	0
La-140	2.34E0	8.17E-1	2.75E-1	ND	ND	ND	2.28E4
La-142	0	0	0	ND	ND	ND	3.49E-7
Ce-141	2.62E3	1.31E3	1.94E2	ND	5.74E2	ND	1.63E6
Ce-143	2.25E1	1.22E4	1.77E0	ND	5.12E0	ND	1.79E5
Ce-144	1.95E5	6.11E4	1.04E4	ND	3.38E4	ND	1.59E7
Pr-143	8.62E1	2.59E1	4.28E0	ND	1.40E1	ND	9.30E4
Pr-144	0	0	0	ND	0	ND	0
Nd-147	5.34E1	4.33E1	3.35E0	ND	2.37E1	ND	6.85E4
W-187	3.49E3	2.07E3	9.27E2	ND	ND	ND	2.90E5
Np-239	2.06E0	1.48E-1	1.04E-1	ND	4.28E-1	ND	1.10E4

(a) The child age group; data from Reg. Guide 1.109, Appendix E.

TABLE 5 (Contd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Vegetation Pathway

(m^2 mrem/yr) per (µCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
H-3	ND	4.01E3	4.01E3	4.01E3	4.01E3	4.01E3	4.01E3
Be-7	3.38E5	5.76E5	3.70E5	ND	5.64E5	ND	3.21E7
C-14	8.89E8	1.78E8	1.78E8	1.78E8	1.78E8	1.78E8	1.78E8
Na-24	3.75E5	3.75E5	3.75E5	3.75E5	3.75E5	3.75E5	3.75E5
P-32	3.37E9	1.57E8	1.30E8	ND	ND	ND	9.30E7
Cr-51	ND	ND	1.17E5	6.50E4	1.78E4	1.19E5	6.21E6
Mn-54	ND	6.65E8	1.77E8	ND	1.86E8	ND	5.38E8
Mn-56	ND	1.88E1	4.24E0	ND	2.27E1	ND	2.72E3
Fe-55	8.01E8	4.25E8	1.32E8	ND	ND	2.40E8	7.8[illegible]E7
Fe-59	3.97E8	6.43E8	3.20E8	ND	ND	1.86E8	6.69E8
Co-57	ND	2.98E7	6.04E7	ND	ND	ND	2.45E8
Co-58	ND	6.44E7	1.97E8	ND	ND	ND	3.76E8
Co-60	ND	3.78E8	1.12E9	ND	ND	ND	2.10E9
Ni-63	3.[illegible]E10	2.11E9	1.34E9	ND	ND	ND	1.42E8
Ni-65	1.05E2	9.89E0	5.77E0	ND	ND	ND	1.21E3
Cu-64	ND	1.10E4	6.64E3	ND	2.66E4	ND	5.16E5
Zn-65	8.12E8	2.16E9	1.35E9	ND	1.36E9	ND	3.80E8
Zn-69	1.09E-5	1.57E-5	1.45E-6	ND	9.52E-6	ND	9.11E-4
Br-82	ND	ND	2.04E6	ND	ND	ND	ND
Br-83	ND	ND	5.37E0	ND	ND	ND	0
Br-84	ND	ND	0	ND	ND	ND	0
Br-85	ND	ND	0	ND	ND	ND	0
Rb-86	ND	4.58E8	2.82E8	ND	ND	ND	2.94E7
Rb-88	ND	0	0	ND	ND	ND	0
Rb-89	ND	0	0	ND	ND	ND	0
Sr-89	3.59E10	ND	1.03E9	ND	ND	ND	1.39E9
Sr-90	1.24E12	ND	3.15E11	ND	ND	ND	1.67E10
Sr-91	5.24E5	ND	1.98E4	ND	ND	ND	1.16E6
Sr-92	7.28E2	ND	2.92E1	ND	ND	ND	1.38E4
Y-90	2.31E4	ND	6.18E2	ND	ND	ND	6.57E7

TABLE 5 (Contd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Vegetation Pathway

(m^2 mrem/yr) per (µCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Y-91m	8.87E-9	ND	3.23E-10	ND	ND	ND	1.74E-5
Y-91	1.86E7	ND	4.99E5	ND	ND	ND	2.48E9
Y-92	1.58E0	ND	4.53E-2	ND	ND	ND	4.58E4
Y-93	3.01E2	ND	8.25E0	ND	ND	ND	4.48E6
Zr-95	3.86E6	8.45E5	7.55E5	ND	1.21E6	ND	8.84E8
Zr-97	5.70E2	8.24E1	4.86E1	ND	1.18E2	ND	1.25E7
Nb-95	4.10E5	1.59E5	1.14E5	ND	1.50E5	ND	2.95E8
Mo-99	ND	7.71E6	1.91E6	ND	1.65E7	ND	6.38E6
Tc-99m	4.71E0	9.24E0	1.53E2	ND	1.34E2	4.69E0	5.26E3
Tc-101	0	0	0	ND	0	0	0
Ru-103	1.54E7	ND	5.90E6	ND	3.87E7	ND	3.97E8
Ru-105	9.16E1	ND	3.32E1	ND	8.05E2	ND	5.98E4
Ru-106	7.45E8	ND	9.30E7	ND	1.01E9	ND	1.16E10
Ag-110m	3.22E7	2.17E7	1.74E7	ND	4.05E7	ND	2.58E9
Cd-109	ND	2.45E8	1.13E7	ND	2.18E8	ND	7.94E8
Sb-124	3.52E8	4.56E6	1.23E8	7.76E5	ND	1.95E8	2.20E9
Sb-125	4.99E8	3.85E6	1.05E8	4.62E5	ND	2.78E8	1.19E9
Te-125m	3.51E8	9.50E7	4.67E7	9.84E7	ND	ND	3.38E8
Te-127m	1.32E9	3.56E8	1.57E8	3.16E8	3.77E9	ND	1.07E9
Te-127	1.00E4	2.69E3	2.14E3	6.91E3	2.84E4	ND	3.90E5
Te-129m	8.38E8	2.34E8	1.30E8	2.70E8	2.46E9	ND	1.02E9
Te-129	1.16E-3	3.23E-4	2.75E-4	8.26E-4	3.39E-3	ND	7.20E-2
Te-131m	1.54E6	5.33E5	5.68E5	1.10E6	5.16E6	ND	2.16E7
Te-131	0	0	0	0	0	ND	0
Te-132	6.98E6	3.09E6	3.73E6	4.50E6	2.87E7	ND	3.11E7
I-130	6.16E5	1.24E6	6.38E5	1.37E8	1.86E6	ND	5.79E5
I-131	1.43E8	1.44E8	8.17E7	4.75E10	2.36E8	ND	1.28E7
I-132	8.38E1	1.58E2	7.25E1	7.31E3	2.41E2	ND	1.86E2
I-133	3.56E6	4.40E6	1.67E6	8.18E8	7.34E6	ND	1.77E6
I-134	1.55E-4	2.88E-4	1.32E-4	6.62E-3	4.40E-4	ND	1.91E-4
I-135	6.62E4	1.13E5	5.33E4	9.97E6	1.70E5	ND	8.58E4

TABLE 5 (Contd.)

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES OTHER THAN NOBLE GASES[a]

Vegetation Pathway

(m^2 mrem/yr) per (µCi/sec)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Cs-134	1.60E10	2.63E10	5.55E9	ND	8.15E9	2.93E9	1.42E8
Cs-136	8.17E7	2.25E8	1.45E8	ND	1.20E8	1.78E7	7.90E6
Cs-137	2.39E10	2.29E10	3.38E9	ND	7.46E9	2.68E9	1.43E8
Cs-138	0	0	0	ND	0	0	0
Ba-139	4.80E-2	2.56E-5	1.39E-3	ND	2.24E-5	1.51E-5	2.77E0
Ba-140	2.77E8	2.42E5	1.62E7	ND	7.89E4	1.45E5	1.40E8
Ba-141	0	0	0	ND	0	0	0
Ba-142	0	0	0	ND	0	0	0
La-140	3.25E3	1.14E3	3.83E2	ND	ND	ND	3.17E7
La-142	2.50E-4	7.98E-5	2.50E-5	ND	ND	ND	1.58E1
Ce-141	6.56E5	3.27E5	4.86E4	ND	1.43E5	ND	4.08E8
Ce-143	1.72E3	9.31E5	1.35E2	ND	3.91E2	ND	1.36E7
Ce-144	1.27E8	3.98E7	6.78E6	ND	2.21E7	ND	1.04E10
Pr-143	1.46E5	4.38E4	7.25E3	ND	2.37E4	ND	1.58E8
Pr-144	0	0	0	ND	0	ND	0
Nd-147	7.17E4	5.81E4	4.50E3	ND	3.19E4	ND	9.20E7
Hf-181	4.90E5	1.79E6	2.21E5	ND	3.62E5	ND	6.59E8
W-187	6.47E4	3.83E4	1.72E4	ND	ND	ND	5.38E6
Np-239	2.55E3	1.83E2	1.29E2	ND	5.30E2	ND	1.36E7

(a) The child age group; data from Reg. Guide 1.109, Appendix E, and UE Safety Analysis Calculation 88-002-00-F.

