UNION ELECTRIC COMPANY

CALLAWAY PLANT

OFFSITE DOSE CALCULATION MANUAL

Table of Contents

Table of Contents (continued)

Table of Contents (continued)

Table of Contents (continued)

List of Figures

List of Tables

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 Radiological Effluent 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 Radiological Effluent Technical Specification 3.11.1.1 are not exceeded. The alarm/trip setpoint of these channels shall be determined in accordance with the methodology described 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. 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 Radiological Effluent 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 $F + f \cong F$}.

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 would become:

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

2.2.1 Continuous Liquid Effluent Monitors

The radiation detection monitors associated with continous liquid effluent releases are:

Monitor I.D.	Description
O-BM-RE-52	Steam Generator Blowdown Discharge Monitor
O-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 Radiological Effluent Technical Specification Table 4.11-1.

The steam generator blowdown discharge monitor continuously monitors the blowdown discharge pump outlet to detect radioactivity due to system demineralizer break through and to provide backup to the steam generator blowdown process radioactivity monitor to prevent discharge of radioactive fluid. The sample point is located on the discharge of the pump in order to monitor discharge or recycled blowdown fluid and upstream of the discharge isolation valve to permit termination of the radioactive release prior to exceeding the instantaneous concentration limits of 10 CFR Part 20. The high radioactivity alarm/trip setpoint initiates control room alarm annunciation and automatic isolation of the blowdown isolation valves and the blowdown discharge valve.

The turbine building drain effluent monitor is provided to monitor turbine building liquid effluents prior to release to the environs. The fixed-volume detector assembly continuously monitors the drain effluent line upstream of the drain line isolation valve. The high radioactivity alarm/trip setpoint initiates control room annunciation and automatic isolation of the drain line isolation valve to prevent the release of radioactive fluids. The sample location ensures that all potentially radioactive turbine building liquid effluents are monitored prior to discharge.

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.

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

The alarm/trip setpoint calculations for the steam generator blowdown discharge monitor and the turbine building drain monitor are based on the representative dilution flow rate, the maximum effluent stream flow

rate, and the MPC for Cs-137 as the controlling isotope. A portion of the total site allocation is provided for the value of the radiation monitor setpoint (c), based on local procedures and operating experience. 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:

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

The liquid radwaste radiation monitor continuously monitors the discharge of the liquid radwaste processing system to prevent the discharge of radioactive fluid to the environs. The fixed-volume detector assembly continuously monitors the system discharge line upstream of the discharge valve. The high radioactivity alarm/trip setpoint initiates control room alarm annunciation and automatic isolation of the liquid radwaste system discharge valve to terminate discharge. The sample point is located to ensure that all potentially radioactive fluids from the liquid radwaste processing system are monitored prior to discharge.

The secondary liquid waste system discharge radioactivity monitor continuously monitors secondary liquid waste system effluents prior to discharge to the environs. The fixed-volume detector assembly monitors the discharge line upstream of the discharge isolation valve. The high radioactivity alarm/trip setpoint initiates control room alarm annunciation and automatic isolation of the secondary liquid waste system discharge valve to prevent the discharge of radioactive fluid. The sample location ensures that all potentially radioactive sources from the system are monitored prior to discharge.

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 Radiological Effluent Technical Specification 3.11.1.1 limits are not exceeded at the site unrestricted area boundary.

2.3 ODCM Methodology for the 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.

2.3.1 Development of ODCM Methodology for the Determination of Liquid Effluent Monitor Setpoints

2.3.1.1 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 Radiological Effluent Technical Specifications Table 4.11-1:

$$\sum_i C_i = \Sigma_g C_g + C_a + C_s + C_t \qquad (2.3)$$

Where:

C_i = the concentration of each radionuclide i, as determined by the analysis of the waste sample.

$\Sigma_g C_g$ = 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 of the monthly composite sample.

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 in liquid waste as determined by analysis of the monthly composite sample.

The C_g term is included in the analysis of each batch; terms for alpha, strontium, 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 the Radiological Effluent Technical Specifications.

2.3.1.2 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 Radiological Effluent 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:

$$F_d = \sum_i \frac{C_i}{MPC_i} \div F_s \qquad (2.4)$$

$$= \sum_g \frac{C_g}{MPC_g} + \frac{C_a}{MPC_a} + \frac{C_s}{MPC_s} + \frac{C_t}{MPC_t} \div F_S \qquad (2.5)$$

Where:

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

MPC_i = MPC_g, MPC_s, MPC_a, and MPC_t are limiting concentrations of the appropriate radionuclide 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 $F_s = 0.9$.

2.3.1.3 For the case $F_d < 1$, 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 Radiological Effluent 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.

$$F_{dn} = F_d \div F_a \tag{2.6}$$

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 Radiological Effluent Technical Specification 3.11.1.1.

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

$$F_a = \frac{1}{n} \tag{2.7}$$

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/n. Prior to initiating a simultaneous release, a check must be made to assure that the sum of the allocation factors assigned to pathways for the simultaneous release is ≤ 1.

2.3.1.5 The calculated maximum permissable 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:

$$F_{eff} = (0.9)F_e \qquad (2.8)$$

Where:

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

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

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

$$f_{max} \leq \frac{F_{eff} + f_p}{F_{dn}} \sim \frac{F_{eff}}{F_{dn}} \qquad (\text{for } f_p << F_{eff}) \qquad (2.9)$$

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 Radiological Effluent 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 Radiological Effluent 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.

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

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

$$c = A \sum_g C_g \frac{\mu Ci}{ml} \qquad \text{(Refer to Note Following)} \qquad (2.10)$$

Where:

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

$$A = \frac{f_{max}}{f_p} \qquad \text{(Refer to Note Following)} \qquad (2.11)$$

If A > 1: Calculate c and determine the maximum value for the actual monitor setpoint (μCi/ml).

If A ≤ 1: No release may be made. This condition must be flagged and the operator instructed to reevaluate 2.3.1.3 and 2.3.1.5 (i.e., reduce effluent flow rate or return radwaste for reprocessing).

NOTE

If F_d < 1, no further dilution is required and the release may be made without regard to available dilution or to other releases made simultaneously. However, it is necessary to establish a monitor setpoint which will provide alarm should the release concentration inadvertently exceed Radiological Effluent Technical Specification 3.11.1.1 limits. This can be accomplished by establishing the adjustment factor as follows:

$$A = \frac{1}{F_d} \qquad (2.12)$$

2.3.2 Summary, Setpoint Calculation Methodology for Liquid Effluent Monitors

The methodology described in 2.3.1 is used to determine setpoints for each of the radiation monitors assigned a

liquid process function. The limiting release concentration can be increased by reducing the discharge flow-rate and decreased by reducing the cooling tower blowdown flow-rate.

2.4 Liquid Effluent Concentration Measurements

2.4.1 Radiological Effluent Technical Specification 3.11.1.1

The concentration of radioactive material released from the site 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 μCi/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 will be 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 be 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 analysis of the liquid batch sample. For alpha, Sr-89, Sr-90 and H-3, the measured concentration from the previous composite analysis is used. Composite samples are collected for each batch release and monthly and quarterly analyses are performed in accordance with Table 4.11-1 of the Radiological Effluent Technical Specifications.

Dose contributions from liquids discharged as continuous releases are determined by utilizing the last measured values of samples required in accordance with Radiological Effluent Technical Specifications Table 4.11-1.

2.5 Individual Dose Due to Liquid Effluents

2.5.1 Radiological Effluent Technical Specification 3.11.1.2

The dose or dose commitment to an individual from radioactive materials in liquid effluents released from the site shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total 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 whole 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.

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, 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%.

Thus, the maximum exposed individual is an adult, receiving 95% of the total dose from eating fish and 5% of the total dose from recreational activities.

2.5.3 ODCM Methodology for Determining Dose Contributions 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 monthly (at least one each 31 days) and a cumulative summation of these total body and any organ dose 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:

$$D_\tau = \sum_i [A_{i\tau} \sum_{\ell=1}^{m} \Delta t_\ell \, C_{i\ell} \, F_\ell] \qquad (2.13)$$

Where:

D_τ = the cumulative dose commitment to the whole 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 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 Table 4.11-1, Radiological Effluent Technical Specifications, (in mrem/hr) per (μCi/ml). These factors are given in Table 1, as derived through the use of Equation (2.17).

$F\ell$ = the near field average dilution factor for $C_{i\ell}$ during any liquid effluent release. Defined as the ratio of the maximum undiluted liquid waste flow during release to the product of the average flow from the site discharge structure to unrestricted receiving waters times 89.77. (89.77 is the site specific applicable factor for the mixing effect of the discharge structure.)

The term $C_{i\ell}$ is the composite undiluted concentration of radioactive material in liquid waste at the common release point determined from the Radioactive Liquid Waste Sampling and Analysis Program, Table 4.11-1 in the Radiological Effluent Technical Specifications. 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 (2.13) 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 will be used from Table E-11 of Regulatory Guide 1.109; thus the list contains critical organ dose factors for various organs. The dose factor is calculated according to:

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

Where:

$A_{i\tau}$ = composite dose parameter for the whole 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 (Regulatory 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) (Regulatory Guide 1-109, Table E-11).

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. Therefore, it is not necessary to evaluate (U_w/D_w) at this time, and Equation (2.14) simplies to:

$$A_{i\tau} = k_o \, (U_F BF_i) DF_i \qquad (2.15)$$

Inserting the appropriate usage factors from Regulatory Guide 1.109 into Equation (2.15) yields the following expression:

$$A_{i\tau} = 1.14E05 \, (21BF_i) DF_i \qquad (2.16)$$

or

$$A_{i\tau} = 2.39E06 \times BF_i \times DF_i \qquad (2.17)$$

2.5.4 Summary, Determination of Individual Dose Due to Liquid Effluents

The dose contribution for the total time period

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

is determined by calculation for each monthly period (at least once per 31 days) and a cumulative summation of these total body and organ doses is maintained for each calendar quarter. The projected dose contribution from batch releases for which radionuclide concentrations are determined by periodic composite and grab sample analysis, as stated in Table 4.11-1 of the Radiological Effluent Technical Specifications, may be approximated by using the last measured value. However, for reporting purposes, the calculated dose contribution from those radionuclides is based on actual composite/grab sample analysis. Dose contributions are determined for all radionuclides identified in liquid effluents released to unrestricted areas. Nuclides which are below the LLD for the analyses are reported as "less than" the nuclide's LLD, 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 (uCi/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
C-14	3.13E+04	6.26E+03	6.26E+03	6.26E+03	6.26E+03	2.26E+03	6.26E+03
N-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	2.81-01	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-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	4.66E+04
ZN-69	4.93E+01	9.44E+01	6.56E+00	No Data	6.13E+01	No Data	1.42E+01
BR-84	No Data	No Data	5.26E+01	No Data	No Data	No Data	4.13E-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	6.68E+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	2.25E-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
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
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	6.57E+01	No Data	8.32E+00	No Data	1.27E+02	No Data	4.25E+03

TABLE 1 (continued)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
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
LA-140	1.50E-01	7.53E-02	1.99E-02	No Data	No Data	No Data	5.53E+03
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	1.09E+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
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
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
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
W	1.2 E + 03
Np	1.0 E + 01

(a) Values taken from Regulatory Guide 1.109, Rev 1, Table A-1.

2.6 Liquid Radwaste Treatment System

2.6.1 Radiological Effluent Technical Specification 3.11.1.3

The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid waste prior to their discharge when the projected doses due to the liquid effluent from the site when averaged over the calendar quarter, would exceed 0.18 mrem to the total body or 0.6 mrem to any organ.

2.6.2 Description of the Liquid Radwaste Treatment System

2.6.3 Operability of the Liquid Radwaste Treatment System

The liquid radwaste 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. To determine operability requirements, doses due to liquid releases are projected once each 31 days in accordance with the methodology described in Section 2.5.3.

3.0 GASEOUS EFFLUENTS

3.1 Radiological Effluent 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 Radiological Effluent Technical Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

3.2 Radiological Effluent Technical Specification 3.11.2.1

The dose rate due to radioactive materials released in gaseous effluents from the site 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 all radioiodines and for all radioactive materials in particulate form and radionuclides (other than noble gases) 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, iodine monitors, and particulate monitors are present on the containment building ventilation system, plant unit ventilation system, and radwaste building ventilation system.

The alarm/trip setpoint for any gaseous effluent radiation monitor is determined based on the instantaneous concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 1, and are applied at the point at which the discharge leaves the restricted area boundary into an unrestricted area.

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.

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. Table 4.3-13 of the Radiological Effluent Technical Specifications provides the instrument surveillance requirements, such as calibration, source checking, functional testing, and channel checking.

3.3.1 Continuous Release Gaseous Effluent Monitors

The radiation detection monitors associated with continuous gaseous effluent releases are:

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

The Unit Vent monitor continuously monitors the effluent from the unit vent for particulate, iodine (halogen), and 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.

All of these systems are filtered before they exhaust to the unit vent. The unit vent monitor measures actual plant effluents and not inplant concentrations. Thus, the system continuously monitors downstream of the last point of potential radioactivity entry. The monitoring system consists of an off-line, three-way airborne radioactivity monitor. An isokinetic sampling probe is located downstream of the last point of potential radioactivity entry for sample collection.

The sample extracted by the isokinetic nozzle is passed through the fixed filter (particulate), charcoal filter (iodine), and fixed-volume (gaseous) detector assemblies and then through the pumping system for discharge back to the unit vent. Indication is provided on the

radioactivity monitoring system CRT in the control room.

The Radwaste Building Ventilation effluent monitor continuously monitors for particulate, halogen, and gaseous radioactivity in the effluent duct downstream of the exhaust filter and fans. The sample point is located downstream of the last possible point of radioactive influent, including the waste gas decay tank discharge line. 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.

The monitoring system consists of a fixed filter particulate monitor, an iodine monitor, and gaseous activity monitor.

The sample is extracted through an isokinetic nozzle to ensure that a representative sample of the air is obtained prior to release to the environment. After passing through the fixed filter (particulate), charcoal filter (halogen), and fixed-volume (noble gas) detector assemblies and the pumping system, the sample is discharged back to the exhaust duct. Indication is provided on the radiation monitoring system CRT in the control room.

This monitor will isolate the waste gas decay tank discharge line if the radioactivity release rate is above the present limit when the waste gas discharge valve has been deliberately or inadvertently opened.

The continuous gaseous effluent monitor setpoints are established using Xe-133 as the controlling isotope. Since there are two continuous gaseous effluent release points, a fraction of the total MPC will be allocated to each release point. Neglecting the batch releases, the plant Unit Vent monitor has been allocated 0.7 MPC and the Radwaste Building Vent monitor has been allocated 0.3 MPC. These will be changed as required, but limited to 1 MPC of Xe-133. Therefore, a particular monitor reaching the fractional MPC setpoint would not necessarily mean the MPC limit at the site boundary is being exceeded; the alarm only indicates that the specific release point is contributing a greater fraction of the MPC limit than was allocated to the associated monitor and will constitute an evaluation of both systems.

3.3.2 Batch Release Gaseous Monitors

The radiation monitors associated with batch release gaseous effluents are:

Monitor I.D.	Description
0-GT-RE-22 0-GT-RE-33	Containment Purge System Monitors
0-GT-RE-31 0-GT-RE-32	Containment Atmosphere Radioactivity Monitors
0-GH-RE-10	Radwaste Building Vent

The Containment Purge System continuously monitors the containment purge exhaust duct during purge operations for particulate, iodine, and gaseous radioactivity. The purpose of these monitors is to isolate the containment purge system on high gaseous activity via the ESFAS. These monitors also serve as backup indication for personnel protection and reactor coolant pressure boundary leakage detection for the containment atmosphere radioactivity monitors.

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

Each monitor is provided with two isokinetic nozzles to ensure that representative samples are obtained for both normal purge and minipurge flow rates. The sample is extracted through the selected nozzle and then passed through the selector valve, the fixed filter (particulate), charcoal filter (iodine), and fixed-volume gaseous detectors. The sample then passes through the pumping system and is discharged back to the duct.

Indication is provided for each monitor on individual indicators on the radioactivity monitoring system control panel and, through isolated signals, on the radioactivity monitoring system CRT in the control room.

The Containment Atmosphere Radioactivity monitors, continuously monitor the containment atmosphere for particulate, iodine, and gaseous radioactivity. They isolate the containment purge system on high gaseous activity via the ESFAS. These monitors also serve for reactor coolant pressure boundary leakage detection and for personnel protection. The containment atmosphere radioactivity monitors provide backup indication for the containment purge monitors.

Samples are extracted from the operating deck level (El. 2047'-6") through sample lines which penetrate the containment. The monitors are located as close as possible to the containment penetrations to minimize the length of the sample tubing and the effects of sample plate out. The sample points are located in areas which ensure that representative samples are obtained. Each sample passes through the penetration, then through the fixed filter (particulate), charcoal filter (iodine), and fixed-volume gaseous detector assemblies. After passing through the pumping system, the sample is discharged back to the containment through a separate penetration.