TABLE 5 NOTES

The values presented in Table 5 were calculated according to the methodology and guidance provided in NUREG 0133, Rev. 0 (1978).

Specific parameters utilized are:

Parameter	Value	Reference
SF	0.7	Ref. 9.11.2
f_p	1.0	Ref. 9.8.2
f_s	1.0	Ref. 9.8.2
H	8.0 g/m^3	Ref. 9.8.2
f_L	1.0	Ref. 9.8.5
f_g	0.76	Ref. 9.8.5

The cumulative critical organ doses for a monthly, quarterly or annual evaluation are based on the calculated dose contribution from each specified time period occurring during the reporting period.

3.6 Gaseous Radwaste Treatment System

3.6.1 Technical Specification 3.11.2.4

The VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY would exceed:

a. 0.2 mrad to air from gamma radiation, or
b. 0.4 mrad to air from beta radiation, or
c. 0.3 mrem to any organ of an Individual

3.6.2 Description of the Gaseous Radwaste Treatment System

The gaseous radwaste treatment system and the ventilation exhaust system are available for use whenever gaseous effluents require treatment prior to being released to the environment. The gaseous radwaste treatment system is designed to allow for the retention of all gaseous fission products to be discharged from the reactor coolant system. The retention system consists of eight (8) waste gas decay tanks, six (6) for use during normal operations and two (2) for use during shutdown conditions. These systems will provide reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept ALARA.

3.6.3 OPERABILITY of the Gaseous Radwaste Treatment System

The OPERABILITY of the gaseous radwaste treatment system ensures this system will be available for use when gases require treatment prior to their release to the environment. OPERABILITY is demonstrated through compliance with Technical Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

Projected doses (gamma air, beta air, and organ dose) due to gaseous effluents at or beyond the SITE BOUNDARY are determined each 31 days by dividing the cumulative annual total by the number of elapsed months.

4.0 DOSE AND DOSE COMMITMENT FROM URANIUM FUEL CYCLE SOURCES

4.1 Technical Specification 3.11.4

The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

4.2 Calculation of Dose and Dose Commitment from Uranium Fuel Cycle Sources

The annual dose or dose commitment to a MEMBER OF THE PUBLIC for Uranium Fuel Cycle Sources is determined as:

a) Dose to the total body and internal organs due to gamma ray exposure from submersion in a cloud of radioactive noble gases, ground plane exposure, and direct radiation from the Unit and outside storage tanks;

b) Dose to the skin due to beta radiation from submersion in a cloud of radioactive noble gases, and ground plane exposure;

c) Thyroid dose due to inhalation and ingestion of radioiodines; and

d) Organ dose due to inhalation and ingestion of radioactive material.

It is assumed that total body dose from sources of gamma radiation irradiates internal body organs at the same numerical rate. (Ref. 9.12.5)

The dose from gaseous effluents is considered to be the summation of the dose at the individual's residence and the dose to the individual from activities within the SITE BOUNDARY.

Since the doses via liquid releases are very conservatively evaluated, there is reasonable assurance that no real individual will receive a significant dose from radioactive liquid release pathways. Therefore, only doses to individuals via airborne pathways and doses resulting from direct radiation are considered in determining compliance to 40 CFR 190. (Ref. 9.12.3)

It should be noted that there are no other Uranium Fuel Cycle Sources within 8km of the Callaway Plant.

4.2.1 Identification of the MEMBER OF THE PUBLIC

The MEMBER OF THE PUBLIC is considered to be a real individual, including all persons not occupationally associated with the Callaway Plant, but who may use portions of the plant site for recreational or other purposes not associated with the plant. (Ref. 9.4 and 9.8.11.) Accordingly, it is necessary to characterize this individual with respect to his utilization of areas both within and at or beyond the SITE BOUNDARY and identify, as far as possible, major assumptions which could be reevaluated if necessary to demonstrate continued compliance with 40 CFR 190 through the use of more realistic assumptions. (Ref. 9.12.3 and 9.12.4)

The evaluation of Total Dose from the Uranium Fuel Cycle should consider the dose to two Critical Receptors: a) The Nearest Resident, and b) The Critical Receptor within the SITE BOUNDARY.

4.2.2 Total Dose to the Nearest Resident

The dose to the Nearest Resident is due to plume exposure from noble gases, ground plane exposure, and inhalation and ingestion pathways. It is conservatively assumed that each ingestion pathway (meat, milk, and vegetation) exists at the location of the Nearest Resident.

It is assumed that direct radiation dose from operation of the Unit and outside storage tanks, and dose from gaseous effluents due to activities within the SITE BOUNDARY, is negligible for the Nearest Resident. The total Dose from the Uranium Fuel Cycle to the Nearest Resident is calculated using the methodology discussed in Section 3, using concurrent meteorlogical data for the location of the Nearest Resident with the highest value of X/Q.

The location of the Nearest Resident in each meteorlogical sector is determined from the Annual Land Use Census conducted in accordance with the Requirements of Technical Specification 3.12.2.

| 4.2.3 Total Dose to the Critical Receptor Within
| the SITE BOUNDARY

The Union Electric Company has entered into an agreement with the State of Missouri Department of Conservation for management of the residual lands surrounding the Callaway Plant, including some areas within the SITE BOUNDARY. Under the terms of this agreement, certain areas have been opened to the public for low intensity recreational uses (hunting, hiking, sightseeing, etc.) but recreational use is excluded in an area immediately surrounding the plant site (Refer to Figure 4.1). Much of the residual lands within the SITE BOUNDARY are leased to area farmers by the Department of Conservation to provide income to support management and development costs. Activities conducted under these leases are primarily comprised of farming (animal feed), grazing, and forestry. (Ref 9.7.2, 9.7.4, 9.14, 9.14.1).

Based on the utilization of areas within the SITE BOUNDARY, it is reasonable to assume that the critical receptor within the SITE BOUNDARY is a farmer, and that his dose from activities within the SITE BOUNDARY is due to exposure incurred while conducting his farming activities. The current tenant has estimated that he spends approximately 1100 hours per year working in this area (Ref 9.5.6). Occupancy of areas within the SITE BOUNDARY is assumed to be averaged over a period of one year.

Any reevaluation of assumptions should include a reevaluation of the occupancy period at the locations of real exposure (e.g. a real individual would not simultaneously exist at each point of maximum exposure).

4.2.3.1 <u>Total Dose to the Farmer from Gaseous Effluents</u>

The Total Dose to the farmer from gaseous effluents is calculated using the methodology discussed in Section 3, utilizing concurrent meteorlogical data at the farmer's residence and historical meteorlogical data from Table 10 for activities within the SITE BOUNDARY. These dispersion parameters were calculated by assuming that the farmer's time is equally distributed over the areas farmed within the SITE BOUNDARY.

The residence of the current tenant is located at a distance of 2380 meters in the SE sector. No meat or milk animals or vegetable gardens were identified by the 1987 Land Use Census for this location, therefore, the gaseous effluents dose at the farmer's residence is due to plume exposure from Noble Gases and the ground plane and inhalation pathways.

It is assumed that food ingestion pathways do not exist within the SITE BOUNDARY, therefore the gaseous effluents dose within the SITE BOUNDARY is due to plume exposure from Noble Gases and the ground plane and inhalation pathways.

4.2.3.2 Total Dose from Direct Radiation

4.2.3.2.1 Direct Radiation Dose from Outside Storage Tanks

The Refueling Water Storage Tank (RWST) has the highest potential for receiving significant amounts of radioactive materials, and constitutes the only potentially significant source of direct radiation dose from outside storage tanks to a MEMBER OF THE PUBLIC. (Ref. 9.6.17, 9.6.18, 9.6.19, and 9.6.20.)

Direct radiation dose from the RWST to a MEMBER OF THE PUBLIC is determined at the nearest point of the Owner Controlled Area fence which is not obscured by significant plant structures. This has been determined to be 450 meters from the RWST.

| The RWST is a right circular cylinder approximately 12 meters in diameter, 14 meters in height with a capacity of approximately 1,514,000 liters. (Ref. 9.6.20.) The walls are of type 304 stainless steel and have an average thickness of .87 cm. (Ref. 9.16.1.)