Indication is provided for each monitor on individual indicators on the radioactivity monitoring system control panel and, through isolated signals, on the radioactivity monitoring system CRT in the control room.

The Radwaste Building Vent monitors are described in Section 3.3.1.

The batch gaseous effluent monitors setpoints are normally established using Xe-133 as the controlling isotope, with the exception of the Radwaste Building Vent monitor, for which Kr-85 is normally the controlling isotope.

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

3.4 ODCM Methodology for the Determination of Gaseous Effluent Monitor Setpoints

3.4.1 Development of ODCM Methodology for the Determination of Gaseous Effluent Monitor Setpoints

The alarm/trip setpoint for gaseous effluent monitors is determined based on the lesser of the whole body dose rate and skin dose rate, as calculated for the unrestricted site boundary.

3.4.1.1 Whole Body Dose Rate Setpoint Calculations

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

$$S_{wb} \leq D_{wb} R_{wb} F_s F_a \tag{3.1}$$

Where:

S_{wb} = the response of the gaseous effluent noble gas monitor at the alarm/trip setpoint based on the whole dose rate.

D_{wb} = Radiological Effluents 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_{wb} = (μCi/cc) per (mrem/yr) to the whole body, determined according to:

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

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 Table 4.11-2 of the Radiological Effluent Technical Specifications. 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 unrestricted area boundary, in (sec/m^3).

K_i = the whole body dose factor due to gamma emissions for each identified noble gas radionuclide, is (mrem/yr per μCi/m^3). (Table 3)

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

3.4.1.2 Skin Dose Rate Setpoint Calculation

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

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

Where:

F_s and F_a are as previously defined.

S_s = the response of the gaseous effluent noble gas monitor at the alarm/trip setpoint based on the skin dose rate.

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

R_s = (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 gase 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.

3.4.1.3 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	Whole 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+03
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 Table B-1 in Reg. Guide 1.109 and are for detected in gaseous effluents.

3.4.2 Summary, Gaseous Effluent Monitors Setpoint Determination

The gaseous effluent monitors setpoints are calculated 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 ODCM Methodology for Determining Dose Contributions From Gaseous Effluents

Dose rate calculations are performed for gaseous effluents to ensure compliance with Radiological Effluent Technical Specification 3.11.2.1 as stated in Section 3.2.

3.5.1 Determination of Dose Rate

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

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 general relationships delineated as follows:

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

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

Where:

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

K_i = Whole 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 unrestricted area boundary.

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

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 all radionuclides and radioactive materials in particulate form and radionuclides other than noble gases is determined according to:

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

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)

$(\overline{X/Q})$ and Q_i are 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.[1] These values were calculated according to:

$$P_i = K' \text{ (BR) DFA:} \tag{3.8}$$

Where:

K' = Units conversion factor: 1μCi = 1E06pCi.

BR= The breathing rate of the child age group, in (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 whole body is considered as an organ in the selection of DFA_i. (Regulatory Guide 1.109, Table E-9)[1]

Note: All radioiodines are assumed to be released in elemental form.

TABLE 4

Dose Parameter (P_i) For Radionuclides Other Than Noble Gases[a]

RADIONUCLIDE	P_i Inhalation Pathway[b] (mrem/yr) per (μCi/m³)
H-3	1.12 E + 03
C-14	3.59 E + 04
Cr-51	1.70 E + 04
Mn-54	1.58 E + 06
Fe-59	1.27 E + 06
Co-58	1.11 E + 06
Co-60	7.07 E + 06
Zn-65	9.95 E + 05
Sr-89	2.16 E + 06
Sr-90	1.01 E + 08
Zr-95	2.23 E + 06
Mo-99	1.35 E + 05
I-131	1.62 E + 07
I-133	3.85 E + 06
I-135	7.92 E + 05
Cs-134	1.01 E + 06
Cs-136	1.71 E + 05
Cs-137	9.06 E + 05
Ba-140	1.74 E + 06
La-140	2.26 E + 05
Ce-141	5.44 E + 05
Ce-144	1.20 E + 07

(a) dose parameters listed are for radionuclides that may be detected in gaseous effluents. Additional dose parameters not listed may be determined using methodology described in NUREG-0133.

(b) the childs age group determination; Table E-9, Reg. Guide 1.109, Rev. 1, 1977

3.5.2 Individual Dose Due To Gaseous Effluents

3.5.2.1 Radiological Effluent Technical Specification 3.11.2.2

The air dose due to noble gases released in gaseous effluents from the site 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 in unrestricted areas due to noble gases released from the site is determined according to the following methodology:

During any calendar quarter, for gamma radiation:

$$D_g = 3.17 \times 10^{-8} \sum_i [M_i \{(\overline{X/Q})\, Q_i + (X/q)\, q_i\}] \leq 5 \text{ mrad} \qquad (3.9)$$

During any calendar quarter, for beta radiation:

$$D_b = 3.17 \times 10^{-8} \sum_i [N_i \{(\overline{X/Q})\, Q_i + (X/q)\, q_i\}] \leq 10 \text{ mrad} \qquad (3.10)$$

During any calendar year, for gamma radiation:

$$D_g = 3.17 \times 10^{-8} \sum_i [M_i \{(\overline{X/Q})\, Q_i + (X/q)\, q_i\}] \leq 10 \text{ mrad} \qquad (3.11)$$

During any calendar year, for beta radiation:

$$D_b = 3.17 \times 10^{-8} \sum_i [N_i \{(\overline{X/Q})\, Q_i + (X/q)\, q_i\}] \leq 20 \text{ mrad} \qquad (3.12)$$

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 unrestricted area boundary for short term releases (equal to or less than 500 hrs/year).

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/m^3). (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 unrestricted area boundary for long-term releases (greater than 500 hrs/yr).

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 Radiological Effluent Technical Specification 3.11.2.3

The dose to an individual from radionuclides and radioactive materials in particulate form, and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents released from the site shall be limited to the following:

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 an individual from radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days in gaseous effluents released to unrestricted areas is determined by 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 an individual from radionuclides other than noble gases.

Q_i = The releases of radionuclides, 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 radionuclides, 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 pathway, in(sec/m^3).

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

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}).

3.17×10^{-8} = The inverse of the number of seconds per year.

$(\overline{D/Q})$ = the average relative deposition of the effluent at the unrestricted area 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 the unrestricted area 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.

TABLE 5

PATHWAY DOSE FACTORS (R_i), FOR RADIONUCLIDES OTHER THAN NOBLE GASES

	Inhalation Pathway (mrem/yr) per (μCi/m^3)		Ground Plane Pathway (m^2 mrem/yr) per (μCi/sec)		Grass-Cow-Milk Pathway (m^2 mrem/yr) per (μCi/sec)		Meat Pathway (m^2 mrem/yr) per (μCi/sec)	Vegetation Pathway (m^2 mrem/yr) per (μCi/sec)
Radionuclide	Infant	Child	Whole Body	Skin	Infant	Child	Child	Child
H-3(χ/Q)	6.47E+02	1.12E+03	---	---	2.38E+03	1.57E+03	2.34E+02	4.01E+03
C-14	2.65E+04	3.59E+04	---	---	2.34E+09	1.20E+09	3.84E+08	8.91E+08
Cr-51	1.28E+04	1.70E+04	4.65E+06	5.50E+06	4.69E+06	5.39E+06	3.26E+05	6.21E+06
Mn-54	1.00E+06	1.58E+06	1.39E+06	1.63E+09	3.89E+07	2.09E+07	8.02E+06	6.65E+08
Fe-59	1.02E+06	1.27E+06	2.72E+08	3.20E+08	3.93E+08	2.03E+08	4.46E+08	6.68E+08
Co-58	7.77E+05	1.11E+06	3.79E+08	4.44E+08	6.05E+07	7.07E+07	9.59E+07	3.77E+08
Co-60	4.51E+06	7.07E+06	2.15E+10	2.53E+10	2.10E+08	2.40E+08	3.84E+08	2.09E+09
Zn-65	6.47E+05	9.95E+05	7.47E+08	8.59E+08	1.90E+10	1.10E+10	9.99E+08	2.16E+09
Sr-89	2.03E+06	2.16E+06	2.16E+04	2.51E+04	1.26E+10	6.62E+09	4.82E+08	3.60E+10
Sr-90	4.09E+07	1.01E+08	---	---	1.22E+11	1.12E+11	1.04E+10	1.24E+12
Zr-95	1.75E+06	2.23E+06	2.45E+08	2.84E+08	8.27E+05	8.80E+05	6.12E+08	8.85E+08
I-131	1.48E+07	1.62E+07	1.72E+07	2.09E+07	1.05E+12	4.34E+11	5.58E+09	4.76E+10
I-133	3.56E+06	3.85E+06	2.45E+06	2.98E+06	9.63E+09	3.78E+09	8.75E+01	8.27E+08
I-135	6.96E+05	7.92E+05	2.51E+06	2.93E+06	1.97E+07	8.49E+06	---	1.05E+07
Cs-134	7.03E+05	1.01E+06	6.86E+09	8.00E+09	6.81E+10	3.72E+10	1.51E+09	2.63E+10
Cs-136	1.35E+05	1.71E+05	1.53E+08	1.74E+08	5.80E+09	2.77E+09	4.42E+07	2.24E+08
Cs-137	6.12E+05	9.06E+05	1.03E+10	1.20E+10	6.04E+10	3.23E+10	1.34E+09	2.39E+10
Ba-140	1.60E+06	1.74E+06	2.05E+07	2.35E+07	2.42E+08	1.18E+08	4.43E+07	2.79E+08
La-140	1.68E+05	2.26E+05	1.92E+07	2.18E+07	1.88E+05	1.90E+05	5.52E+02	3.20E+07
Ce-141	5.17E+05	5.44E+05	1.37E+07	1.54E+07	1.37E+07	1.36E+07	1.38E+07	4.08E+08
Ce-144	9.84E+06	1.20E+07	6.96E+07	8.04E+07	1.33E+08	1.32E+08	1.89E+08	1.04E+10
Mo-99	1.35E+05	1.35E+05	3.98E+06	4.62E+06	3.10E+08	1.74E+08	2.44E+05	1.65E+07