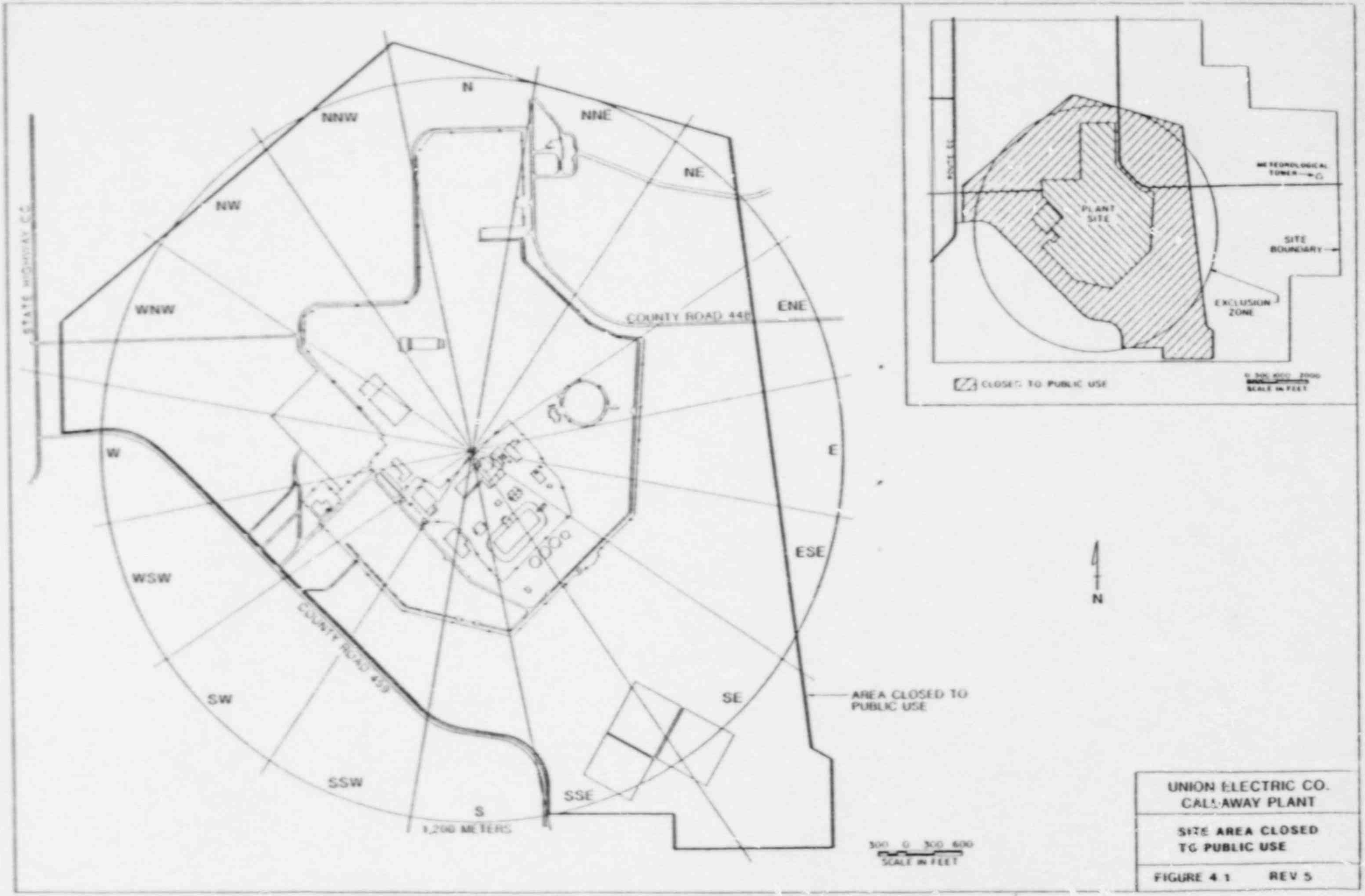
| The direct radiation dose from the RWST is calculated based on the tank's average isotopic content and the parameters discussed above, considering buildup and attenuation within the volume source. Appropriate methodology for calculating the dose rate from a volume source is given in TID-7004, "Reactor Shielding Design Manual" (Ref. 9.19). The computer program ISOSHLD (Ref. 9.20, 9.21, 9.22) will normally be utilized to perform this calculation.

|

4.2.3.2.2 Direct Radiation Dose from the Reactor

The maximum direct radiation dose from the Unit to a MEMBER OF THE PUBLIC has been determined to be 7E-2 mrads/calendar year, based on a point source of primary coolant N-16 in the steam generators. This source term was then projected onto the inside surface of the containment dome, taking credit for shielding provided by the containment dome and for distance attenuation. No | credit was allowed for shielding by other structures or | components. The number of gammas per second was generated and then converted to a dose rate at the given distance by use of ANSI/ANS-6.6.1, "Calculation and Measurement of Direct and Scattered Gamma Radiation | from LWR Nuclear Power Plant 1979", which considers attenuation and buildup in air. The final value is based on one unit operating at 100% Power. The distance was determined to be 367 meters, which is approximately the closest point of the boundary of the Owner Controlled Area fence which is not obscurred by significant plant structures. (Ref. 9.16.4)

The maximum direct radiation dose from the Unit to a MEMBER OF THE PUBLIC due to activities within the SITE BOUNDARY is thus approximately 9E-3 mrads per year, assuming a maximum occupancy of 1100 hours per year.



UNION ELECTRIC CO.
CALLAWAY PLANT

SITE AREA CLOSED TO PUBLIC USE

FIGURE 4.1 REV 5

5.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

5.1 Radiological Effluent Technical Specification 3.12.1

The radiological environmental monitoring program shall be conducted as specified in Technical Specification Table 3.12-1.

5.2 Description of the Radiological Environmental Monitoring Program

The Radiological Enviormental Monitoring Program is intended to act as a background data base for preoperation and to supplement the radiological effluent release monitoring program during plant operation. Radiation exposure to the public from the various specific pathways and direct radiation can be adequately evaluated by this program.

Some deviations from the sampling frequency may be necessary due to seasonal unavailability, hazardous conditions, or other legitimate bases. Efforts are made to obtain all required samples within time frame outlines. Any deviation(s) in sampling frequency or location is documented in the Annual Radiological Environmental Operating Report.

The Environmental samples are collected and analyzed at the frequency outlined in Table 6. Reporting levels and lower limits of detection (LLD) are given in Tables 7 and 8.

Airborne, waterbore, and ingestion samples collected under the monitoring program are analyzed by an independent, third-party laboratory. This laboratory is required to participate in the Environmental Protection Agency's (EPA) Environmental Radioactivity Laboratory Intercomparison Studies (Crosscheck) Program or an equivalent program. Participation includes all of the determinations (sample medium - radionuclide combination) that are offered by the EPA and that are also included in the monitoring program.

5.3 Performance Testing of Environmental Thermoluminescence Dosimeters

Thermoluminescence Detectors (TLD's) used in the Environmental Monitoring Program are tested for accuracy and precision to demonstrate compliance with Regulatory Guide 4.13. (Ref. 9.18).

Energy dependence is tested at several energies between 30keV and 3MeV corresponding to the approximate energies of the predominant Noble Gases (80, 160, 200 keV), Cs-137 (662 keV), Co-60 (1225 keV), and at least one energy less than 80 keV. Other testing is performed relative to either Cs-137 or Co-60.

TABLE 6

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation	40 routine monitoring station either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, plaed as follows: An inner ring of si - -en stations, one in each meteorological sector in the general area of the SITE BOUNDARY.	At least once per 92 days	Gamma Dose

Station Code	Sector	Site Description	Location
04	A	0.3 Miles East of Hwy O and CC Junction, Callaway Electric Coop. Utility Pole No. 18892	1.9 mi. @ 349° N
47	B	County Road 448, 0.9 Miles South of Hwy O, Callaway Electric Coop. Utility Pole No. 28151	0.9 mi. @ 17° NNE
48	C	County Road 448, 1.5 Miles South of Hwy O, Plant Security Area Sign Post	0.4 mi. @ 45° NE
05	D	Primary Meteorological Tower	1.3 mi. @ 78° ENE
49	E	County Road 448, Callaway Electric Coop Utility Pole No. 06959. Reform Wildlife Management Parking Area	1.7 mi. @ 98° E
52	F	Light Pole Near East Plant Security Fence	0.4 mi. @ 114° ESE
51	G	Located in the "Y" of the abandoned Railroad spur, northwest of sludge lagoon	0.7 mi. @ 137° SE
50	H	County Road 459, 3.3 Miles North of Hwy 94, Callaway Electric Coop. Utility Pole No. 35086	0.9 mi. @ 163° SSE
07	J	County [illegible] 459, 2.6 Miles North of Hwy 94, Callaway Electric [illegible]op. Utility Pole No. 35097	1.3 mi. @ 181° S
37	K	County Road 459, 0.9 Miles South of Hwy CC, Callaway Electric Coop. Utility Pole No. 35077	0.7 mi. @ 202° SSW
43	L	County Road 459, 0.7 Miles South of Hwy CC, Callaway Electric Coop Utility Pole No. 35073	0.5 mi. @ 230° SW
44	M	Highway CC, 1.0 Miles South of County Road 459, Callaway Electric Coop. Utility Pole No. 18769	1.7 mi. @ 257° WSW