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 Radiological Effluent Technical Specification 3.11.2.4

The Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be OPERABLE. The appropriate portions of the gaseous radwaste treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air dose due to gaseous effluent releases from the site when averaged over the calendar quarter, would exceed 0.6 mrad for gamma radiation and 1.2 mrad for beta radiation. The appropriate portions of the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site when averaged over the calendar quarter would exceed 0.9 mrem to any organ.

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) gas decay storage 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. To determine operability requirements, doses due to gas release are projected once each 31 days in accordance with the methodology described in Section 3.5.2.

4.0 DOSE AND DOSE COMMITMENT FROM URANIUM FUEL CYCLE SOURCES

4.1 Radiological Effluent Technical Specification 3.11.4

The dose or dose commitment to any member of the public 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) over 12 consecutive months.

4.2 ODCM Methodology for Determining Dose and Dose Commitment from Uranium Fuel Cycle Sources

For the purpose of calculating the annual population-integrated dose within a 50 mile radius of the site, the whole body, skin and thyroid are considered the critical organs. Units of man-rem apply to the whole body and skin dose, and units of man-thyroid-rem apply to the summation of thyroid dose.

4.2.1 Calculation of Total Dose

The cumulative dose to a member of the public due to radioactive releases from the site can be determined by summing the calculated doses to critical organs (whole body, skin and thyroid) from effluent sources and methodology previously discussed. The methodology presented in Equations (2.13) and (2.14) is used to determine the annual dose to members of the public from liquid effluent releases. Through the use of the methodology presented in Equations (3.9), (3.10), (3.11), (3.12), and (3.13), the annual dose to members of the public from gaseous effluents is determined.

4.2.1.1 Calculation of Total Dose from Noble Gas Effluents

4.2.1.1.1 Dose to the whole body is calculated according to:

$$D_{wb} = 3.17\ E\text{-}08 \sum_i K_i\ [(\overline{X/Q})Q_i + (X/q)q_i] \leq 25 \text{ mrem} \quad (4.1)$$

Where:

D_{wb} = whole body dose

K_i, $(\overline{X/Q})$, (X/q), Q_i, and q_i are previsouly defined (refer to Sections 3.5.1.1 and 3.5.2.1.1).

4.2.1.1.2 Dose to the skin is calculated according to:

$$D_s = 3.17\ E\text{-}08 \sum_i (L_i + 1.1\ M_i)[(X\overline{/Q})Q_i + (X/q)q_i] \leq 25 \text{ mrem} \quad (4.2)$$

Where:

D_s = skin dose

L_i, M_i, $(\overline{X/Q})$, (X/q), Q_i, and q_i are as previously defined (refer to Sections 3.5.1.1 and 3.5.2.1.1).

4.2.1.2 Calculation of Total Dose from Radionuclides Other Than Noble Gases

4.2.1.2.1 Dose to the thyroid is calculated according to:

$$D_{th} = 3.17\ E\text{-}08 \sum_i R_i\ [WQ_i + wq_i] \leq 75 \text{ mrem} \quad (4.3)$$

Where:

D_{th} = thyroid dose

R_i, W, w, Q_i, and q_i are as previously defined (refer to Section 3.5.2.2.1)

4.2.1.3 Summary, Determination of Total Dose

The methodology developed in Equations (2.13) and (2.14) is used to determine the annual dose to members

of the public from liquid effluent releases. Equations (4.1), (4.2), and (4.3) are used to determine the annual dose to members of the public from gaseous effluents.

No attempt has been made to include the dose reduction due to shielding provided by residential structures in the development of Equations (4.1) and (4.2). A shielding factor of 0.7 was used in the calculation of the Ground Plane Pathway Factor (R_i), presented in Table 3. The annual average relative concentration ($\overline{X/Q}$) and the average relative deposition rate ($\overline{D/Q}$) are at the approximate receptor location in lieu of the site boundary for these calculations.

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 Table 3.12-1. (ODCM Table 6)

5.2 Description of the Radiological Environmental Monitoring Program

The environmental monitoring program is intended to act as a background data base for pre-operation 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 basis. Efforts will be made to obtain all required samples within time frame outlines. Any deviation(s) in sampling frequency or location will be 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 outlined in Tables 7 and 8. Figures 5.1 through 5.5 provide locations of samples required in Table 6.

TABLE 6

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE Radioiodine and Particulates	5	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. Particulate Sampler. Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION	40 (2 dosimeters for continuously measuring and recoridng dose rate at each location.)	At least one per 31 days.	Gamma Dose. At least once per 31 days.

** Sample locations are given on Figures 5.1-5.5.

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	2	Composite sample collected over a period of ≤ 31 days.	Gamma isotopic analysis of each sample. Tritium analysis of sample at least once per 92 days.
b. Ground	2	At least one per 92 days.	Gamma isotopic and tritium analyses of each sample.
c. Drinking	1*	Grab sample collected and composited at least once per 31 days.	Gamma isotopic and Gross Beta analyses of each sample. Tritium analyses of sample at least once per 92 days.
d. Sediment from Shoreline	1	At least once per 184 days.	Gamma isotopic analysis of each sample.
4. INGESTION			
a. Milk	4	At least once per 15 days when animals are on Pasture; at least once per 31 days at other times.	Gamma isotopic and I-131 analysis of each sample.

* No drinking water intake within 50 miles of plant discharge. Sample collected from the City of St. Louis, MO water intake.

** Sample locations are given on Figures 5.1-5.5.

TABLE 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
b. Fish and Invertebrates	2	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following. 1. Bottom Feeder Species 2. Predator Species	Gamma isotopic analysis on edible portions.
c. Food Products	2	At time of harvest. Sample from each location of broad leaf vegation.	Gamma isotopic and I-131 analyses of each sample.

** Sample locations are given on Figures 5.1-5.5.

TABLE 7

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg), wet	Milk (pCi/l)	Food Product (pCi/kg, wet)
H-3	2×10^4 (a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zr-Nb-95	4×10^2 (b)				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2 (b)			3×10^2 (b)	

(a) For drinking water samples. Values are from 40 CFR 141.

(b) Total for parent and daughter.

TABLE 8

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

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
Gross Beta	4[b]	1×10^{-2}				
^{3}H	2000 (1000[b])					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{95}Zr-Nb	15[c]					
^{131}I	1[b]	7×10^{-2}		1	60	
$^{134,137}Cs$	15 (10[b]), 18	1×10^{-2}	130	15	60	150
^{140}Ba-La	15[c]			15[c]		

TABLE 8 (CONTINUED)

TABLE NOTATION

(a) The LLD is the smallest concentration of radioactive material in a sample 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 transformation).

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

2.22 = the number of transformation per minute per picocurie.

Y = the fractional radiochemical yield (when applicable).

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

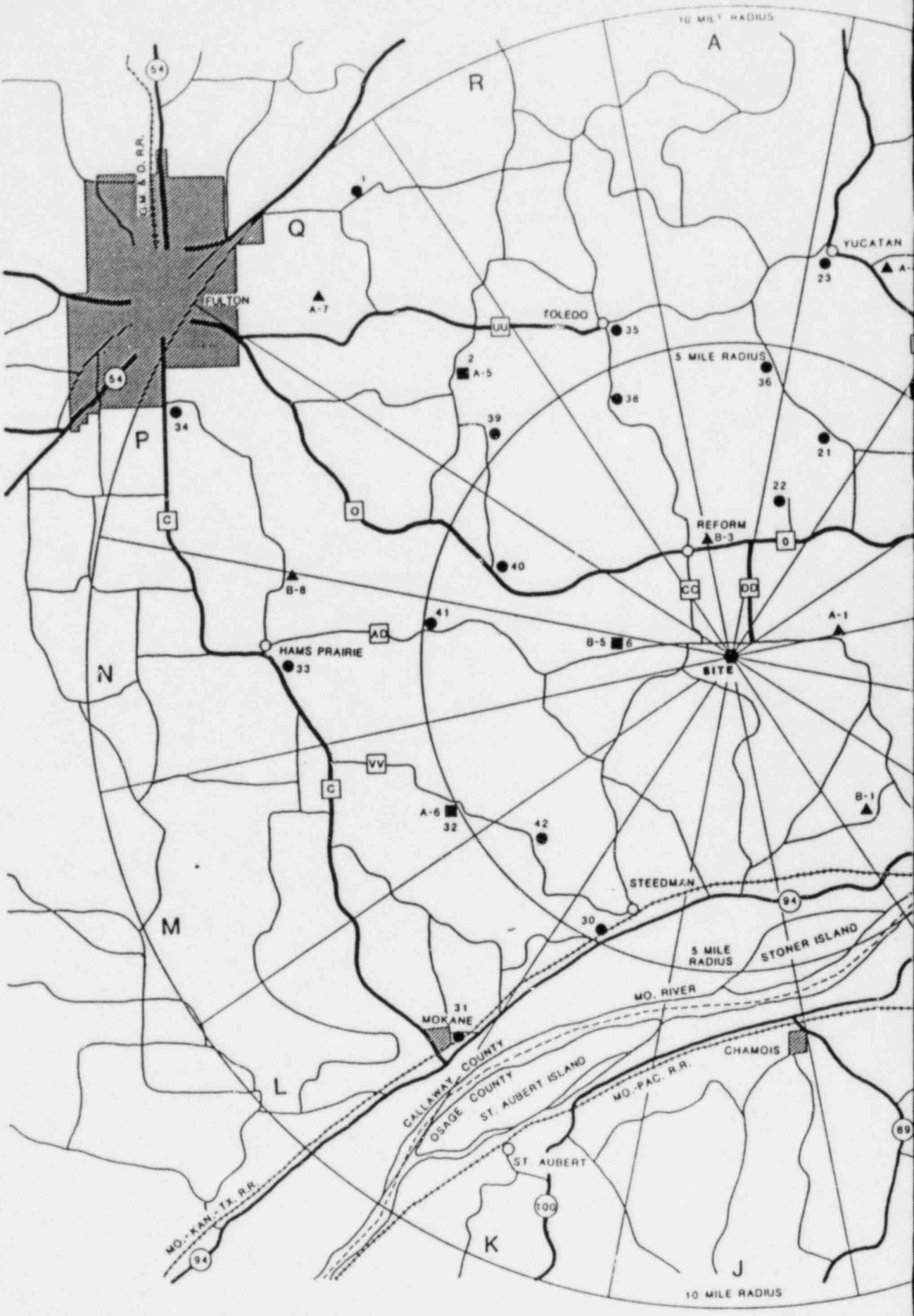
Δt = the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of S_b used in the calculation of the LLD for a detection system is based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background includes the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

(b) LLD for drinking water.

(c) Total for parent and daughter.

(d) Other peaks which are measurable and identifiable, together with the radionuclides in Table 8, are identified and reported.


10 MILE RADIUS
A
R
Q
P
N
M
L
K
J
FULTON
C.M. & O. R.R.
TOLEDO
YUCATAN
5 MILE RADIUS
REFORM
B-3
SITE
HAMS PRAIRIE
STEEDMAN
STONER ISLAND
MO. RIVER
MOKANE
CHAMOIS
CALLAWAY COUNTY
OSAGE COUNTY
ST. AUBERT ISLAND
MO.-PAC. R.R.
ST. AUBERT
MO.-KAN.-TX. R.R.
A-7
A-5
B-8
B-5
A-1
B-1
A-6
54
94
89
100
UU
O
C
AD
VV
CC
DD
0
1
2
6
21
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30
31
32
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34
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36
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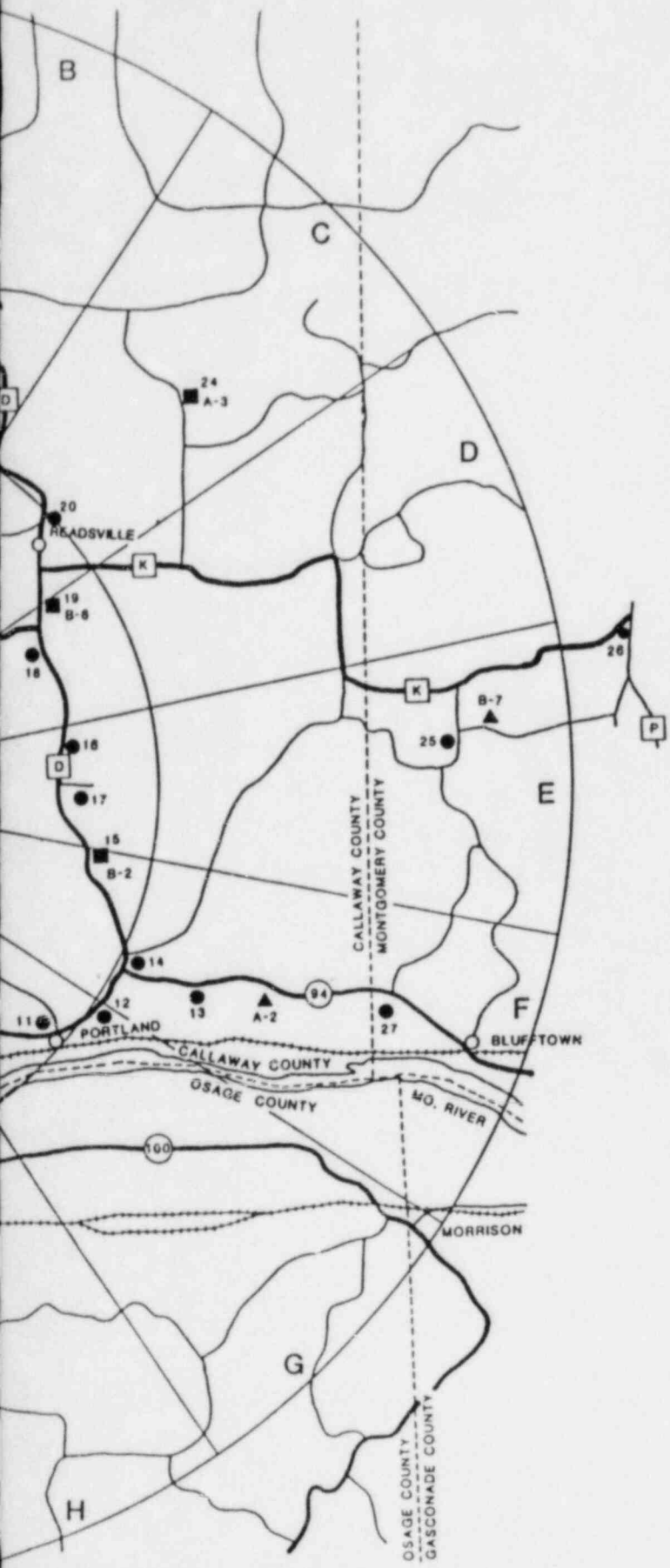
39
40
41
42



N

LEGEND:

- INCORPORATED COMMUNITIES
- UNINCORPORATED COMMUNITIES
- AIR SAMPLER LOCATIONS
- TLD LOCATIONS
- TLD & AIR SAMPLER LOCATIONS