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Station Code	Sector	Site Description	Location
06	N	County Road 428, 1.2 Miles West of Hwy CC, Callaway Electric Coop. Utility Pole No. 18609	2.0 mi. @ 277° W
45	P	County Road 428, 0.1 Miles West of Hwy CC, Callaway Electric Coop. Utility Pole No. 18580	1.4 mi. @ 290° WNW
03	Q	0.1 Miles West of Hwy CC on Gravel Road 0.8 Miles South Hwy O, Callaway Electric Coop. Utility Pole No. 18559	1.3 mi. @ 308° NW
46	R	North-East side of Hwy CC and County Road 446 Intersection Callaway Electric Coop. Utility Pole No. 28242	1.5 mi @ 333° NNW
An outer ring of sixteen stations, one in each meteorological sector in the 6- to 8-km range from the site.			
36	A	County Road 155, 0.8 Miles South of County Road 132, Callaway Electric Coop. Utility Pole No. 19137	5.2 mi. @ 7° N
21	B	County Road 155, 1.9 miles north of Hwy O, Callaway Electric Coop. Utility Pole No. 19100	4.0 mi @ 23° NNE
20	C	Highway D, 0.4 Miles North of Hwy K, Callaway Electric Coop. Utility Pole No. 12830	4.8 mi. @ 47° NE
18	D	Highway D, 0.4 Miles South of Hwy O, Callaway Electric Coop. Utility Pole No. 12952	3.8 mi @ 63° ENE
17	E	County Road 4053, 0.3 Miles East of Hwy D, Kingdom Telephone Company Pole No. 3 X 12	4.0 mi. @ 89° E
14	F	South-East Side of Hwy 94 and Hwy D Intersection Callaway Electric Coop. Utility Pole No. 11940	5.0 mi @ 121° ESE
11	G	City of Portland, Callaway Electric Coop. Utility Pole No. 12112	4.8 mi. @ 139° SE
10	H	Highway 94, 1.8 Miles East of County Road 459 Callaway Electric Coop. Utility Pole No. 12182	4.0 mi. @ 157° SSE
09	J	North-West Side of Hwy 94 and County Road 459 Junction, Callaway Electric Coop. Utility Pole No. 06754	3.7 mi. @ 183° S

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Station Code	Sector	Site Description	Location
30	K	West Side of County Road 447 at the Junction with County Road 463, Kingdom Telephone Company Pole No. 2K1	4.6 mi. @ 208° SSW
42	L	County Road 447, 2.6 Miles North of County Road 463, Callaway Electric Coop. Utility Pole No. 06326	4.4 mi. @ 233° SW
32	M	Highway VV, 0.6 Miles West of County Road 447, Callaway Electric Coop. Utility Pole No. 27031	5.4 mi. @ 251° WSW
41	N	Higway AD, 2.8 Miles East of Hwy C, Callaway Electric Coop. Utility Pole No. 18239	4.8 mi. @ 279° W
40	P	North-East Side of County Road 112 and Hwy O Junction Callaway Electric Coop. Utility Pole No. 06326	4.2 mi. @ 294° WNW
39	Q	County Road 112, 0.7 Miles East of County Road 111 Callaway Electric Coop. Utility Pole No. 17516	5.4 mi. @ 315° NW
38	R	County Road 133, 1.5 Miles South of Hwy UU Callaway Eleictric Coop. Utility Pole No. 34708	4.8 mi. @ 337° NNW
Eight Stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.			
33	N	City of Hams Prairie, South-East of the Hwy C and Hwy AD Junction	7.4 mi. @ 273° W
31	L	City of Mokane, Callaway Electric Coop. Utility Pole No. 06039	7.4 mi. @ 218° SW
26	E	Town of Americus, Cllaway Electric Coop. Utility Pole No. 11159	12.1 mi. @ 82° E
27	F	Town of Bluffton, Callaway Electric Coop. Utility Pole No. 11496	9.6 mi. @ 110° ESE
35	R	City of Toledo, Callaway Electric Coop. Utility Pole No. 17684	5.8 mi. @ 342° NNW
23	B	City of Yucatan, Callaway Electric Coop. Utility Pole No. 12670	6.8 mi. @ 16° NNE

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Station Code	Sector	Site Description	Location
11	G	City of Portland, Callaway Electric Coop. Utility Pole No. 12112	4.8 mi. @ 139° SE
20	C	City of Readsville, Callaway Electric Coop. Utility Pole No. 12830	4.8 mi. @ 47° NE
34 (P-Control)	P	North-East Side of Hwy C and County Road 408 Junction	9.7 mi. @ 293° WNW
01 (Q-Control)	Q	Highway Z, 0.8 Miles East of Business 54, Callaway Electric Coop. Utility Pole No. 21544	11.0 mi. @ 312° NW

2. Airborne

Radioiodine and Particulates	Samples from five locations	Continuous operations of sampler with sample collection as required by dust loading, but at least once per 7 days.	**Radioiodine Canister:** Analyze at least once per 7 days for I-131.
	Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground level D/Q.		**Particulate Sample:** Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic analysis[d] on those samples for which the gross beta activity is >10 times the yearly mean of control samples. Perform gamma isotopic analysis[d] on composite samples (by location) at least once per 92 days.

Station Code	Sector	Site Description	Location
A1	D	Primary Meteorological Tower	1.3 mi. @ 78° ENE
A8	B	County Road 448, 1.0 Miles South of Hwy O	0.8 mi @ 24° NNE
B3	A	0.3 Miles East of Hwy O and Hwy CC Junction	1.9 mi. @ 349° N

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

	Station Code	Sector	Site Description			Location
	One sample from the community with the highest D/Q.					
	29	R	Community of Reform			1.7 mi. @ 338° NNW
	One sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.					
	A7	Q	C. Bartley Farm			9.5 mi. @ 312° NW
3. Waterborne						
a. Surface[e]	One sample upstream			Composite sample[f] over a period of less than or equal to 31 days.	Gamma isotopic analysis[d] of each sample. Tritium analysis of composite sample at least once per 92 days.	
	S01	H	84 feet upstream of discharge, north bank			4.8 mi. @ 144° SE
	One sample downstream					
	S02	G	1.1 miles downstream of discharge, north bank			5.2 mi. @ 133° SE
b. Drinking	One sample of each one to three of the nearest water supplies within 10 miles downstrea, that could be affected by its discharge. One sample from a control location.			Composite sample over 2-week period when I-131 analysis is performed, monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year. Composite for gross beta and gamma isotopic analysis monthly. Composite for tritium analysis quarterly.	
	As there are no drinking water intakes within 10 miles downstream of the discharge point, the drinking water pathway is currently not included as part of the Callaway Plant Radiological Environmental Monitoring Program. Should future water intakes be constructed within 10 river miles downstream of the discharge point, then the program will be revised to include this pathway. (Ref. 9.6.6)					
c. Sediment from Shoreline	One sample from downstream area with existing or potential recreational value.			Semiannually	Gamma isotopic analysis (d) semiannually.	

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Station Code	Sector	Site Description	Location
C	G	1.0 river mile downstream of discharge, north bank	5.1 mi. @ 135° SE

4. Ingestion

a. Milk

Samples from milking animals in three location within 5 km distance having the highest dose potential[9]. If there are none, then one sample from milking animals in each of three areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per year

Semimonthly when animals are on pasture, monthly at other times.

Gamma isotopic[4] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

Due to a lack of milk animals which satisfy these reqirements, the milk pathway is currently not included as a part of the Callaway Plant Radiological Environmental Monitoring Program. Should the Annual Land Use Census identify the existence of milking animals in locations which satisfy these requirements, then the program will be revised to include this pathway.

b. Fish

One sample of each commercially and recreationally important species in vicinity of plant discharge area.

Sample in season, or semiannylally if they are not seasonal.

Gamma isotopic analysis[4] on edible portions.

Station Code	Sector	Site Description	Location
C	G	1.0 river mile downstream of discharge, north bank	5.1 mi. @ 135° SE

One sample of same species in areas not influenced by plant discharge.

A	H	0.6 river miles upstream of discharge, north bank	4.9 mi. @ 154° SSE

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Station Code	Sector	Site Description		Location

c. Food Products

One sample of each principal class of products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest (h)	Gamma isotopic analysis[d] on edible portion.

As there are no areas irrigated by water in which liquid plant wastes have been discharged within 50 river miles downstream of the discharge point, this sample type is not currently included as part of the Callaway Plant Radiological Environmentla Monitoring Program. Should future irrigation water intakes be constructed within 10 river miles downstream of the discharge point, then the program will be revised to include this sample type. (Ref. 9.7.6 and 9.7.7)

Samples of three different kinds of broad leaf vegatation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available	Gamma isotopic[d] and I-131 analysis

Station Code	Sector	Site Description	Location
V6	R	Becker's Farm	1.8 mi. @ 344° NNW
V7	A	Meehan's Farm	1.8 mi. @ 356° N

One sample of each of similar broad leaf vegetation grown 15 to 30 km distant in the least prevalent wind direction (if milk sampling is not performed).	Monthly when available	Gamma Isotopic[d] analysis.

Station Code	Sector	Site Description	Location
V3	L	Beazley's Farm	15.0 mi. @ 227° SW

TABLE 6 (Continued)

TABLE NOTATION

(a) Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunctions, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedual shall be documented in the Annual Radiological Environmental Operating Report. It is recognized that, at times, it may not be possible or practical to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program. Identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

(b) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

(c) The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

(d) Gamma isotopic analysis is defined as the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(e) The "upstream" sample shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond, but near the mixing zone.

(f) In this program, constant volume sample aliquots are collected at time intervals that are short (e.g., monthly).

(g) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(h) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 7

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg), wet	Milk (pCi/l)	Food Product (pCi/kg, wet)
H-3	20,000 *				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zr-Nb-95	400 **				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200 **			300**	

* For drinking water samples. For surface water samples a value of 30,000 pCi/l is used.

** Total activity, parent plus daughter activity.

TABLE 8

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a],[b].

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg), wet	Milk (pCi/l)	Food Product (pCi/kg, wet)	Sediment (pCi/kg, dry)
Gross Beta	4	.01				
H-3	2000 *					
Fe-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zr-Nb-95	15 **					
I-131	1(d)	.07		1	60	
Cs-134	15	.05	130	15	60	150
Cs-137	18	.06	150	18	80	180
Ba-La-140	15 **			15**		

* For surface water samples, a value of 3000 pCi/l is used.

** Total activity, parent plus daughter activity.

TABLE 8 (CONTINUED)
TABLE NOTATION

(a) The LLD is defined for purposes of compliance with the Radiological Effluent Technical Specifications as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ S_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = The lower limit of detection as defined above (as picocurie per unit mass or volume).

S_b = The standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E = The counting efficiency (as counts per disintegration).

V = The sample size (in units of mass or volume).

2.22 = The number of disintegrations per minute per picocurie.

Y = The fractional radiochemical yield (when applicable).

λ = The radioactive decay constant for the particular radionuclide and,

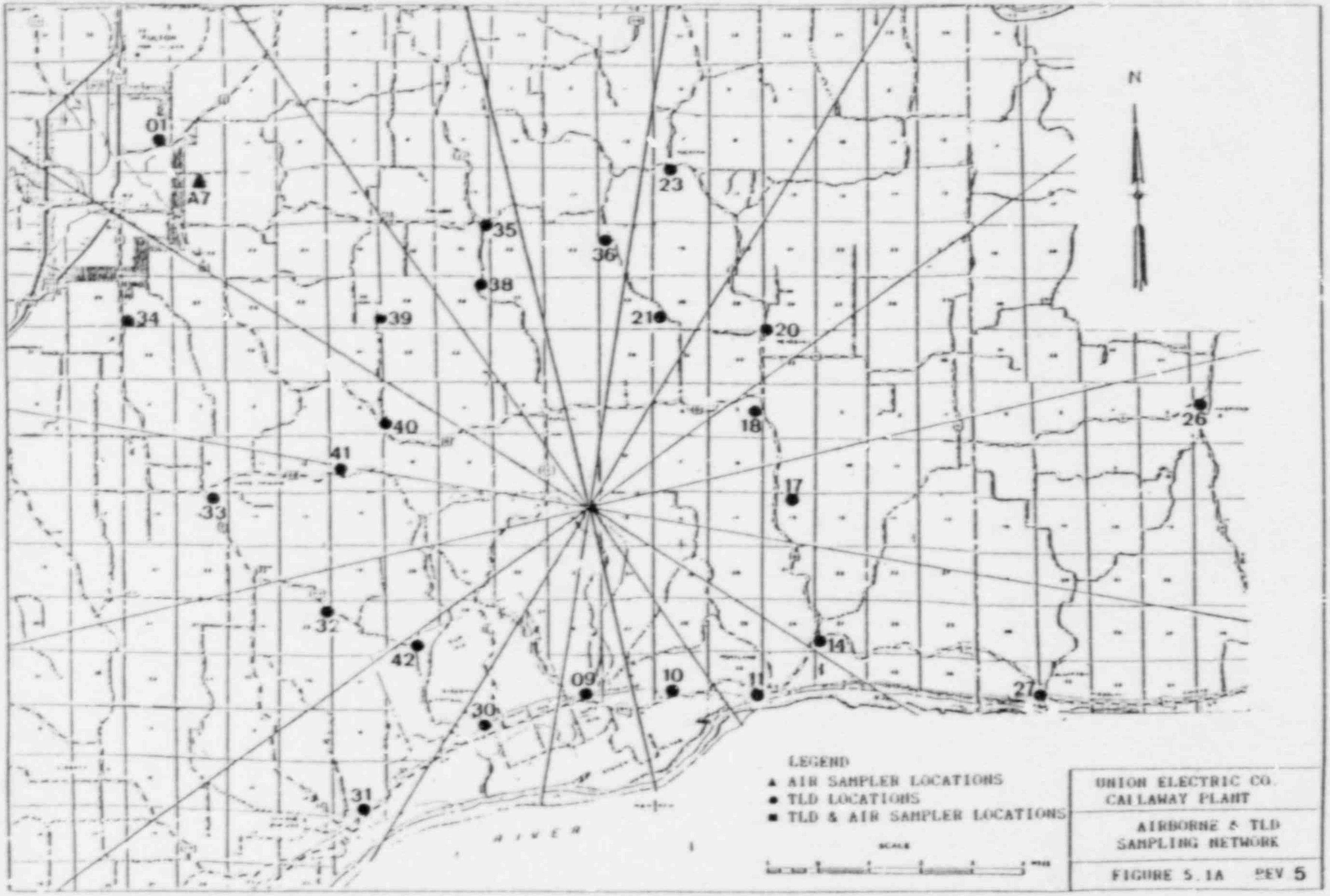
Δt = the elapsed time between sample collection (or end of the smaple collection period) and time of counting (for environmental samples, not plant effluent samples).

Typical values of E, V, Y and Δt shall be used in the calculations.

It should be recognized that the LLD is defined as a *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement. Analyses are performed in such a manner that the stated LLDs are achieved under routine conditions. Occassionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

(b) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be anlayzed and reported in the Annual Radiological Environmental Operating Report.

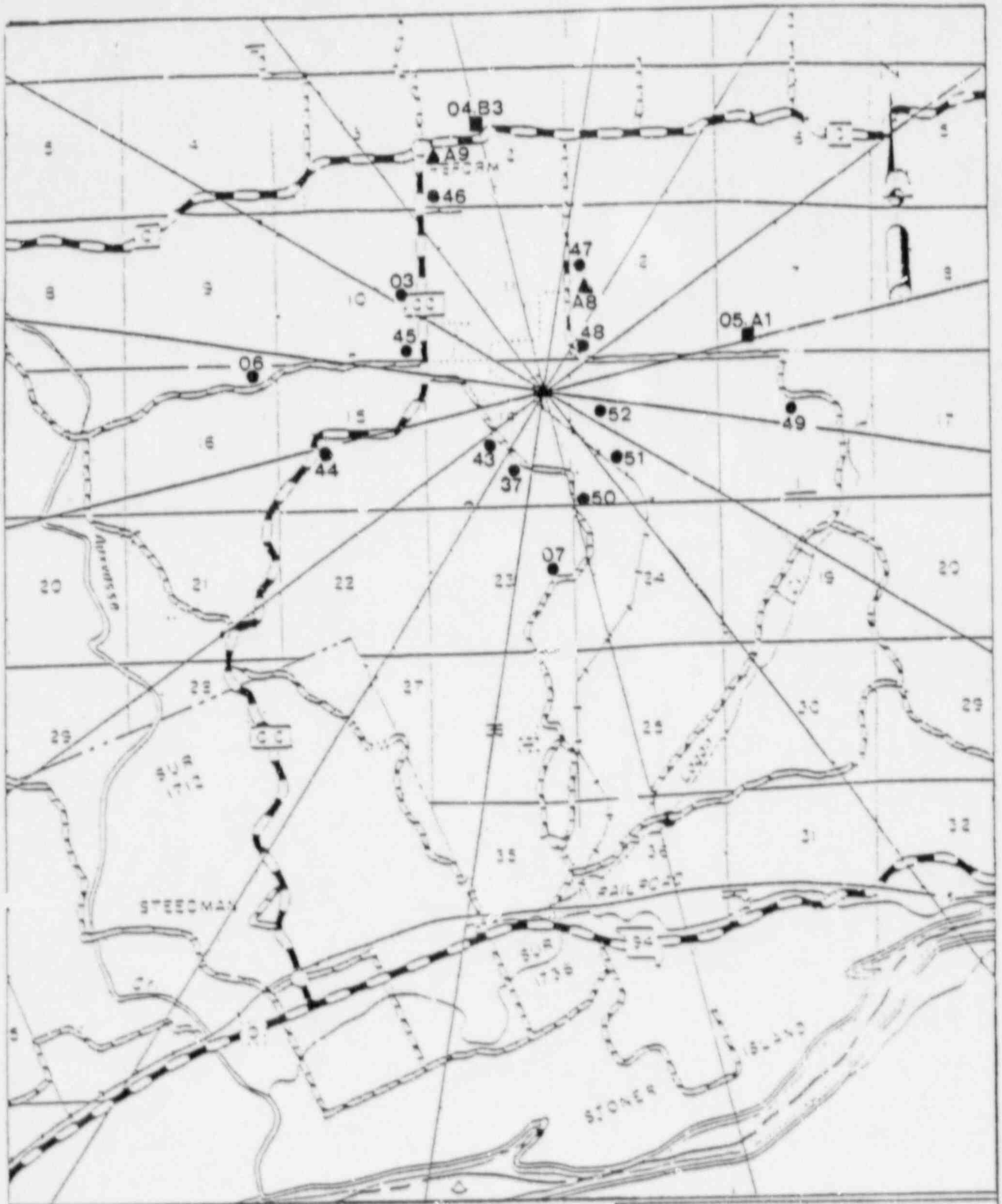
(c) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the reccommendations of Regulatory Guide 4.13, Revision 1, July 1977. (Refer to Section 5.3)