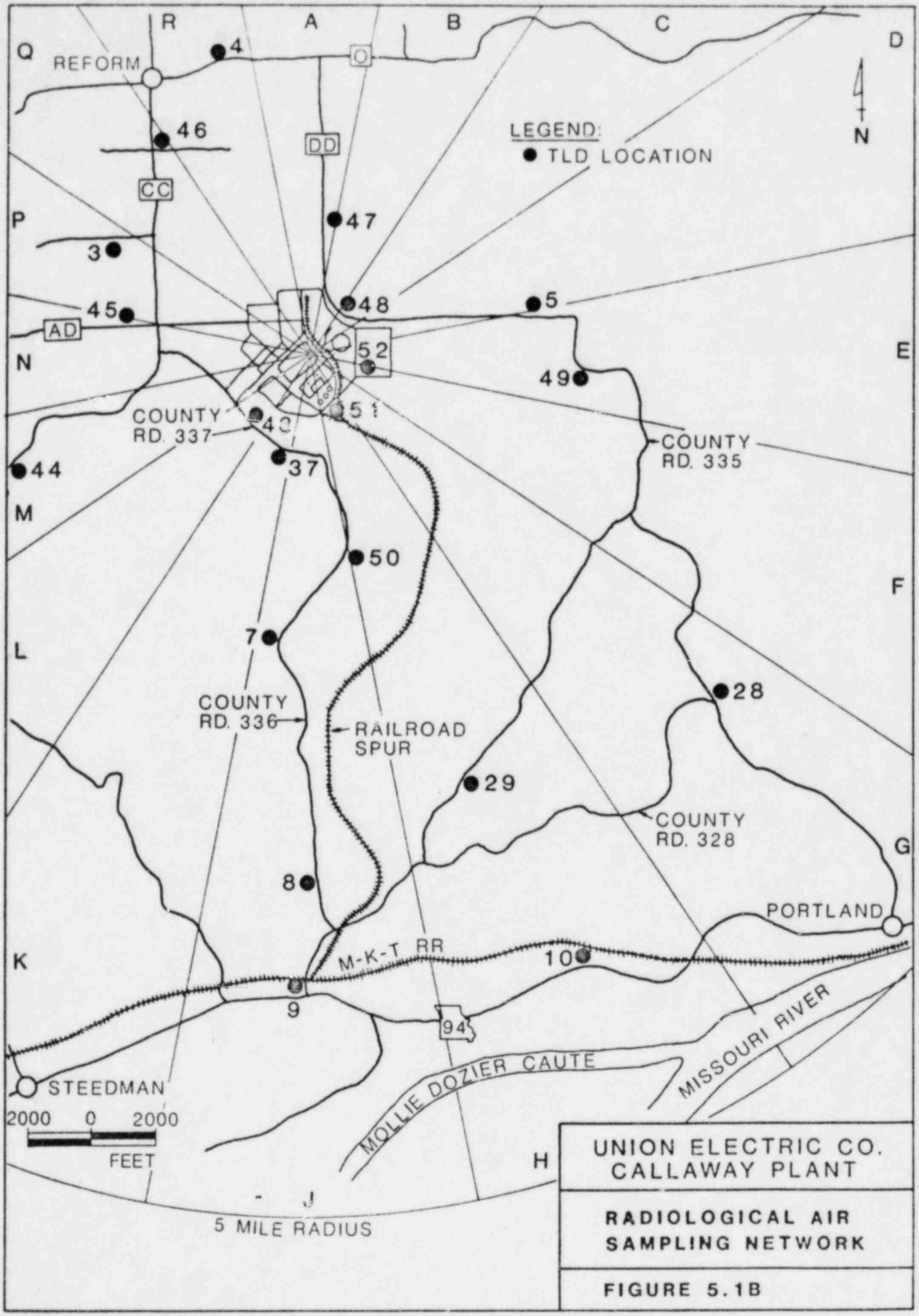
REFERENCE:
THIS MAP WAS PREPARED FROM A PORTION OF THE FOLLOWING U.S.G.S. MAP: ST. LOUIS, MO., 1962.