UNION ELECTRIC CO.
CALLAWAY PLANT

AIRBORNE & TLD
SAMPLING NETWORK

FIGURE 5.1A REV 5



LEGEND

- ▲ AIR SAMPLER LOCATIONS
- ● TLD LOCATIONS
- ■ TLD & AIR SAMPLER LOCATIONS

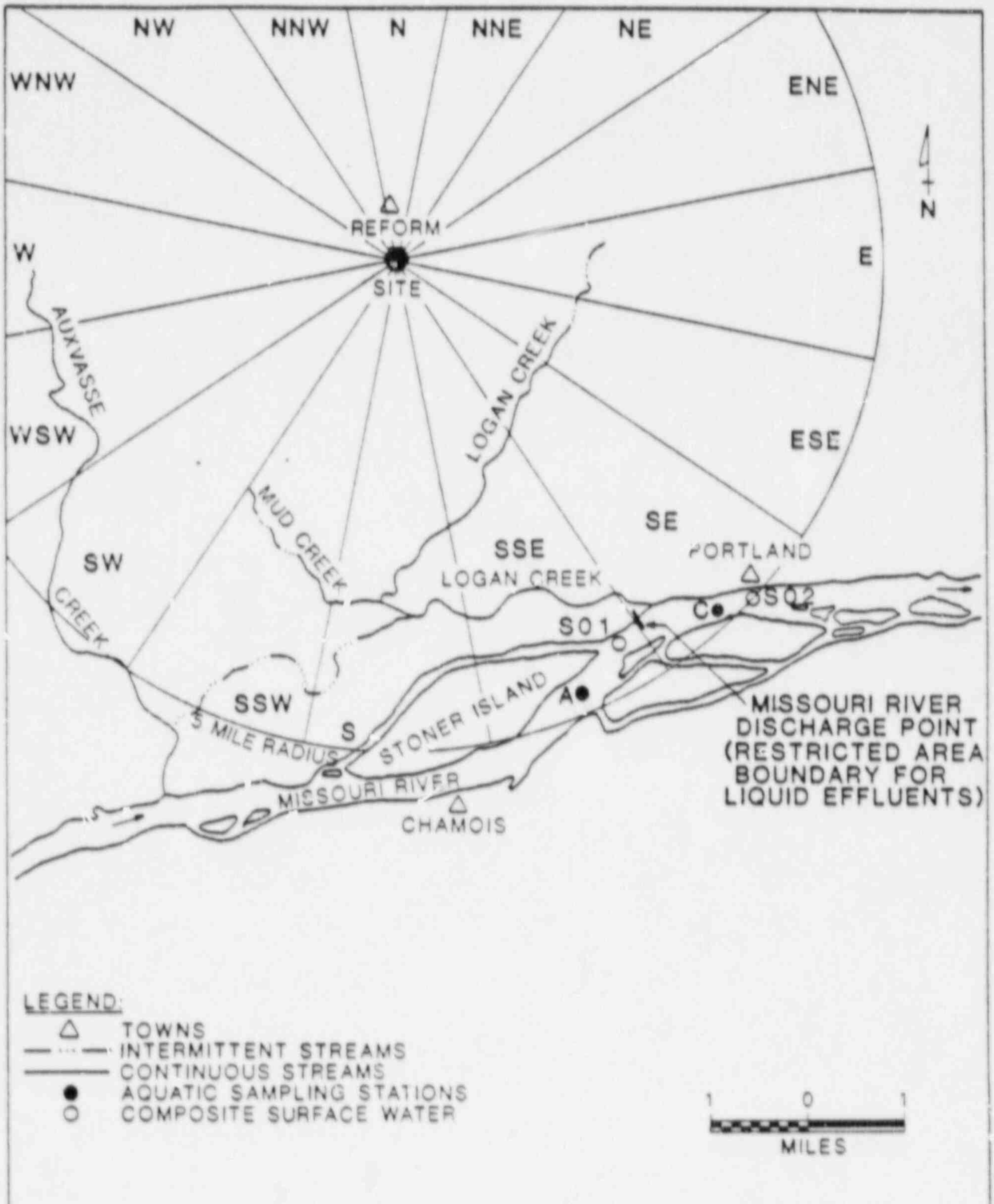
SCALE

1 0 1 2 3

UNION ELECTRIC CO.
CALLAWAY PLANT

AIRBORNE & TLD
SAMPLING NETWORK

FIGURE 5.1B REV. 5



UNION ELECTRIC CO.
CALLAWAY PLANT

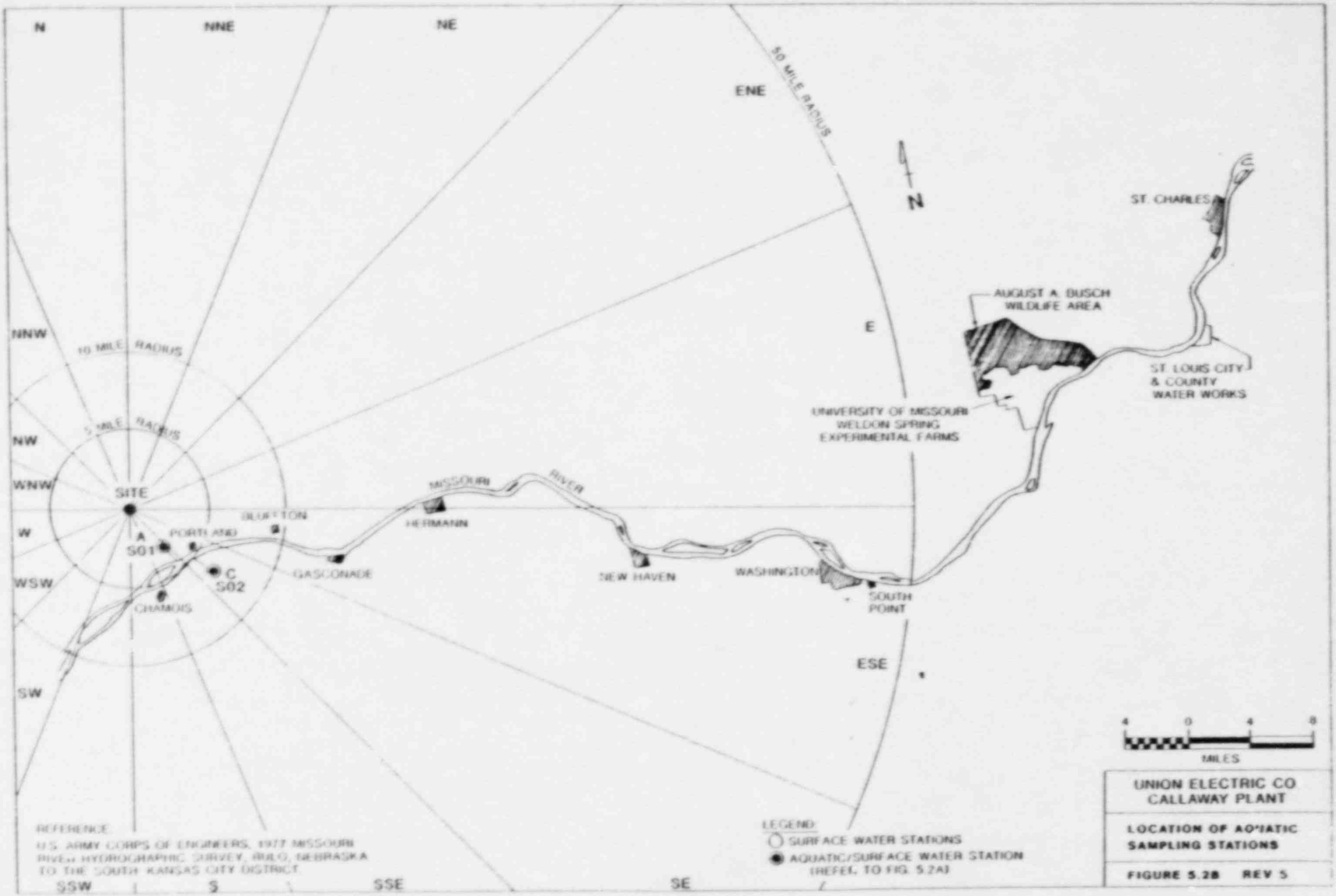
LOCATION OF AQUATIC SAMPLING STATIONS

FIGURE 5.2A REV 5

UNION ELECTRIC CO.
CALLAWAY PLANT

FOOD PRODUCTS SAMPLING LOCATIONS

FIGURE 8.3 REV 5



UNION ELECTRIC CO
CALLAWAY PLANT

LOCATION OF AQUATIC SAMPLING STATIONS

FIGURE 5.2B REV 5

6.0 DETERMINATION OF ANNUAL AVERAGE AND SHORT TERM ATMOSPHERIC DISPERSION PARAMETERS

6.1 Atmospheric Dispersion Parameters

The values presented in Table 9 and Table 10 were determined through the analysis of on-site meterological data collected during the three year period of May 4, 1973 to May 5, 1975 and March 16, 1978 to March 16, 1979.