1 0 1 2 3
MILES

UNION ELECTRIC CO.
CALLAWAY PLANT

RADIOLOGICAL AIR SAMPLING NETWORK

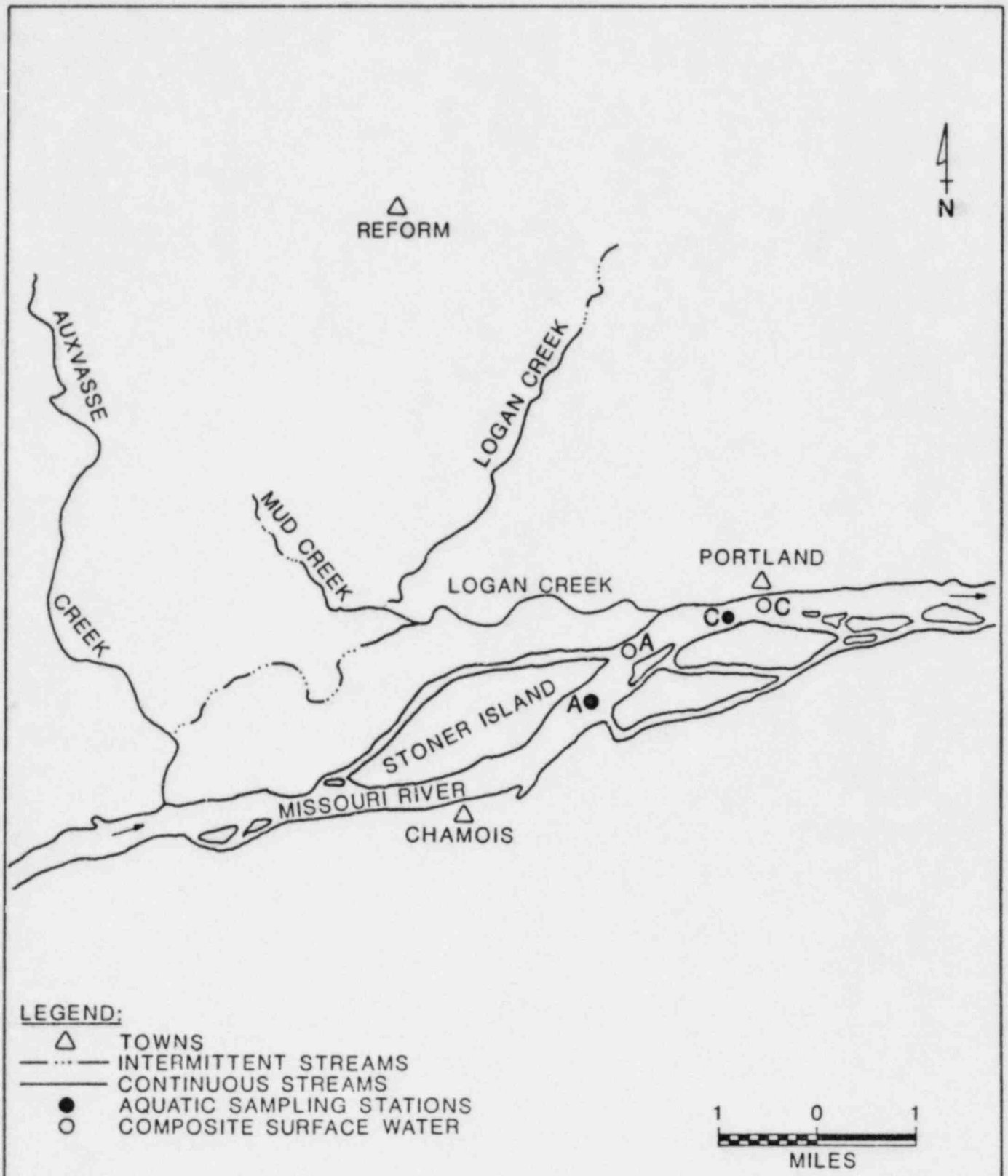
FIGURE 5.1A



UNION ELECTRIC CO.
CALLAWAY PLANT

RADIOLOGICAL AIR
SAMPLING NETWORK

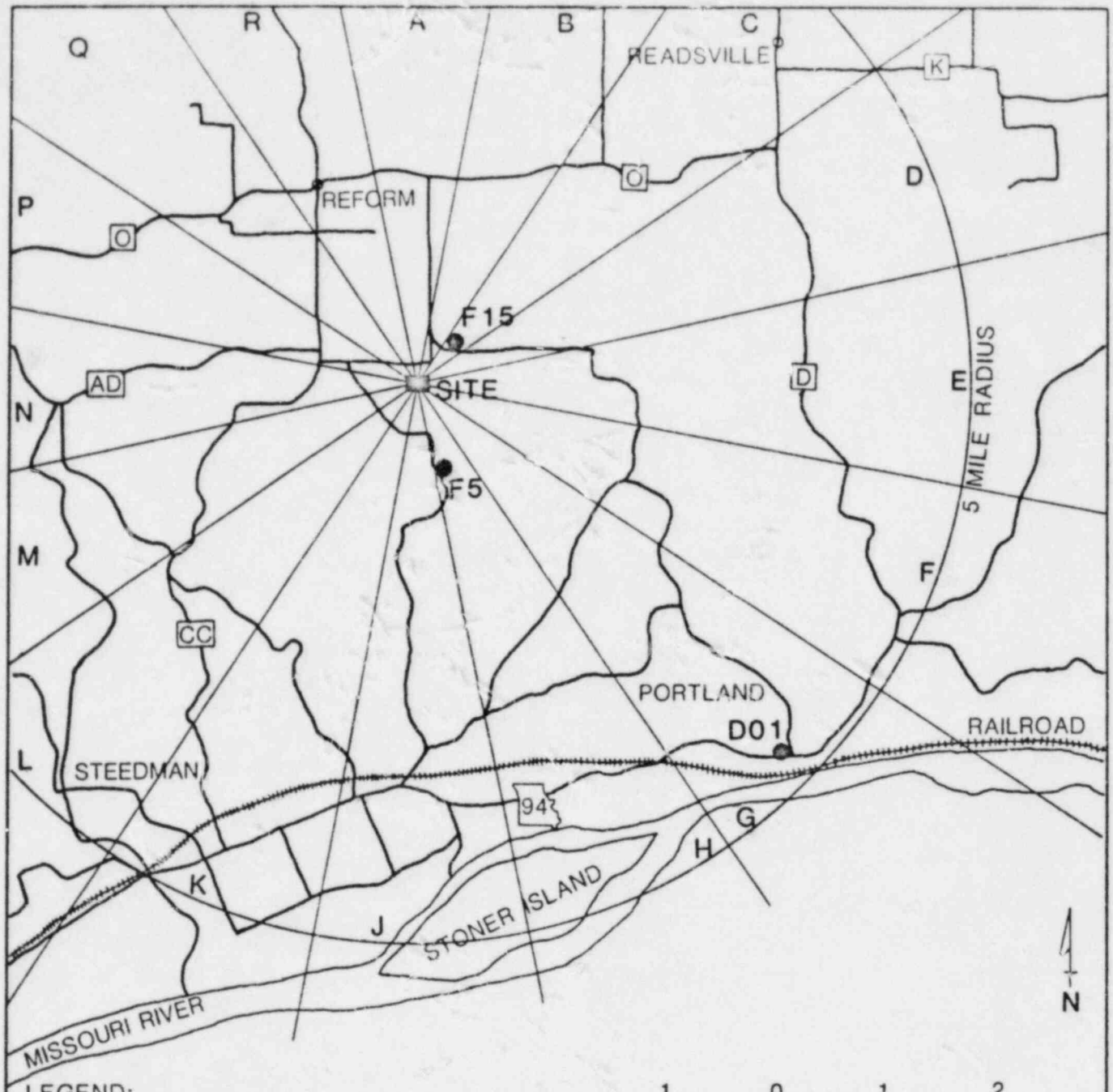
FIGURE 5.1B



UNION ELECTRIC CO.
CALLAWAY PLANT

LOCATION OF AQUATIC SAMPLING STATIONS

FIGURE 5.2



LEGEND:
●- WELL SAMPLING LOCATION
F-GROUND WATER
D-DRINKING WATER

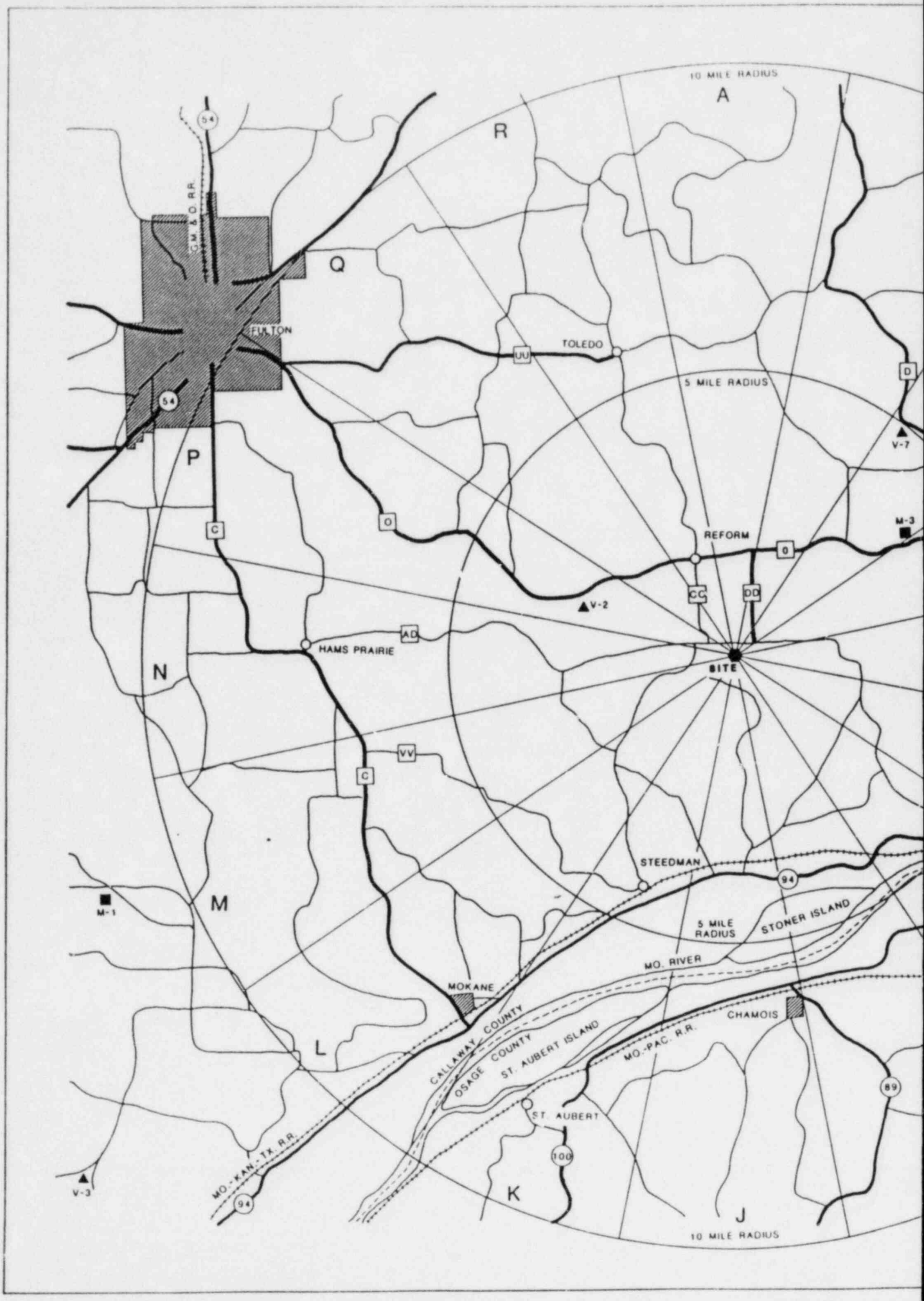
1 0 1 2
MILES

REFERENCE:
MISSOURI STATE HIGHWAY DEPARTMENT, GENERAL HIGHWAY MAP CALLAWAY COUNTY, MISSOURI, OCTOBER 1, 1977.

UNION ELECTRIC CO.
CALLAWAY PLANT

GROUNDWATER QUALITY MONITORING LOCATIONS

FIGURE 5.3


10 MILE RADIUS
A
R
Q
P
N
M
L
K
J
54
GM & O. R.R.
FULTON
TOLEDO
UU
5 MILE RADIUS
D
V-7
M-3
REFORM
O
C
CC
DD
V-2
AD
HAMS PRAIRIE
SITE
VV
STEEDMAN
94
M-1
5 MILE RADIUS
STONER ISLAND
MO. RIVER
MOKANE
CHAMOIS
CALLAWAY COUNTY
OSAGE COUNTY
ST. AUBERT ISLAND
MO.-PAC. R.R.
89
ST. AUBERT
100
MO.-KAN.-TX. R.R.
V-3
10 MILE RADIUS

N

LEGEND:

- INCORPORATED COMMUNITIES
- UNINCORPORATED COMMUNITIES
- MILK
- FOOD PRODUCT

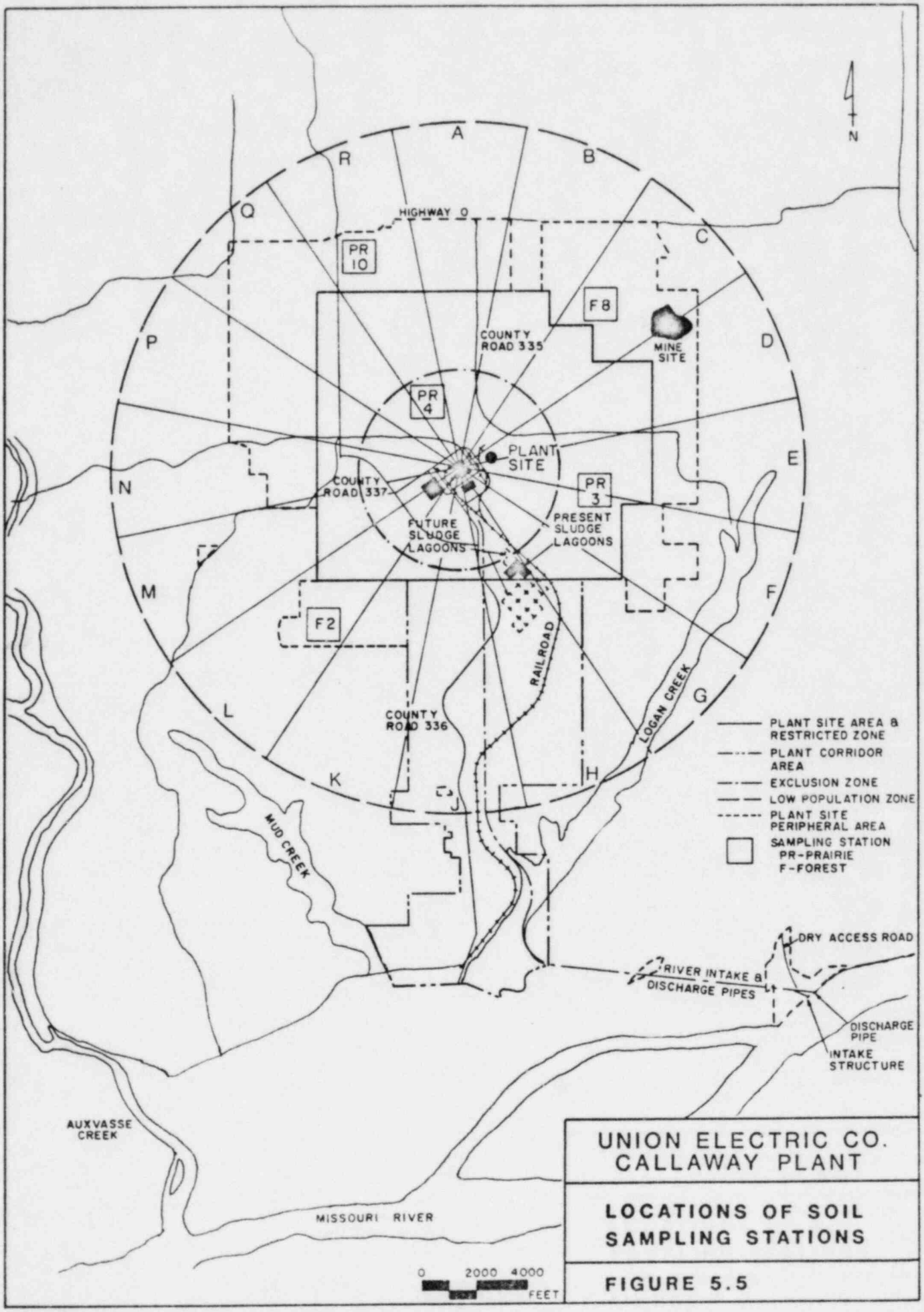
REFERENCE:
THIS MAP WAS PREPARED FROM A PORTION OF THE FOLLOWING U.S.G.S. MAP: ST. LOUIS, MO., 1962.

1 0 1 2 3
MILES

UNION ELECTRIC CO.
CALLAWAY PLANT

FOOD PRODUCTS & MILK SAMPLING LOCATIONS

FIGURE 5.4



UNION ELECTRIC CO.
CALLAWAY PLANT

LOCATIONS OF SOIL SAMPLING STATIONS

FIGURE 5.5

6.0 METEOROLOGICAL DATA COLLECTION AND PROCESSING

6.1 Meterological Data Collection

The Meterological System has been established to meet the requirements of NUREG-0654 and Regulatory Guide 1.23. A 90 meter primary and a 10 meter backup tower have been erected in the vicinity of the plant site.

The primary meterological tower is located in an open field approximately 1.4 miles east-northeast of the plant. The backup tower is located approximately 1 mile west of the plant, adjacent to the EOF. Meterological instruments are mounted on the primary meterological tower at heights of 10, 60, and 90 meters and on the backup tower at 10 meters. These instruments monitor wind speed, wind direction, reference temperature, temperature differential, precipitation, and dew point.

A readout of these parameters is available in the Control Room, TSC, and the EOF. Primary tower recorder equipment is housed in a shelter located near the tower and the backup tower equipment is located in the Communications Equipment Room of the EOF. Meterological parameters, along with radiological monitoring and ventilation system flows, are input to the Radioactive Release Information System.

There is an emergency electric generator for both tower locations in the event of power failure, and the National Weather Service in Columbia, Missouri is available for backup meterological data.

6.2 Meterological Data Processing

Real time windspeed, wind direction, atmospheric stability, and precipitation data is processed and stored for generation of the reports required by Regulatory Guide 1.21 and for calculation of the various relative atmospheric concentrations (X/Q) and relative deposition (D/Q).

X/Q (undepleted and undecayed, undepleted and decayed, and depleted and decayed) is calculated for each 22.5° sector (as defined in NUREG 0654) at 21 distances. The determination of (X/Q) employs the constant mean wind direction model with building wake correction and assumes a ground level release. Radioactive decay and dry deposition are considered during plume transport.

The value of D/Q is determined by solving polynomial regression equations for each of the deposition curves

in Regulatory Guide 1.111 as a function of height of release, stability class, and distance.

The methodology for calculating X/Q and D/Q is consistent with that outlined in Regulatory Guide 1.111.

7.0 SEMI ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

Regulatory Guide 1.21 requires that potential doses to individuals and populations are calculated using measured effluent and meteorological data. Each Semi Annual Radioactive Effluent Release Report contains the following information:

* Whole body and significant organ doses to individuals in unrestricted areas from receiving-water-related exposure pathways.

* Whole body and skin doses to individuals exposed at the point of maximum offsite ground-level concentrations of radioactive materials in gaseous effluents.

* Organ doses to individuals in unrestricted areas from radioactive material in particulate form from all pathways of exposure.

* Whole body doses to individuals and populations in unrestricted areas from direct radiation from the facility.

* Whole body doses to the population and average doses to individuals in the population from all receiving-water-related pathways.

* Whole body doses to the population and average doses to individuals in the population from gaseous effluents to a distance of 50 miles from the site.

The specific pathways to man from airborne releases are:

* Plume exposure
* Ground exposure
* Inhalation
* Cow-milk pathway
* Vegetable pathway
* Meat pathway
* Goat-milk pathway

The specific pathways to man from liquid releases are:

* Aquatic food chain
* Dose from shoreline deposits
* Dose from swimming
* Dose from boating
* Drinking of potable water (if applicable; refer to Section 2.5.2)

The age groups considered are:

* Adult
* Teen
* Child
* Infant

The organ doses are:

* Bone
* Liver
* Total body
* Thyroid
* Kidney
* Lung
* GI-LLI

Dose calculations for the Semi-Annual Radioactive Effluent Release Report are performed consistent with the previously defined methodologies, utilizing the dose commitment factors obtained from Regulatory Guide 1.109.

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 have been developed which employ Regulatory Guide 1.109 calculational techniques to implement the ODCM methodologies. These calculational methods include the same general features as provided in the ODCM. These codes will be verified to produce results consistent with the ODCM methodologies.

9.0 REFERENCES

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

9.2 Callaway Radiological Effluent Technical Specifications, Sections 3.3.3.9, 3.3.3.10, 3/4.11 and 3/4.12, as submitted to the U.S. Nuclear Regulator Commission, May 1982.

9.3 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.

9.4 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.

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

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

9.7 Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, August 1975.

9.8 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, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, June 1974.

9.9 U.S. Nuclear Regulatory Commission, "Criteria For Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in

Support of Nuclear Power Plants", USNRC NUREG-0654, Revision 1, Appendix 2, Washington, D.C. 20555, November 1980.

9.10 U.S. Nuclear Regulatory Commission, "Final Environmental Statement Related to the Operation of Callaway Plant, Unit No. 1", USNRC NUREG-0813, Section 5.9, Washington, D.C. 20555, January 1982.

9.11 Final Safety Analysis Report, Standardized Nuclear Unit Power Plant Systems (SNUPPS), Sections 11 and 12.

9.12 Final Safety Analysis Report, Site Addendum, Sections 2, 11, and 12.

9.13 Regulatory Guide 1.23, "Onsite Meteorology Program," Revision 0, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, February, 1972.