6.1.1 Long-Term Dispersion Estimates

The PUFF (fluctuating plume) model and the straight-line Gaussian (constant mean wind direction) model were used for determination of the long-term atmospheric dispersion parameters. A more detailed discussion of the methodology and data utilized to calculate these parameters can be found elsewhere (Ref. 9.6.12).

The Unit Vent and Radwaste Building Vent releases are at elevations 66.5 meters and 20 meters above grade, respectively. Both release points are within the building wake of the structures on which they are located, and the Unit Vent is equipped with a rain cover which effectively eliminates the possibility of the exit velocity exceeding five times the horizontal wind speed. All gaseous releases are thus considered to be ground-level releases, and therefore no mixed mode or elevated release dispersion parameters were determined. (Ref. 9.5.2)

6.1.2 Determination of Long-Term Dispersion Estimates for Special Receptor Locations

Calculations utilizing the PUFF model were performed for 22 standard distances to obtain the desired dispersion parameters. Dispersion parameters at the SITE BOUNDARY and at special receptor locations were estimated by logarithmic interpolation according to (Ref. 9.6.14):

$$X = X_1 \left(\frac{d}{d_1}\right)^B \qquad (6.2)$$

Where:

B = $\ln(X_2/X_1)/\ln(d_2/d_1)$.

X_1, X_2 = Atmospheric concentrations at distances d_1 and d_2, respectively, from the source (in Ci/m^3).

The distances d_1 and d_2 were selected such that $d_1 < d < d_2$.

6.1.3 Short Term Dispersion Estimates

Airborne releases are classified as short term if they are less than or equal to 500 hours during a calendar year and not more than 150 hours in any quarter. Short term dispersion estimates are determined by multiplying the appropriate long term dispersion estimate by a correction factor (Ref. 9.9.1 and 9.17.4):

$$F = (T_s/T_a)^S \qquad (6.2)$$

Where:

T_S = The total number of hours of the short term release.

T_a = The total number of hours in the data collection period from which the long term diffusion estimate was determined (Refer to Section 6.1).

Values of the slope factor (S), are presented in TABLE 11.

Short term dispersion estimates are applicable to short term releases which are not sufficiently random in both time of day and duration (e.g., the short term release periods are not dependent solely on atmospheric conditions or time of day) to be represented by the annual average dispersion conditions. (Ref. 9.8.12.)

6.1.3.1 The Determination of the Slope Factor (S).

The general approach employed by subroutine PURGE of XOQDOQ (Ref. 9.17.4) was utilized to produce values of the slope of the (X/Q) curves (Slope Factor (S)) for both the Radwaste Building Vent and the Unit Vent. However, instead of using approximation procedures to produce the 15 precentile (X/Q) values, the 15 percentile (X/Q) value for each release and at each location was determined by ranking all the 1-hour $(X/Q)_1$ values for that release and at the location in descending order. The $(X/Q)_1$ value which corresponded to the 15 percentile of all the calculated (X/Q) values within a sector was extracted for use in the intermittent release (X/Q) calculation.

The intermittent release (X/Q) curve was constructed using the calculated 1-hour 15 percentile $(X/Q)_1$ and its corresponding annual average $(X/Q)_a$. A graphic representation, of how the computational procedure works is illustrated by Figure 4.8 of reference 9.17.4. The straight line connecting these points represents $(X/Q)_i$ values for intermittent releases, ranging in duration from one (1) hour to 8760 hours. The slope (S) of the curve is expressed as:

$$S = \frac{-\log\left((X/Q)_1 / (X/Q)_a \right)}{\log (Ta/T_1)} \qquad (6.3)$$

or

$$S = \frac{-(\log (X/Q)_1 - \log (X/Q)_a)}{\log Ta - \log T_1} \qquad (6.4)$$

TABLE 9

HIGHEST ANNUAL AVERAGE ATMOSPHERIC DISPERSION PARAMETERS (a)

RADWASTE BUILDING VENT

LOCATION (b)	SECTOR	DISTANCE (METERS)	χ/Q	χ/Q Decayed/Undepleted	χ/Q Decayed/Depleted	D/Q
			(sec/m^3)	(sec/m^3)	(sec/m^3)	(m^{-2})
SITE BOUNDARY	S	1300	1.3E-6	1.3E-6	1.2E-6	4.4E-9
Nearest Cow (c)	NW	5053	4.3E-7	4.2E-7	3.3E-7	1.1E-9
Nearest Goat (c)	NW	5053	4.3E-7	4.2E-7	3.3E-7	1.1E-9
Nearest Meat Animal	NNW	2736	7.6E-7	7.6E-7	6.4E-8	2.4E-9
Nearest Vegtable (c) Garden	NNW	2865	8.2E-7	8.1E-7	6.8E-7	2.5E-9
Nearest Residence(c)	NNW	2865	8.2E-7	8.1E-7	6.8E-7	2.5E-9

(a) Values given are from FSAR, Table 2.3-84, and Table 2.3-86

(b) Data from 1987 Land Use Census

(c) Values derived from FSAR, Table 2.3-81, using the methodology presented in Equation (6.1) (Ref. 9.16.2 and 9.16.3)

Building Shape Parameter (C) = 0.5 (Ref. 9.5.4)
Vertical Height of Highest Adjacent Building (V) = 19.96 meters (Ref. 9.5.4)

TABLE 10

HIGHEST ANNUAL AVERAGE ATMOSPHERIC DISPERSION PARAMETERS (a)

UNIT VENT

LOCATION (b)	SECTOR	DISTANCE (METERS)	X/Q	X/Q Decayed/Depleted	X/Q Decayed/Depleted	D/Q
			(sec/m³)	(sec/m³)	(sec/m³)	(m⁻²)
SITE BOUNDARY	S	1300	9.9E-7	9.8E-7	8.8E-7	4.4E-9
Nearest Cow (c)	NW	5053	3.6E-7	3.6E-7	2.8E-7	1.1E-9
Nearest Goat (c)	NW	5053	3.6E-7	3.6E-7	2.8E-7	1.1E-9
Nearest Meat Animal	NNW	2736	5.9E-7	5.9E-7	5.0E-7	2.4E-9
Nearest Vegtable (c) Garden	NNW	2865	6.4E-7	6.3E-7	5.3E-7	2.5E-9
Nearest Residence(c)	NNW	2865	6.4E-7	6.3E-7	5.3E-7	2.5E-9
Areas Within the Site Boundary(d)	NNW	N/A	4.4E-8	4.4E-8	3.9E-8	2.1E-10

(a) Values given are from FSAR, Table 2.3-82, and Table 2.3-85
(b) Data from 1987 Land Use Census
(c) Values derived from FSAR, Table 2.3-83, using the methodology presented in Equation (6.1) (Ref. 9.16.2 and 9.16.3)
(d) Ref. 9.5.7 and 9.5.8.

Building Shape Parameter (C) = 0.5 (Ref. 9.5.4)
Vertical Height of Highest Adjacent Building (V) = 66.45 meters (Ref. 9.5.4)

TABLE 11

SHORT TERM DISPERSION PARAMETERS (a) (c)

Location (b)	Sector	Distance (meters)	Slope Factor(S) Unit Vent	Slope Factor(S) Radwaste Building Vent
Site Boundary	S	1300	-.328	-.320
Nearest Cow	NW	5053	-.263	-.266
Nearest Goat	NW	5053	-.263	-.266
Nearest Meat Animal	NNW	2736	-.262	-.268
Nearest Vegetable Garden	NNW	2865	-.264	-.268
Nearest Residence	NNW	2865	-.264	-.268

(a) Reference 9.5.4
(b) Data from 1987 Land Use Census
(c) Recirculation Factor = 1.0

TABLE 12

APPLICATION OF ATMOSPHERIC DISPERSION PARAMETERS

DOSE PATHWAY	ODCM REFERENCE	DISPERSION PARAMETER	CONTROLLING AGE GROUP	CONTROLLING LOCATION
Noble Gas, Beta Air	3.5.2.1	X/Q, decayed/undepleted	--	Site Boundary
Noble Gas, Gamma Air	3.5.2.1	X/Q, decayed/undepleted	--	Site Boundary
Noble Gas, Total Body	3.4.1 & 3.5.1.1	X/Q, decayed/undepleted	--	Site Boundary
Noble Gas, Skin	3.4.1 & 3.5.1.1	X/Q, decayed/undepleted	--	Site Boundary
Ground Plane Deposition	3.5.2.2.1	D/Q	--	Nearest Resident
Inhalation	3.5.2.2.1	X/Q, decayed/depleted	Child	Nearest Resident
Vegetation	3.5.2.2.1	D/Q*	Child	Nearest Resident
Milk	3.5.2.2.1	D/Q*	Child	Nearest Resident
Meat	3.5.2.2.1	D/Q*	Child	Nearest Resident

*For H-3 and C-14, X/Q, decayed/depleted is used instead of D/Q (Reference 9.11.1).

7.0 SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation are submitted within 60 days after January 1 and July 1 of each year. The period of the first report begins with the date of initial criticality.

The Radioactive Effluent Release Reports include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B is supplemented with three additional categories: class of solid waste (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity), and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year includes an annual summary of hourly meteorological data collected over the previous year which may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation, or in the form of joint frequency distributions of wind speed wind direction, and atmospheric stability.* This same report includes an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report also includes, the asessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, is included in these reports. Acceptable methods for calculating the dose contributions from liquid and gaseous effluents are given in Regulatory Guide 1.109, and the ODCM.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year also includes, as required by Technical Specification 3.11.4, an assessment of radition doses to the likely most exposed MEMBER OF THE PUBLIC from Reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calender year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation".

The Radioactive Effluent Release Reports include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the ODCM, pursuant to Specification 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment System, pursuant to Specification 6.15. It also includes a listing of new locations for dose calculations and or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Radioactive Effluent Release Reports also include the following information: An explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.5, respectively.

*In lieu of submission, the Union Electric Company has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

(Ref.9.4)

8.0 IMPLEMENTATION OF ODCM METHODOLOGY

The ODCM provides the mathematical relationships used to implement the Radiological Effluent Technical Specifications.

For routine effluent release and dose assessment, computer codes are utilized to implement the ODCM methodologies. These codes have been evaluated by a qualified independent reviewer to ensure that they produce results consistent with the methodologies presented in the ODCM. (Ref. 9.5.5)

9.0 REFERENCES

9.1 Title 10, "Energy", Chapter 1, Code of Federal Regulations, Part 20; U.S. Government Printing Office, Washington, D.C. 20402.

9.2 Title 10, "Energy", Chapter 1, Code of Federal Regulations, Part 50, Appendix I; U.S. Government Printing Office, Washington, D.C. 20402

9.3 Title 40, "Protection of Environment", Chapter 1, Code of Federal Regulations, Part 190; U.S. Government Printing Office, Washington, D.C. 20402.

9.4 U.S. Nuclear Regulatory Commission, "Technical Specifications Callaway Plant, Unit NO. 1", NUREG-1058 (Rev. 1), October 1984.

9.5 Communications

9.5.1 Letter NEO-54, D.W. Capone to S.E. Miltenberger, dated January 5, 1983; Union Electric Company correspondence.

9.5.2 Letter BLUE 1285, "Callaway Annual Average X/Q and D/Q Values", J. H. Smith (Bechtel Power Corporation), to D. W. Capone (Union Electric Co.), dated February 27, 1984.

9.5.3 (Reference Deleted)

9.5.4 Letter BLUE 1232, "Callaway Annual Average X/Q Values and "S" Values", J. H. Smith (Bechtel Power Corporation) to D. W. Capone (Union Elecctric Co.), dated February 9, 1984.

9.5.5 Letter BLUE 1358, "Comparison of Callaway Plant Offsite Dose Calculations for Routine Effluents", J.H. Smith (Bechtel Power Corporation) to D.W. Capone (Union Electric Company), dated March 22, 1984.

9.5.6 Private Communication, H.C. Lindeman & B.F. Holderness, August 6, 1986

| 9.5.7 Letter, N. G. Slaten to A. C. Passwater, "Commercial Use of Lands Inside the Callaway Plant Site Boundary", dated August 25, 1987. (UENE Safety Analysis Calculation 87-050-001)

| 9.5.8 Letter, N. G. Slaten to C. C. Graham, "Farming Meteorlogical Parameter, Restricted Area", dated December 14, 1987.

9.6 Union Electric Company Callaway Plant, Unit 1, Final Safety Analysis Report.

9.6.1 Section 11.5.2.2.3.1

9.6.2 Section 11.5.2.2.3.4

9.6.3 Section 11.5.2.1.2

9.6.4 Section 11.5.2.2.3.2

9.6.5 Section 11.5.2.2.3.3

9.6.6 Section 11.2.3.3.4

9.6.7 Section 11.2.3.4.3

9.6.8 Section 11.5.2.3.3.1

9.6.9 Section 11.5.2.3.3.2

9.6.10 Section 11.5.2.3.2.3

9.6.11 Section 11.5.2.3.2.2

9.6.12 Section 2.3.5

| 9.6.13 (Reference Deleted)

9.6.14 Section 2.3.5.2.1.2

9.6.15 (Reference Deleted)

9.6.16 (Reference Deleted)

9.6.17 Section 9.2.5

9.6.18 Section 9.2.7.2.1

9.6.19 Section 6.3.2.2

9.6.20 Table 11.1-6

9.6.21 Tab'` 9.4-6

9.6.22 Table `.4-8

9.6.23 Table 9.4-11

| 9.6.24 Table 9.4-12

| 9.6.25 (Reference Deleted)

9.6.26 `. ble 2.3-68

9.7 Union Electric Company Callaway Plant Environmental Report, Operating License Stage.

9.7.1 Table 2.1-19

9.7.2 Section 2.1.2.3

9.7.3 (Reference Deleted)

9.7.4 Section 2.1.3.3.4

9.7.5 (Reference Deleted)

9.7.6 Section 5.2.4.1

9.7.7 Table 2.1-19

9.8 U.S. Nuclear Regulatory Commission, "Preparation of Radiological Effluent Technical Specification For Nuclear Power Plants", USNRC NUREG-0133, Washington, D.C. 20555, October 1978.

9.8.1 Pages AA-1 th. ugh AA-3

9.8.2 Section 5.3.1.3

9.8.3 Section 4.3

9.8.4 Section 4.3.1

9.8.5 Section 5.3.1.5

9.8.6 Section 5.1.1

9.8.7 Section 5.1.2

9.8.8 Section 5.2.1

9.8.9 Section 5.2.1.1

9.8.10 Section 5.3.1

9.8.11 Section 3.8

9.8.12 Section 3.3

9.9 U.S. Nuclear Regulatory Commission, "XOQDOQ, Program For the Meterological Evaluation Of Routine Effluent Releases At Nuclear Power Stations", USNRC NUREG-0324, Washington, D.C. 20555.

9.9.1 Pages 19-20 Subroutine PURGE

9.10 Regulatory Guide 1.111, "Methods For Estimating Atmospheric Transport And Dispersion of Gaseous Effluents In Routine Releases From Light-Water-Cooled Reactors", Revision 1, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, July, 1977.

9.10.1 Section c.1.b

9.10.2 (Reference Deleted)

9.10.3 Figures 7 through 10

9.10.4 (Reference Deleted)

9.10.5 Section c.4

9.11 Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases Of Reactor Effluents For the Purpose Of Evaluating Compliance With 10 CFR Part 50, Appendix I", Revision 1, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, October 1977.

9.11.1 Appendix C, Section 3.a

9.11.2 Appendix E, Table E-15

9.11.3 Appendix C, Section 1

9.11.4 Appendix E, Table E-11

9.11.5 Appendix E, Table E-9

9.12 U.S. Nuclear Regulatory Commission, "Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190)", USNRC NUREG-0543, Washington, D.C. 20555, January 1980.

9.12.1 Section I, Page 2

9.12.2 Section IV, Page 8

9.12.3 Section IV, Page 9

9.12.4 Section III, Page 6

9.12.5 Section III, Page 8

9.13 (Reference Deleted)

9.14 Management Agreement for the Public Use of Lands, Union Electric Company and the State of Missouri Department of Conservation, December 21, 1982.

9.14.1 Exhibit A

9.15 (Reference Deleted)

9.16 Miscellaneous References

9.16.1 Drawing Number M-109-0007-06, Revision 5.

9.16.2 HPCI 87-01, "Determination of Annual Average Dispersion Parameters at the Owner Controlled Area Fence", January 28, 1987.

9.16.3 HPCI 87-02, "1986 Land Use Census and Dispersion Parameters", January 28, 1987.

9.16.4 UE Safety Analysis Calculation 87-001-00.

9.16.5 UE Safety Analysis Calculation 88-002-00-F.

9.17 U.S. Nuclear Regulatory Commission, "XOQDOQ: Computer Program for the Meterological Evaluation of Routine Effluent Releases at Nuclear Power Stations", USNRC NUREG/CR-2929, September, 1982, Washington, D.C. 20555.

9.17.1 (Reference Delted)

9.17.2 (Reference Deleted)

9.17.3 (Reference Deleted)

9.17.4 Section 4, "Subroutine PURGE", pages 27 and 28.

9.18 Regulatory Guide 4.13, "Performance, Testing, and procedural specifications for Thermoluminiscence Dosimetry: Environmental Applications "(Revision 1), July 1977; USNRC, Washington, D.C. 20555

9.19 TID-7004, "Reactor Shielding Design Manual", Rockwell, Theodore, ed; March 1956.

9.20 BNWL-236, "ISOSHLD - A computer code for General Purpose Isotope Shielding Analysis", Engel, R.C., Greenberg, J., Hendrichson, M.M.; June 1966.

9.21 BNWL-236, Supplement 1, "ISOSHLD-II: Code Revision to include calculation of Dose Rate from Shielded Bremsotrahlung Sources", Simmons, G.L., et al; March 1967.

9.22 BNWL-236, Supplement 2, "A Revised Photon Probability Library for use with ISOSHLD-III", Mansius, C.A.; April 1969.